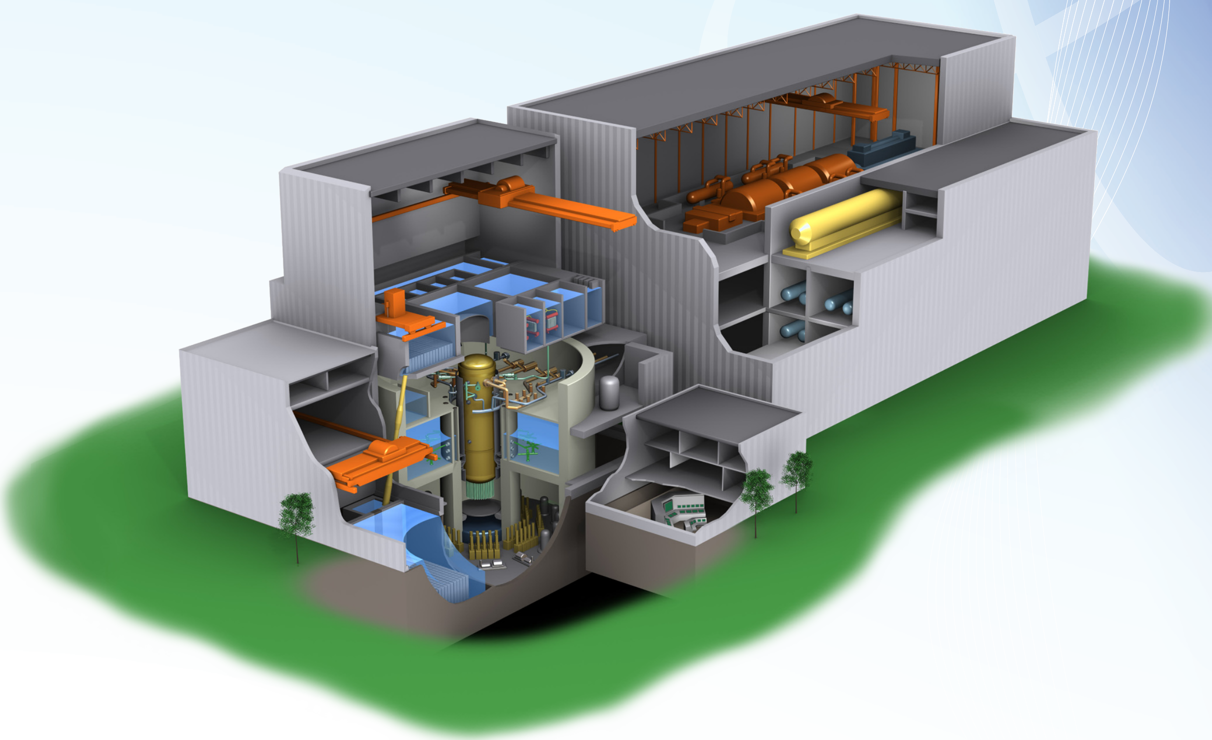
GE Hitachi Nuclear Energy

The ESBWR Plant General Description



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ESBWR
Plant General Description

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Table of Contents

Balance of Plant

Chapter 9 Major Balance of Plant Features

Chapter 10 Radioactive Waste Systems

Evaluations

Chapter 11 Safety Evaluations

Appendices

List of Figures

List of Tables

Acronyms

ABWR	Advanced Boiling Water Reactor
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
AFIP	Automated Fixed In-Core Probe
AHS	Auxiliary Heat Sink
AHU	Air Handling Unit
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
APR	Automatic Power Regulator System
APRM	Average Power Range Monitor
ARM	Area Radiation Monitoring
ARI	Alternate Rod Insertion
ASD	Adjustable Speed Drive
ASHRAE	American Society of Heating, Refrigerating and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
AST	Alternate Source Term
ATIP	Automatic Traversing In-Core Probe
ATLM	Automatic Thermal Limit Monitor
ATP	Authorization to Proceed
ATWS	Anticipated Transient Without Scram
BAF	Bottom of Active Fuel
BiMAC	Basemat-internal Melt Arrest Coolability
BOP	Balance of Plant
BWR	Boiling Water Reactor
C&FS	Condensate and Feedwater System
CB	Control Building
C/C	Cooling and Clean-Up
CCC	Control Cell Core
CCFP	Contingent Containment Failure Probability
CDF	Core Damage Frequency
CIRC	Circulating Water System
CIV	Combined Intermediate Valves
CLAVS	Clean Air Ventilation Subsystem
CMS	Containment Monitoring System
COE	Cost of Electricity
COL	Combined Operating License
CONAVS	Controlled Area Ventilation Subsystem
CP	Construction Permit/Control Processor
CPR	Critical Power Ratio
CPS	Condensate Purification System
CRB	Control Rod Blade
CRD	Control Rod Drive
CRDH	Control Rod Drive Housing
CRDHS	Control Rod Drive Hydraulic System
CRGT	Control Rod Guide Tube
CRHA	Control Room Habitability Area
CRHAVS	CRHA HVAC Subsystem
CRT	Cathode Ray Tube
CST	Condensate Storage Tank
CWS	Chilled Water System
DAW	Dry Active Waste
DBA	Design Basis Accident
DC	Direct Current
DCIS	Distributed Control & Information System
DCPS	DC Power Supply
DCS	Drywell Cooling System
DCV	Drywell Connecting Vent
DG	Diesel Generator
DMC	Digital Measurement Controller
DoE	Department of Energy
DPS	Diverse Protection System
DPV	Depressurization Valve
DW	Drywell
DZO	Depleted Zinc Oxide
EAB	Exclusion Area Boundary
EB	Electrical Building
ECCS	Emergency Core Cooling System
ECP	Electrochemical Potential
ECW	Emergency Chilled Water
EDG	Emergency Diesel Generator
EHC	Electro-hydraulic Control (Turbine Control System)
EFU	Emergency Filter Unit
EMI	Electro-Magnetic Interference
EMS	Essential Multiplexing System
EOF	Emergency Operations Facility
EPD	Electrical Power Distribution
EPRI	Electric Power Research Institute
ESF	Essential Safeguards Feature
FAPCS	Fuel and Auxiliary Pool Cooling System

FB	Fuel Building
FCU	Fan Cooling Unit
FDA	Final Design Approval
FFTR	Final Feedwater Temperature Reduction
FIV	Flow-Induced Vibration
FMCRD	Fine Motion Control Rod Drive
FOAKE	First-of-a-Kind Engineering
FPS	Fire Protection System
FSAR	Final Safety Analysis Report
FSC	First Structural Concrete
FTDC	Fault Tolerant Digital Controller
FW	Feedwater
FWC	Feedwater Control System
FWP	Feedwater Pump
FWTI	Feedwater Temperature Increase
FWTR	Feedwater Temperature Reduction
FWP	Feedwater Pump
GDCS	Gravity Driven Cooling System
GE	General Electric Company
GEH	GE Hitachi Nuclear Energy
GETAB	General Electric Thermal Analysis Basis
GIST	GDCS Integral System Test
GNF	Global Nuclear Fuel
GPM	Gallons per minute
GT	Gamma Thermometer
HCU	Hydraulic Control Unit
HCW	High-Conductivity Waste
HEPA	High Efficiency Particulate Air
HFE	Human Factor Engineering
HFF	Hollow Fiber Filter
HIC	High Integrity Container
HPCP	High Pressure Condensate Pump
HPNSS	High Pressure Nitrogen Supply System
HSI	Human System Interface
HVAC	Heating, Ventilation and Air-Conditioning
HWC	Hydrogen Water Chemistry
HX	Heat Exchanger
I&C	Instrumentation and Control
IASCC	Irradiation-Assisted Stress Corrosion Cracking
ICPR	Initial Critical Power Ratio
IC	Isolation Condenser
ICP	Independent Control Platform
ICS	Isolation Condenser System
IDIF	ICS/DPV Isolation Function
IEEE	Institute of Electrical and Electronic Engineers
IFTS	Inclined Fuel Transfer System
IGSCC	Intergranular Stress Corrosion Cracking
ILRT	Integrated Leak Rate Test
IMC	Induction Motor Controller
IMS	Information Management System
IRM	Intermediate Range Monitor
ISI	In-Service Inspection
KRB	Kernkraftwerke Gundremmingen Betriebsgesellschaft, unit A
LA	Low Activity
LCW	Low Conductivity Waste
LD	Laundry Drain
LDW	Lower Drywell
LD&IS	Leak Detection and Isolation System
LFCV	Low Flow Control Valve
LHGR	Linear Heat Generation Rate
LLRT	Local Leak Rate Test
LOCA	Loss-of-Coolant Accident
LOFW	Loss of Feedwater
LOOP	Loss of Offsite Power
LOPP	Loss of Preferred Power
LPCI	Low-Pressure Coolant Injection
LPCP	Low-Pressure Condensate Pump
LPCRD	Locking Piston Control Rod Drive
LPRM	Local Power Range Monitor
LPZ	Low Population Zone
LTP	Lower Tie Plate
LU	Logic Unit
LWMS	Liquid Radwaste Management System
MCC	Main Control Console or Motor Control Center
MCES	Main Condenser Evacuation System
MCOPS	Manual Containment Overpressure Protection System
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
M-G	Motor-Generator
MITI	Ministry of International Trade and Industry (Japan)
MLHGR	Maximum Linear Heat Generation Rate
MMI	Man-Machine Interface
MO	Motor-Operated
MOD	Motor-Operated Disconnect
MOV	Motor-Operated Valve
MRBM	Multi-Channel Rod Block Monitoring System
MS	Main Steam Subsystem

MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSR	Moisture Separator Reheater
MUX	Multiplexer
MWS	Makeup Water System
NBS	Nuclear Boiler System
NDT	Nil Ductility Temperature
N-DCIS	Nonsafety-Related DCIS
NEMS	Non-Essential Multiplexing System
NI	Nuclear Island
NMO	Nitrogen Motor Operated (Valve)
NMS	Neutron Monitoring System
NO	Nitrogen (piston) Operated (Valve)
NPHS	Normal Power Heat Sink
NRC	Nuclear Regulatory Commission
NRHX	Non-Regenerative Heat Exchanger
NSS	Nuclear Steam Supply
NSSP	Non-safety Surveillance Panel
NSSS	Nuclear Steam Supply System
O&M	Operation and Maintenance
OGS	Offgas System
OPRM	Oscillation Power Range Monitor
PARS	Passive Autocatalytic Recombiners
PAS	Plant Automation System
PCCS	Passive Containment Cooling System
PCI	Pellet Clad Interaction
PCS	Plant Computer System
PCT	Peak Fuel Clad Temperature
PCV	Primary Containment Volume
PG	Power Generation (loads)
PGCS	Power Generation Control System
PIP	Plant Investment Protection (loads)
PIP	Position Indicator Probe
PLR	Part-Length Fuel Rod
PRA	Probabilistic Risk Assessment
PRMS	Process Radiation Monitoring System
PRNM	Power Range Neutron Monitor System
PSWS	Plant Service Water System
PWR	Pressurized Water Reactor
Q-DCIS	Safety-related DCIS
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RBC	Rod Brake Controller
RC&IS	Rod Control and Information System
RCCV	Reinforced Concrete Containment Vessel
RCCWS	Reactor Component Cooling Water System
RCIC	Reactor Core Isolation System
RCPB	Reactor Coolant Pressure Boundary
RDT	Resistance Temperature Detector
RG	Regulatory Guide
RHR	Residual Heat Removal
RHX	Regenerative Heat Exchanger
RMS	Radiation Monitoring Subsystem
RMU	Remote Multiplexer Unit
RO	Reverse Osmosis
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSS	Remote Shutdown System
RTD	Resistance Temperature Detector
RTIF	Reactor Trip Isolation Function(s)
RTNDT	Reference Temperature Nil Ductility Transition
RTNSS	Regulatory Treatment of Non-Safety Systems
RW	Radwaste Building
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling System
RWM	Rod Worth Minimizer
S&PC	Steam and Power Conversion System
SA	Severe Accident
SAR	Safety Analysis Report
SB	Service Building
SB&PC	Steam Bypass and Pressure Control System
SBO	Station Blackout
SBWR	Simplified Boiling Water Reactor
SCRI	Select Control Rod Insert
SDV	Scram Discharge Volume
SIM	Simulator
SJAE	Steam Jet Air Ejector
SLCS	Standby Liquid Control System
SOE	Sequence of Events
SP	Suppression Pool
SPC	Suppression Pool Cooling
SPDS	Safety Parameter Display System
SRM	Source Range Monitor
SRNM	Startup Range Neutron Monitor
SRV	Safety/Relief Valve
SSAR	Standard Safety Analysis Report
SSC	Shift Supervisor's Console
SSE	Safe Shutdown Earthquake
SSLC	Safety System Logic and Control

SSPV	Scram Solenoid Pilot Valve
SWMS	Solid Waste Management System
TAF	Top of Active Fuel
TB	Turbine Building
TBCE	Turbine Building Compartment Exhaust
TBS	Turbine Bypass System
TBV	Turbine Bypass Valve
TCCWS	Turbine Component Cooling Water System
TCS	Turbine Control System
TCV	Turbine Control Valve
TEDE	Total Effective Dose Equivalent
TEPCO	Tokyo Electric Power Company
TGSS	Turbine Gland Steam System
TIP	Traversing In-Core Probe
TIU	Technician Interface Unit
TLU	Trip Logic Unit
TMSS	Turbine Main Steam System
TPC	Taiwan Power Company
TRA	Transient Recording and Analysis
TSC	Technical Support Center
UAT	Unit Auxiliary Transformer
UDW	Upper Drywell
UHS	Ultimate Heat Sink
UPS	Uninterruptable Power Supply
URD	Utility Requirements Document
UTP	Upper Tie Plate
V&V	Verification and Validation
VAC	Volts-Alternating Current
VB	Vacuum Breaker
VBIF	Vacuum Breaker Isolation Function
VDC	Volts-Direct Current
VDU	Video Display Unit
WDP	Wide Display Panel
WW	Wetwell

Chapter 1 Introduction

Nuclear Energy for the New Millennium

Nuclear energy plays a major role in meeting the world's energy needs. Currently, there are 440 nuclear power plants operating in 30 countries, with 59 more units under construction. These plants account for 17% of the world's electricity. The industry remains dynamic, as evidenced by the fact that several new plants enter commercial operation every year and there are typically 30 or more in various stages of construction at any given time.

Generating electricity with nuclear energy permits economic and social development to be sustainable; that is, not limited by encroaching environmental concerns. A non-nuclear, baseload power plant generates electricity by burning fossil fuels day in and day out and releasing the by-products to the environment. A nuclear plant, on the other hand, generates large amounts of electricity with virtually no impact on the environment. In quantitative terms, if the world's nuclear plants were replaced with coal-fired plants, global CO_2 emissions would increase by 8% every year. This would amount to 1,600 million tons per year at a time when the world is trying to reduce emissions by 4,200 million tons per year. Similarly, if the world's growing appetite for new electricity is met without nuclear energy playing a key role, CO_2 emissions would quickly rise to levels that curtail economic growth.

The ESBWR advanced nuclear plant will play an important role in meeting the conflicting needs of developed and developing economies for massive amounts of new electricity and the need worldwide to limit CO_2 emissions. It continues to use advanced technologies first applied in the Advanced Boiling Water Reactor (ABWR) with simplifications in the recirculation system and ECCS. Four ABWRs have been constructed in Japan, and Taiwan is constructing two more ABWRs. Other countries have similar strategies to deploy advanced nuclear plants, and the successful deployment of ABWRs in Japan and Taiwan, coupled with international agreements to limit CO_2 emissions, will only reinforce these plans.

The ESBWR represents an entirely new approach to the way nuclear plant projects are undertaken, modeled after the successful process used for ABWR. The ABWR was licensed and designed in detail before construction ever began. Once construction did begin, it proceeded smoothly from start to finish in less than four years for the lead units in Japan.

The successful design, licensing, construction, and operation of the ESBWR nuclear power plant will usher in a new era of safe, economic, and environmentally-friendly nuclear electricity. The ESBWR is the first of a new generation of nuclear plants equipped with advanced technologies and features that raise plant safety to new levels that significantly improve the economic competitiveness of this form of generation.

Fifty Years in the Making

The Boiling Water Reactor (BWR) nuclear plant, like the Pressurized Water Reactor (PWR), has its origins in the technology developed in the 1950s for the U.S. Navy's nuclear submarine program. The first BWR nuclear plant to be built was the 5 MWe

Vallecitos plant (1957) located near San Jose, California. The Vallecitos plant confirmed the ability of the BWR concept to successfully and safely produce electricity for a grid. The first large-scale BWR, Dresden 1 (1960), then followed. The BWR design has subsequently undergone a series of evolutionary changes with one purpose in mind—simplify.

The BWR design has been simplified in two key areas—the reactor systems and the containment design. Table 1-1 chronicles the development of the BWR.

Dresden 1 was based upon a dual steam cycle, not the direct steam cycle that currently characterizes BWRs. Steam was generated in the reactor but then flowed to an elevated steam drum and a secondary steam generator before making its way to the turbine.

The first step down the path of simplicity that led ultimately to the ESBWR was the elimination of the external steam drum by introducing two technical innovations—the internal steam separator and dryer (KRB, 1962). This practice of simplifying the design with technical innovations was to be repeated over and over.

The first large direct-cycle BWRs (e.g., Oyster Creek) appeared in the mid-1960s and were characterized by the elimination of the steam generators and the use of five external recirculation loops. Later, reactor systems were further simplified by the introduction of internal jet pumps. These pumps sufficiently boosted recirculation flow so that only two external recirculation loops were needed. This change first appeared in the Dresden-2 BWR/3 plant.

Product Line	**First Commercial Operation Date**	**Representative Plant/ Characteristics**
BWR/1	1960	Dresden 1 Initial commercial-size BWR
BWR/2	1969	Oyster Creek Plants purchased solely on economics Large direct cycle
BWR/3	1971	Dresden 2 First jet pump application Improved ECCS: spray and flood capability
BWR/4	1972	Vermont Yankee Increased power density (20%)
BWR/5	1977	Tokai 2 Improved ECCS Valve flow control
BWR/6	1978	Clinton Compact control room Solid-state nuclear system protection system
ABWR	1996	Kashiwazaki-Kariwa 6 Reactor internal pumps Fine-motion control rod drives Advanced control room, digital and fiber optic technology Improved ECCS: high/low pressure flooders
ESBWR	-	Natural circulation Passive ECCS

Table 1-1. Evolution of the GE BWR

The use of reactor internal pumps in the ABWR design took this process of simplification another step. By using pumps attached directly to the vessel itself, the jet pumps and the external recirculation systems, with all their pumps, valves, piping, and snubbers, have been eliminated altogether.

The ESBWR, and its smaller predecessor, the SBWR, took the process of simplification to its logical conclusion with the use of a taller vessel and a shorter core to achieve natural recirculation without the use of any pumps. Figure 1-1 illustrates the evolution of the reactor system design.

The first BWR containments were spherical "dry" structures. Dry containments in spherical and cylindrical shape are still used today in PWR designs. The BWR, however, quickly moved to the "pressure suppression" containment design with a suppression pool for its many advantages. Among these are:

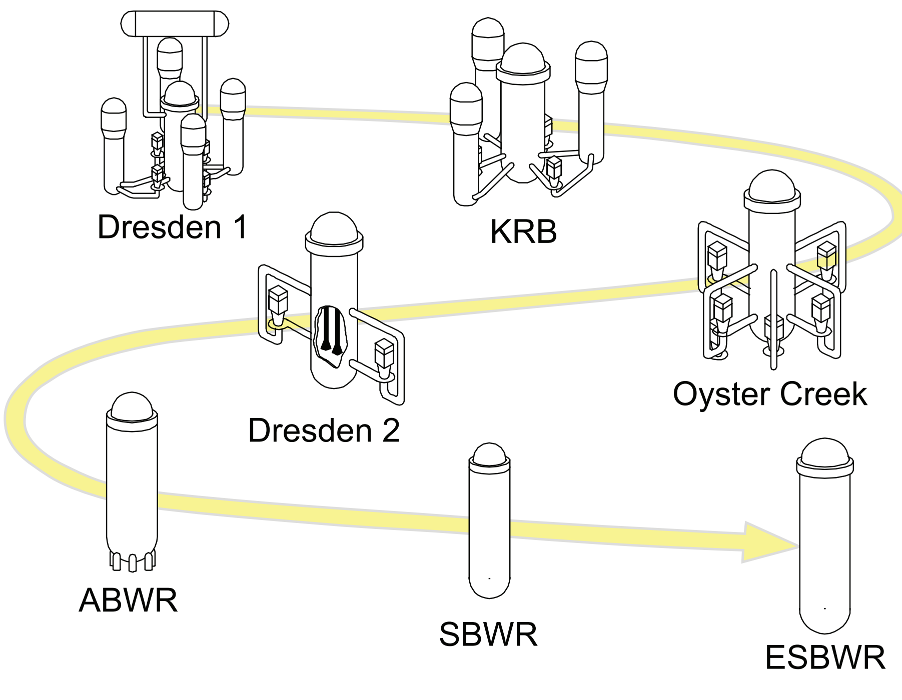

Figure 1-1. Evolution of the Reactor System Design

- High heat capacity
- Lower design pressure
- Superior ability to accommodate rapid depressurization
- Unique ability to filter and retain fission products
- Provision of a large source of readily available makeup water in the case of accidents
- Simplified, compact design

It is the reduction in containment design pressures, together with the elimination of the external recirculation loops, that allows the containment (and, by extension, the reactor building) to be more compact.

The Mark I containment was the first of the new containment designs. The Mark I design has a characteristic light bulb configuration for the reinforced concrete drywell, surrounded by a steel torus that houses a large water pressure suppression pool. The conical Mark II design has a less-complicated arrangement. A key feature is the large containment drywell that provides more room for the steam and ECCS piping. The Mark III containment design, used worldwide with BWR/6s and some BWR/5s, represented a major improvement in simplicity. Its containment structure is a right-circular cylinder that is easy to construct, and provides ready access to equipment and ample space for maintenance activities. Other features of the Mark III include horizontal vents to reduce overall loss-of-coolant accident (LOCA) dynamic loads and a freestanding all-steel structure to ensure leak-tightness.

The ABWR containment is significantly smaller than the Mark III containment because the elimination of the recirculation loops translates into a significantly more compact containment and reactor building. The structure itself is made of reinforced concrete with a steel liner from which it derives its name—RCCV, or reinforced concrete containment vessel. The ESBWR containment is similar in construction to the ABWR, but slightly larger to accommodate the passive ECCS systems.

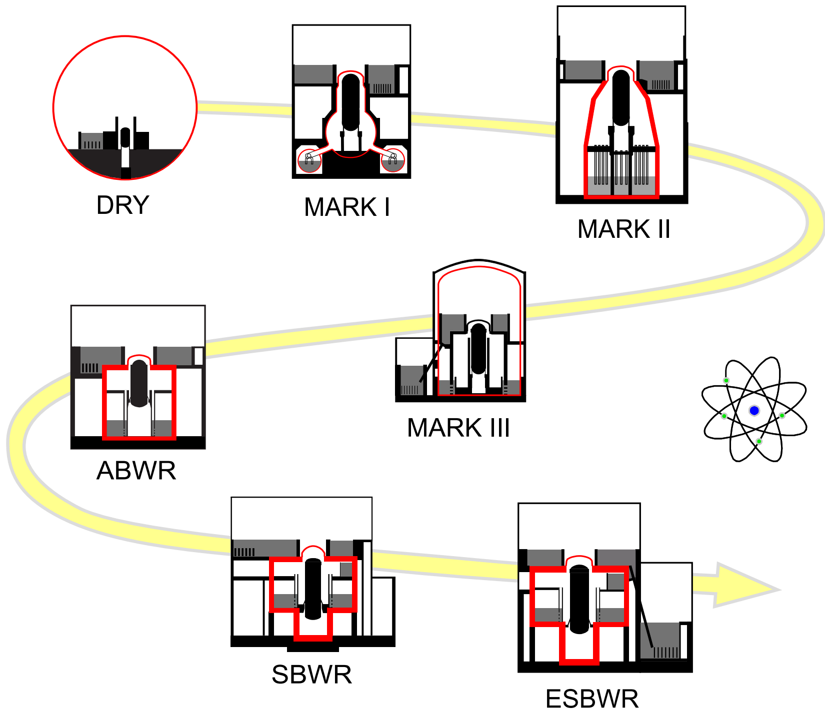

Figure 1-2. Evolution of the Containment Design

internal study of a new BWR concept based on these principles and the Simplified Boiling Water Reactor (SBWR) was born in the early 1980s. This concept attracted development support from the U. S. Department of Energy (DOE), Electric Power Research Institute (EPRI) and a number of US Utilities. Key new features, such as the Gravity Driven Core Cooling System (GDCS), Depressurization Valves (DPV), and leak-tight wetwell/drywell vacuum breakers were tested. As interest grew, an International Team was formed to complete the design, and additional separate effects, component and integrated system tests, particularly of the innovative new feature, the Passive Containment Cooling System (PCCS), were run in Europe and Japan.

A Design Certification Program was started in the late 1980s with the objective of obtaining a standardized license, similar to that obtained for the ABWR. However, as more of the design details became known, it became clear that, at 670 MWe, the SBWR was too small to be economically competitive with other utility options for electrical generation. The certification program was stopped, but GE continued to look for ways to make an SBWR attractive for power generation. With European Utility support, the SBWR was uprated gradually to its current power level of approximately 1520 MWe. This was made possible by staying within the Reactor Pressure Vessel (RPV) size limit established by the ABWR, and by taking advantage of the modular approach to passive safety afforded by Isolation Condensers (IC) and PCCS.

The ESBWR has achieved its basic plant simplification by using innovative adaptations of operating plant systems, e.g., combining shutdown cooling and reactor water cleanup systems, and combining the various pool cooling and cleanup systems. In addition, several systems were eliminated, e.g., standby gas treatment and flammability control. There is a high confidence that the design is proven because of the following basic approach to the design:

Figure 1-2 illustrates the evolution of the BWR containment from the earliest versions to today's ESBWR RCCV design.

There are approximately 90 BWRs, including four ABWRs, worldwide. Many are among the best operating plants in the world, performing in the "best of class" category. Numerous countries rely heavily upon BWR plants to meet their needs for electricity. In the United States, there are 35 operating BWRs.

To date, the ABWR plant is the only advanced nuclear plant in operation.

ESBWR Development and Design Approach

Following the Three Mile Island accident in 1979, there was a lot of interest in developing a reactor with passive safety features and less dependence on operator actions. Utilities also took this opportunity to request a reactor which was simpler to operate, had fewer components and no dependence on diesel-generators for safety actions. GE began an

- Utilize BWR features that have been successfully used before in operating BWRs, (e.g., natural circulation, isolation condensers)
- Utilize standard systems where practical, e.g. utilize features common to ABWR - vessel size, fine motion control rod drives, pressure suppression containment, fuel designs, materials and chemistry
- Extend the range of data to ESBWR parameters, e.g., separators, large channel two phase flow, isolation condensers (IC)
- Perform extensive separate effects, component and integral tests at different scales for the PCCS
- Test any new components, e.g., squib actuated DPVs, IC heat exchangers, wetwell/drywell vacuum breakers.

The ESBWR program, as a result, inherited a technologically rich legacy of design, development, and analysis work passed along from the SBWR and ABWR programs. Some systems required duty or rating up-sizing to adjust to a higher power level. Other systems needed an addition of yet another duplicate equipment train. Plant electrical (even though significantly simplified), cooling water, and heat cycle systems benefited tremendously from the on-going systems work underway on all of GE's ABWR design activities

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Related Projects Worldwide

Operating ABWRs in Japan

Four ABWR units in Japan are now constructed and fully operational. Two of these units are located at TEPCO's Kashiwazaki-Kariwa site, 100 miles north of Tokyo on the Sea of Japan. The world's first advanced nuclear plant, Unit 6, began commercial operation in 1996. Unit 7, the second ABWR, followed shortly thereafter with commercial operation commencing in 1997.

Both TEPCO ABWR units were constructed in world record times. From first concrete to fuel load, it took just 36.5 months to construct Unit 6 and 38. 3 months for Unit 7, the former being 10 months less than the best time achieved for any of the previous BWRs constructed in Japan. In addition, both units were built within budget, which is an impressive record of performance, since these were first-of-a-kind units.

Figure 1-3. Kashiwasaki Units 6 & 7

Two more ABWRs are now operational in Japan—Hamaoka-5, which began commercial operation in January, 2005; and Shika-2, which was connected to the grid in July, 2005, and achieved commercial operation in March, 2006.

Both TEPCO units have completed many cycles of operation. By all measures, these ABWRs have lived up to their promise. Other than regulatory mandated outages and seismic events, both plants have operated essentially at full power for each fuel cycle. The thermal efficiency of the plant is 35%, slightly higher than previous designs. See Figure 1-3 for a photo of the Kashiwazaki Units 6 & 7.

The ABWR in the United States

The ABWR was the first plant to use the new standard plant licensing process in the U.S. (10CFR50.52). The efforts of the NRC and GE came to fruition in 1997 when the ABWR Design Certification was signed into law. This was rightly hailed by the U.S. industry as a significant accomplishment, one that has been envisioned for a long time—pre-approval of a standard design of an advanced nuclear plant.

The ABWR in Taiwan

Two more ABWRs are being constructed for the Taiwan Power Company (TPC) at TPC's Lungmen site, located on the Pacific Ocean about 40 miles northeast of Taipei.

Commercial operation of Lungmen Unit 1 is expected to begin in early 2012. The schedule for Unit 2, including the start of commercial operation, is about one year later.

ESBWR Status

The Design Certification application for the ESBWR was submitted to the U.S. Nuclear Regulatory Commission (NRC) in August 2005, and is nearing completion of its technical review. The Safety Evaluation Report was issued in March 2011, with rule making for design certification to be issued later that year.

The ESBWR concept has been designed to higher levels of safety, including being designed to prevent and mitigate the consequences of a Severe Accident. Also, recent concerns and regulations, such as airplane crash and cyber security, are addressed in the design.

The ESBWR design will be captured electronically using the latest state-of-the-art information management technology. The benefits appear not only in construction, where it has been shown over and over with fossil plants that use of this engineering tool reduces construction time and cost, but also during the operation and maintenance of the plant. Nuclear plants today are constructed much differently than in the past. Design simplifications and the use of new construction technologies and techniques make this possible.

Of course, there is no substitute for experience. GEH and its alliance partner, HGNE, have proven construction experience, including application of state-of-the-art civil and equipment modularization techniques. This experience allows projects to proceed with high certainty in construction costs and schedule.

Chapter 2 Plant Overview

ESBWR Program Goals

The ESBWR builds on the very successful Advanced Boiling Water Reactor (ABWR) technology and construction programs, as well as the Simplified Boiling Water Reactor (SBWR) development program. The key design objectives for the ABWR were established during its development program. The key goals, all of which were achieved, are as follows:

- Design life of 60 years
- Plant availability factor of 87% or greater
- Less than one unplanned scram per year
- 18- to 24-month refueling interval
- Operating personnel radiation exposure limit <1 Sv/year
- Reduced calculated core damage frequency by at least a factor of 10 over previous BWRs (goal $<10^{-6}$/yr)
- Radwaste generation less than that of the 10% best operating BWRs
- 48-month construction schedule
- 20% reduction in capital cost ($/kWh) vs. previous 1,100 MWe class BWRs

To these objectives, the following additional goals were established for ESBWR:

- All Essential Safeguards Features (ESF) shall be passive, eliminating the need for safety grade diesel generators
- Following design basis events, no operator action shall be required for 72 hours
- Shortened construction schedule
- Cost advantage over competing baseload electrical generating technologies
- Plant availability target of 95%

Summary of the ESBWR Key Features

A cutaway rendering of the ESBWR plant (Figure 2-1) illustrates the general configuration of the plant for a single unit site in the U.S. Shown in the foreground are the Reactor and Fuel Buildings, and in the background is the Turbine Building. In front of the Reactor Building is the Control Building. A comparison of key features of the ESBWR to previous models is shown in Table 2-1.

An artist's rendering of the major systems and how they are interconnected is shown in Figure 2-2. This shows the reactor, ECCS, containment, turbine equipment and the key auxiliary mechanical systems.

Design Philosophy

Recognizing the desire for simplification of the typically complex safety systems with attendant cost, quality assurance requirements, and technical specifications, the ESBWR has adopted passive safety systems, together with a natural circulation primary system.

By shortening the active fuel length, adding an approximately 9-m tall chimney above the core, and lengthening the reactor vessel, the ESBWR elimi-

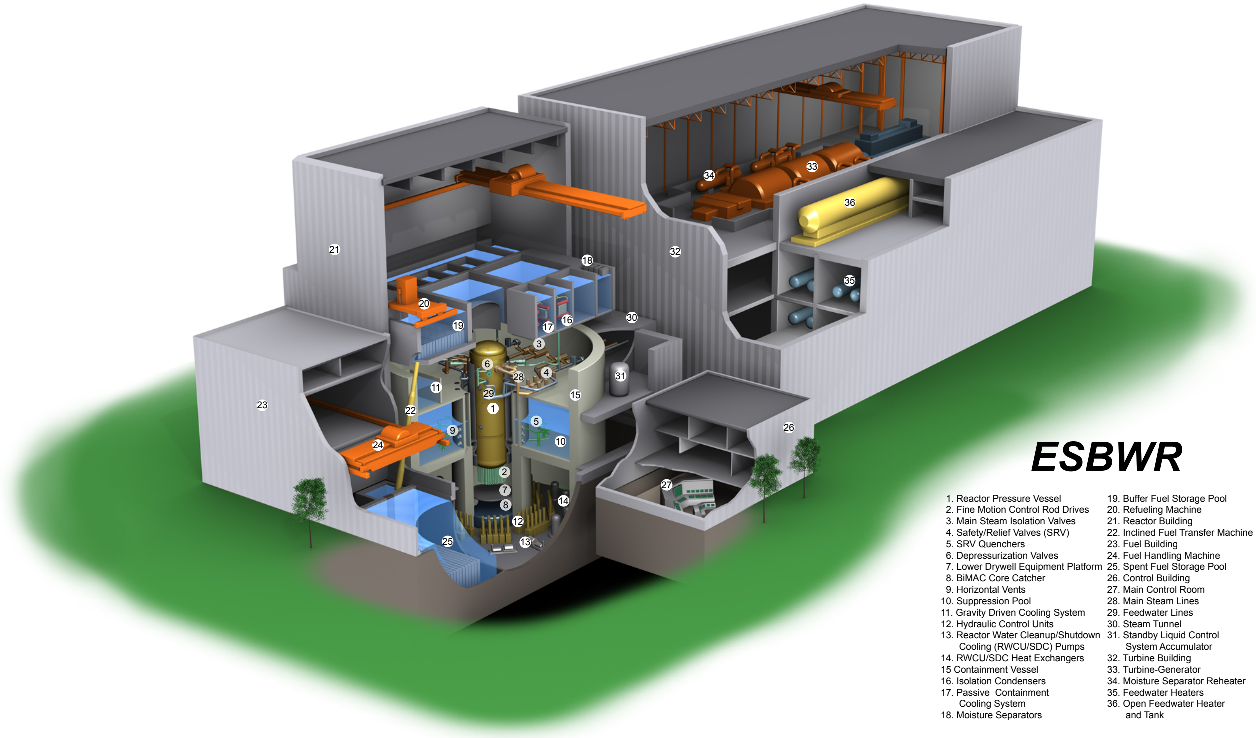


Figure 2-1. Cutaway Rendering of the ESBWR

nated the recirculation system, relying completely on natural circulation for core flow (see Figure 2-3).

High pressure inventory control and heat removal is accomplished with the use of isolation condensers if the reactor becomes isolated from the normal heat sink.

The reactor can also be depressurized rapidly to allow multiple sources of non-safety systems to provide makeup. However, the ultimate safety features are passive, both for core flooding as well as for containment heat removal.

Response to Anticipated Transient Without Scram (ATWS) is improved by the adoption of Fine-Motion Control Rod Drives (FMCRDs), which allow reactor shutdown either by hydraulic or electric insertion. In addition, the need for rapid operator action to mitigate an ATWS is avoided by automation of emergency procedures such as feedwater runback and passive Standby Liquid Control System (SLCS) injection from borated water stored in pressurized accumulators.

Calculated core damage frequency is reduced by more than a factor of fifty relative to the BWR/6 design and five relative to the ABWR. Furthermore, the ESBWR also improved the capability to mitigate severe accidents, even though such events are extremely unlikely. Through nitrogen inerting and the addition of Passive Autocatalytic Recombiners (PARS), containment integrity threats from hydrogen detonation were eliminated. Sufficient spreading area in the lower drywell, together with a drywell flooding system and a core catcher located under the Reactor Pressure Vessel (RPV) provide further assurance against containment basemat attack. Manual connections make it possible to use onsite or offsite water systems to maintain core cooling. The result of this design effort is that in the event of a severe accident, the whole body dose consequence at the calculated site boundary is very low. More information on this subject can be found in Chapter 11.

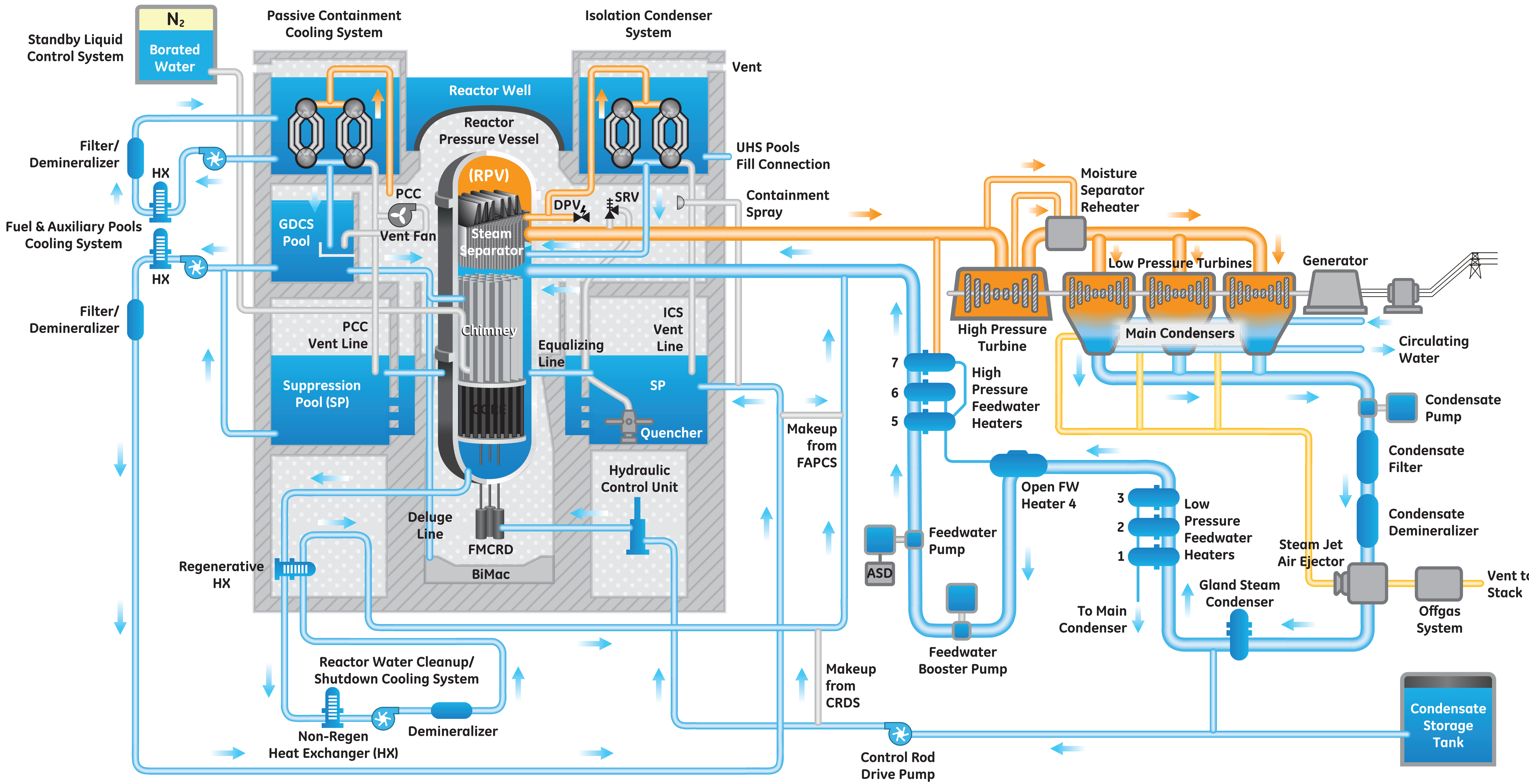


Figure 2-2. ESBWR Major Systems

Feature	BWR/6	ABWR	ESBWR
Recirculation System inside RPV	Two external loop Recirc system with jet pumps	Vessel-mounted reactor internal pumps	Natural circulation
Control Rod Drives	Locking piston CRDs	Fine-motion CRDs	Fine-motion CRDs
ECCS	2-division ECCS plus HPCS	3-division ECCS	4-division, passive, gravity-driven
Reactor Vessel	Welded plate	Extensive use of forged rings	Extensive use of forged rings
Primary Containment	Mark III - large, low pressure, not inerted	Compact, inerted	Compact, inerted
Isolation Makeup Water	RCIC	RCIC	Isolation condensers, passive
Shutdown Heat Removal	2-division RHR	3-division RHR	Non-safety system combined with RWCU
Containment Heat Removal	2-division RHR	3-division RHR	Passive
Emergency AC	3 safety-grade D/G	3 safety-grade D/G	Non-safety D/G
Alternate shutdown	2 SLC pumps	2 SLC pumps	2 SLC accumulators
Control & Instrumentation	Analog, hardwired, single channel	Digital, multiplexed, fiber optics, multiple channel	Digital, multiplexed, fiber optics, multiple channel
In-core Monitor Calibration	TIP system	A-TIP system	Fixed, In-core Gamma thermometers
Control Room	System-based	Operator task-based	Operator task-based
Severe Accident Mitigation	Not specifically addressed	Inerting, drywell flooding, containment venting	Inerting, drywell flooding, core catcher

Table 2-1. Comparison of Key ESBWR Features to previous BWRs

Improvements to Operation and Maintenance

With the goal of simplifying the utility's burden of operation and maintenance (O&M) tasks, the design of every ESBWR electrical and mechanical system, as well as the layout of equipment in the plant, is focused on improved O&M.

The reactor vessel lower sections are made of forged rings rather than welded plates. This eliminates 30% of the welds from the core beltline region, for which periodic in-service inspection is required.

The FMCRDs permit a number of simplifications. First, scram discharge piping and Scram Discharge Volumes (SDVs) were eliminated, since the hydraulic scram water is discharged into the reactor vessel. By supporting the drives directly from the core plate, shootout steel located below the reactor vessel to mitigate the rod ejection accident was

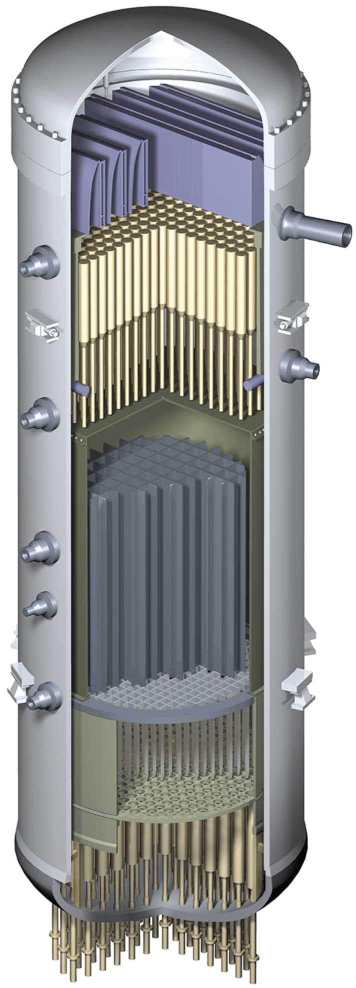


Figure 2-3. ESBWR Reactor Pressure Vessel and Internals

eliminated. The number of hydraulic control units (HCUs) was reduced by connecting two drives to each HCU, as was done on the ABWR. The number of rods per gang was increased up to 26 rods, greatly improving reactor startup times. Finally, since there are no organic seals, only two or three drives will be inspected per outage, rather than the 30 specified in most current plants.

Responses to transients and accidents are first attempted by non-safety makeup systems, together with the isolation condensers. At high pressure, the CRD pumps of the Control Rod Drive system can add water directly to the RPV via a feedwater line. Postulated Loss-of-Coolant Accidents (LOCAs) are mitigated by automated reactor pressure blowdown followed by passive gravity-driven ECCS (GDCS), which has sufficient water stored in the containment to completely flood the lower drywell and the reactor to 1 meter above the top of fuel. Residual decay heat is removed from the containment passively via heat exchangers located directly above and outside the containment boundary.

By combining the reactor water cleanup function with shutdown heat removal, simplification was achieved in the reduction of equipment. A side benefit is that decay heat removal after shutdown can be accomplished at high pressure.

Lessons learned from operating experience were applied to the selection of ESBWR materials. Stainless steel materials that are qualified as resistant to Intergranular Stress Corrosion Cracking (IGSCC) were used. In areas of high neutron flux, materials were also specially selected for resistance to Irradiation-Assisted Stress Corrosion Cracking (IASCC). Hydrogen Water Chemistry (HWC) and on-line noble chemistry is recommended for normal operation to further mitigate any potential for stress corrosion cracking.

The use of material producing radioactive cobalt was minimized. The main condenser uses titanium or stainless steel The use of low alloy steel or stainless steel in applications that currently use carbon steel was expanded. Depleted Zinc Oxide (DZO) injection to the feedwater system is recommended to further control radiation buildup. These materials choices reduce plant-wide radiation levels and radwaste and will accommodate more stringent water chemistry requirements.

Also contributing to good reactor water chemistry is the increase of the Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDC) capacity to approximately two percent of feedwater flow.

The ESBWR Reactor Building (including containment) was configured to simplify and reduce the O&M burden. Figure 2-4 illustrates some of the key design features of the ESBWR containment. In-containment elevated water tanks (GDCS) plus a raised suppression pool provide the means to passively provide ECCS, if necessary, and assure core coverage for all design basis events. Natural convection heat exchangers located outside and just above the containment provide passive heat removal. The

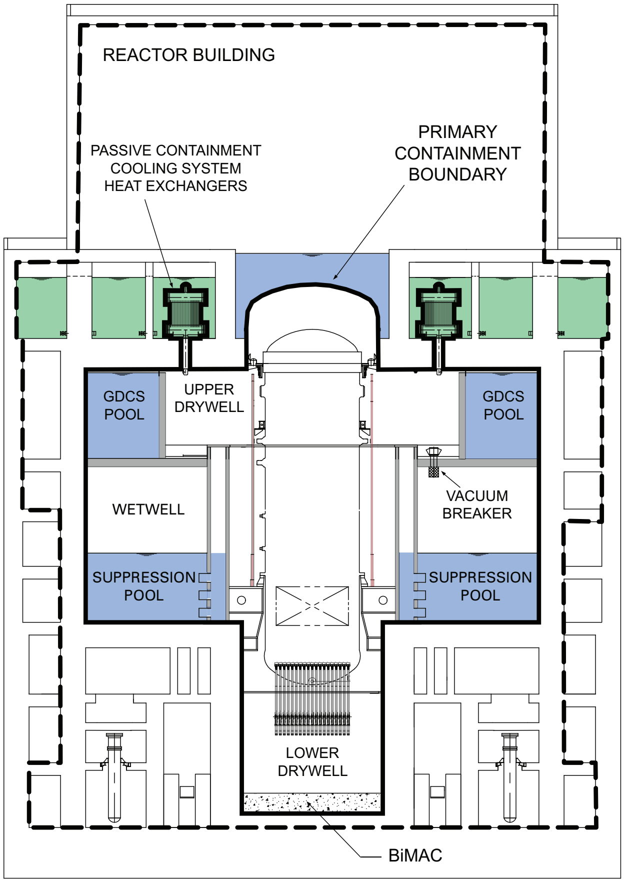


Figure 2-4. ESBWR Reactor Building and Containment

containment itself is a reinforced concrete containment vessel (RCCV).

Within the containment itself, no equipment requires servicing during plant operation and the amount of equipment that requires maintenance during outages is significantly reduced. The containment is significantly smaller than that of the preceding BWR/6, but about the same size as ABWR. However, primarily due to the elimination of the recirculation system, there is actually more room to conduct maintenance operations. To simplify maintenance and surveillance during scheduled outages, permanently installed monorails and platforms permit 360° access, and both the upper and lower drywells have separate personnel and equipment hatches. To simplify FMCRD maintenance, a rotating platform is permanently installed in the lower drywell, and semi-automated equipment was specially designed to remove and install that equipment. The wetwell area is compact and isolated from the rest of the containment, thus minimizing the chance for suppression pool contamination with foreign material.

A new Reactor Building design surrounds the containment. Its volume (including containment) is about 30% less than that of the BWR/6 and requires substantially lower construction quantities. Its layout is integrated with the containment, providing 360° access with servicing areas located as close as practical to the equipment requiring regular service. Clean and contaminated zones are well defined and kept separate by limited controlled access. The fuel pool is sized to store at least ten years of spent fuel plus a full core.

Controls and instrumentation were enhanced through incorporation of digital technologies with automated, self-diagnostic features. The use of multiplexing and fiber optic cable has eliminated 1.3 million feet of cabling. Within the safety systems, the adoption of a two-out-of-four trip logic and the fiber optic data links have significantly reduced the number of required nuclear boiler safety system related transmitters. In addition, a three-channel controller architecture was adopted for the primary process control systems to provide system failure tolerance and on-line repair capability. These new I&C features were first added in ABWR.

A number of improvements were made to the Neutron Monitoring System (NMS). Fixed wide-range neutron detectors have replaced retractable source and intermediate range monitors. In addition, an automatic, period-based protection system replaced the manual range switches used during startup. The Traversing Incore Probe (TIP) calibration system has been replaced by fixed Gamma Thermometers (GT).

The man-machine interface was significantly improved and simplified for the ESBWR using advanced technologies such as large, flat-panel displays, touch-screen CRTs and function-oriented keyboards. The number of alarm tiles was reduced by almost a factor of ten. Many operating processes and procedures are automated, with the control room operator performing a confirmatory function. Figure 2-5 illustrates a main control room for the ABWR, which uses similar technology and serves as a base-

Figure 2-5. ABWR (Lungmen) Main Control Room Panels

line for further optimization and streamlining of the ESBWR control room.

The plant features discussed above, while simplifying the operator's burden, have an ancillary benefit of increased failure tolerance and/or reduced error rates. ABWR experience since 1996 and studies for ESBWR show that less than one unplanned scram per year will be experienced. Increased system redundancies will also permit on-line maintenance. Thus, both forced outages and planned maintenance outages will be significantly reduced.

The ESBWR combines advanced facility design features and administrative procedures designed to keep the occupational radiation exposure to personnel as low as reasonably achievable (ALARA). During the design phase, layout, shielding, ventilation and monitoring instrument designs were integrated with traffic, security, and access control. Operating plant results were continuously integrated during the design phase. Clean and controlled access areas are separated.

Reduction in the plant personnel radiation exposure was achieved by: (1) minimizing the necessity for and amount of personnel time spent in radiation areas and (2) minimizing radiation levels in routinely occupied plant areas in the vicinity of plant equipment expected to require personnel attention.

Changes in the materials will lead to a significant reduction in the quantity of radwaste generated through radioactive corrosion products. In addition, the condensate treatment system was improved to include both pre-filtration and deep bed demineralizers without regeneration, which reduces liquid and solid radwaste input. Extensive use of mobile radwaste technology is used in the ESBWR radwaste system design. This also contributes to minimizing radiation exposure to operating personnel.

Chapter 3 Nuclear Steam Supply Systems

Overview

The Nuclear Steam Supply System (NSSS) produces steam from the nuclear fission process, and direct this steam to the main turbine. The NSSS is comprised of: (1) the reactor vessel, which serves as a housing for the nuclear fuel and associated components, (2) the control rod drive system, (3) the nuclear boiler system and (4) the isolation condenser system. Other supporting systems are described in Chapter 5, Auxiliary Systems.

Reactor Vessel and Internals

The reactor vessel houses the reactor core, which is the heat source for steam generation. The vessel contains this heat, produces the steam within its boundaries, and serves as one of the fission product barriers during normal operation. The ESBWR reactor assembly is shown in Figure 3-1. The diameter of the ESBWR reactor pressure vessel (RPV) is the same size as for the ABWR. The RPV is approximately 27.6 m in height and 7.1 m in diameter.

The most important new features of the ESBWR RPV and internals are as follows:

- Steam nozzles with flow restrictors
- Double feedwater nozzle thermal sleeve
- Sliding block vessel support
- Relatively flat bottom head
- Elimination of large nozzles below the core
- Use of forged shell rings at and below core elevation
- A tall partitioned chimney to promote natural circulation core flow

The RPV design is based on proven BWR technology. A noteworthy feature is the lack of any large nozzles below the elevation of the top of the core. This RPV nozzle configuration precludes any large pipe ruptures at or below the elevation of the core. It is a key factor in the ability of ESBWR safety systems to keep the core completely and continuously flooded for the entire spectrum of design basis Loss-of-Coolant Accidents (LOCAs). Many of the features listed above were introduced in the ABWR.

The vessel contains the core support structure that extends to the top of the core. The presence of a large volume of steam and water results in two very important and beneficial characteristics. First, it provides a large reserve of water above the core, which translates directly into a much longer period of time being available before core uncovery can occur as a result of feed flow interruption or a LOCA. Consequently, this gives an extended period of time during which automatic systems or plant operators can reestablish reactor inventory control using any normal, non-safety-related system capable of injecting water into the reactor. Timely initiation of these systems is designed to preclude initiation of the emergency safety equipment. This easily controlled response to loss of normal feedwater is a significant operational benefit. Second, the larger RPV volume leads to a reduction in the ESBWR pressure rise that would occur after a rapid isolation of the reactor from the normal heat sink.

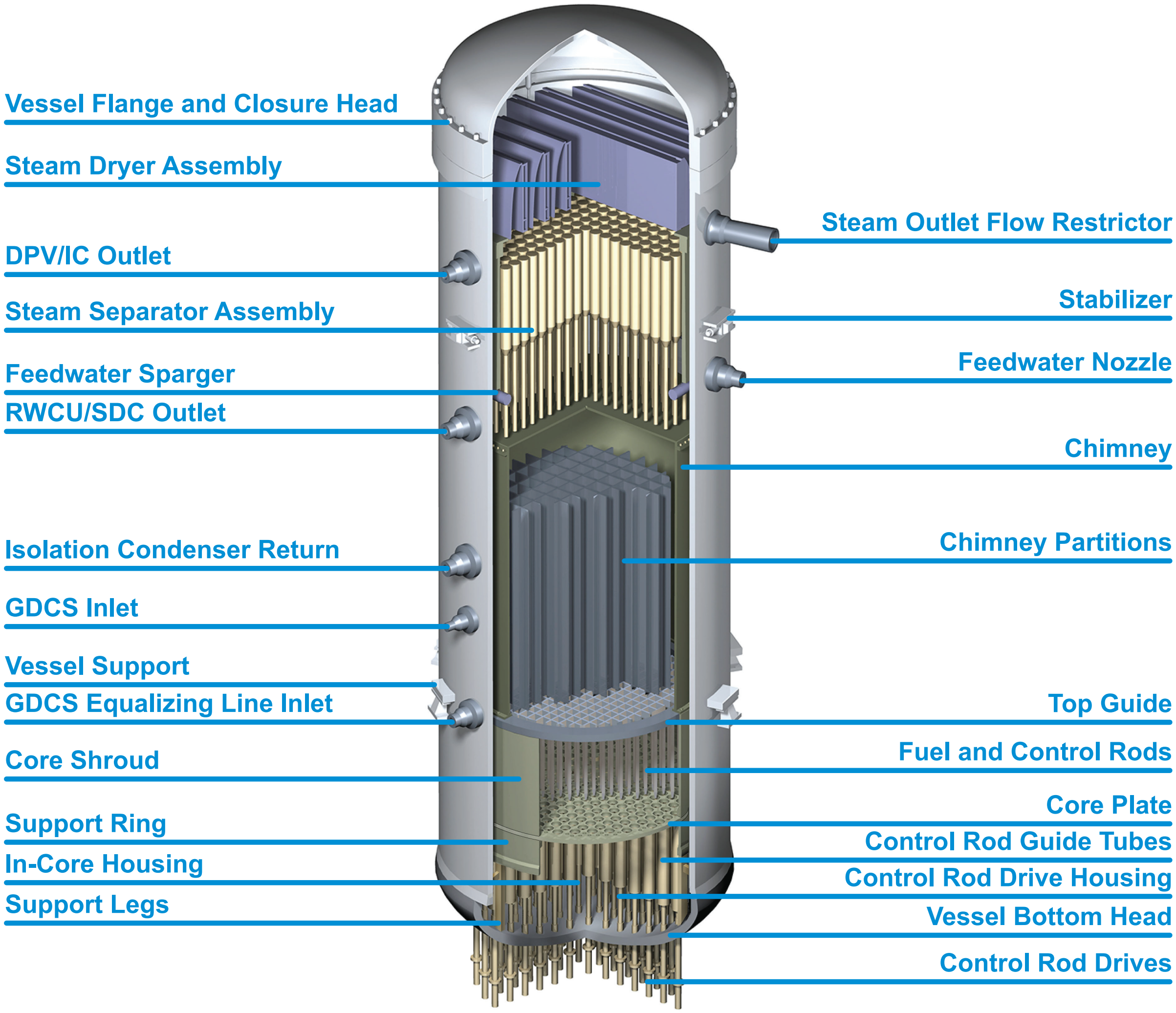


Figure 3-1. ESBWR Reactor Assembly

The following sections provide further descriptions of the unique features of the ESBWR RPV and internals.

RPV Closure Head

The RPV closure head is elliptical in shape and is fabricated of low alloy steel, per ASME SA-508, Grade 3, Class 1. It is secured to the RPV by 84 sets of fasteners (studs and nuts). These nuts are tightened in groups of (typically) four at a time, using an automatic or semiautomatic four-stud tensioner device. The vessel closure seal consists of two concentric O-rings which perform without detectable leakage at all operating conditions, including hydrostatic testing.

Steam Nozzle with Flow Restrictor

The ESBWR RPV has flow restricting venturi located in the steam outlet nozzles. Besides providing an outlet for steam from the RPV, the steam outlet nozzles will provide for: (1) steam line break detection by measuring steam flow to signal a trip for the main steam isolation valves; (2) steam flow measurement for input to the feedwater control system; and (3) a flow-choking device to limit blowdown and associated loads on the RPV and internals in the event of a postulated main steam line break. Calculations show that the pressure drop in the nozzle is within the requirements of the steady-state performance specification.

Feedwater Nozzle Thermal Sleeve

There are three feedwater nozzles for each of the two feedwater lines that utilize double thermal

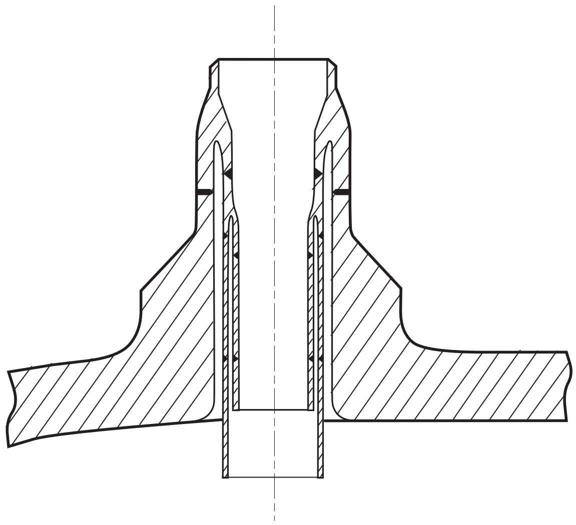


Figure 3-2. ESBWR Reactor Pressure Vessel Feedwater Nozzle

sleeves welded to the nozzles. The double thermal sleeve protects the vessel nozzle inner blend radius from the effects of high frequency thermal cycling. A schematic of the feedwater nozzle is shown in Figure 3-2.

Feedwater Spargers

The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger in two halves is fitted to each feedwater nozzle by a tee and is shaped to conform to the curve of the vessel wall. The sparger tee inlet is connected to the thermal sleeve arrangement. Feedwater flow enters the center of the spargers and is discharged radially inward to mix the cooler feedwater with the downcomer flow from the steam separators and steam dryer.

Vessel Support

The vessel supports are of the sliding block type geometry and are provided at a number of positions around the periphery of the vessel. Multiple vessel supports along with the corresponding pedestal RPV support brackets provide:

- Openings to permit water to pass from the upper to lower drywell
- Access for in-service inspection (ISI) of the bottom head weld

More information on the vessel supports can be found in Chapter 8.

Reactor Vessel Bottom Head

The bottom head consists of a spherical bottom cap, made from a single forging, extending to the toroidal knuckle between the head and vessel cylinder and encompassing the control rod drive (CRD) penetrations. With a bottom head thickness of approximately 260-mm, the bottom head meets the ASME allowables for the specified design loads. The main advantage of using a single forging for the bottom head is that it eliminates all RPV welds within the CRD pattern, thus reducing future (ISI) requirements.

Stabilizers

Stabilizers are located around the periphery of the RPV toward its upper end. These provide reaction points to resist horizontal loads and suppress RPV motion due to earthquakes and postulated pipe rupture events.

Forged Shell Rings

The ESBWR RPV utilizes low alloy forged shell rings adjacent to and below the core belt line region. The flanges and large nozzles are also low alloy steel. The shell rings above the core beltline region and the RPV closure head are made from low alloy steel forgings or plate per ASME SA-533, Type B, Class 1. The required Reference Temperature Nil Ductility Transition (RTNDT) of

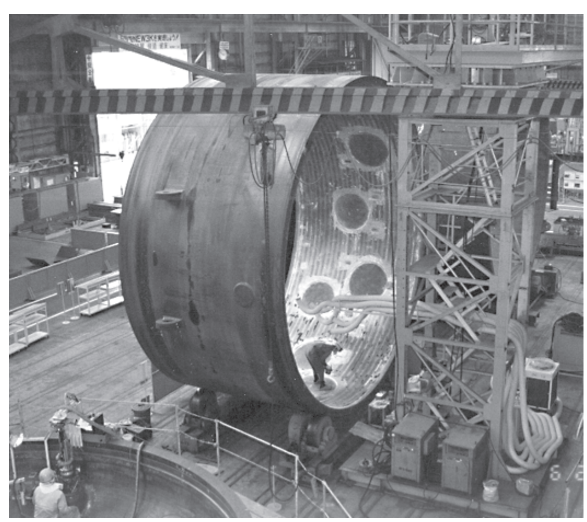


Figure 3-3. ABWR RPV Forged Steel Ring

the vessel shell forgings is -20°C. Figure 3-3 shows one of the RPV forged shell rings during fabrication of an ABWR vessel. A similar process will be used on ESBWR.

Core Shroud

The shroud is a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downcomer annulus flow. The upper shroud is bounded at the bottom by the core plate. The lower shroud, surrounding part of the lower plenum, is bolted to a support ring and support legs. The shroud provides lateral support for the core by supporting the core plate and top guide.

Support Legs

Support legs are welded to the inside of the vessel and are made of Ni-Cr-Fe conforming to ASME B&PV Code Case N-580-1. The support legs support the weight of the steam separators, chimney, top guide, shroud, core plate, support ring, and the peripheral fuel bundles.

Core Plate

The core plate consists of a circular plate with round openings. The core plate provides lateral support and guidance for the control rod guide tubes, in-core flux monitor guide tubes, peripheral fuel supports, and startup neutron sources. The last two items are also supported vertically by the core plate. The entire assembly is bolted to a support ledge in the shroud. The core plate also forms a partition within the shroud, which causes the recirculation flow to pass into the orificed fuel supports and through the fuel assemblies.

Top Guide

The top guide consists of a grid that gives lateral support of the top of the fuel assemblies. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, two or three fuel assemblies. Holes are provided in the bottom of the support intersections to anchor the in-core flux monitors and startup neutron sources. The top guide is bolted to the top of the core shroud.

Fuel Supports

The fuel supports are of two basic types; namely, peripheral fuel supports and main fuel supports. The peripheral fuel supports are located at the outer edge of the active core and are not adjacent to control rods. Each peripheral fuel support sustains one fuel assembly and contains an orifice designed to assure proper coolant flow to the peripheral fuel assembly. Each main fuel support sustains four fuel assemblies vertically upward and horizontally and is provided with orifices to assure proper coolant flow distribution to each fuel bundle. The main fuel support sits on the top of the control rod guide tube, which carries the weight of the fuel rods down to the bottom of the RPV. The control rods pass through cruciform openings in the center of the main fuel support.

Control Rod Drive Housing

The control rod drive housing provides extension of the RPV for installation of the control rod drive, and the attachment of the CRD line. It also supports the weight of a control rod, control rod drive, control rod guide tube, main fuel support and four fuel assemblies.

Control Rod Guide Tubes

The control rod guide tubes extend from the top of the control rod drive housings up through holes in the core plate. Each guide tube is designed as the guide for the lower end of a control rod and as the support for a main fuel support. This locates the four fuel assemblies surrounding the control rod, which, in turn, transmits the weight of the guide tube, fuel support, and fuel assemblies to the reactor vessel bottom head. The control rod guide tube also contains holes, near the top of the control rod guide tube and below the core plate, for coolant flow to the orificed fuel supports. In addition, the guide tube provides a connection to the Fine Motion Control Rod Drive (FMCRD) to restrain a hypothetical downward ejection of the FMCRD in case of a postulated RPV weld failure.

In-Core Housing

The in-core housings provide extensions of the RPV at the bottom head for the connection of various in-core flux monitoring sensor assemblies, which are components of the Neutron Monitoring System. It also supports the weight of an in-core flux monitoring sensor assembly, in-core guide tube, and part of the in-core guide tube stabilizer assembly.

Chimney

The chimney is a long stainless steel cylinder that supports the steam separators and is bolted to the top guide. It provides the driving head necessary to create and sustain the natural circulation flow.

Chimney Partitions

Partitions are located inside the chimney that separate groups of up to 16 fuel bundles. These partitions act to channel the mixed steam and water flow exiting the core into smaller chimney sections to limit the cross flow and minimize the potential for recirculating eddies that could result from a much larger open chimney.

Steam Separator Assembly

The steam separator assembly consists of a flat base on top of which is welded an array of standpipes with a three-stage steam separator located at the top of each standpipe. The steam separator assembly rests on the top flange of the chimney and forms the cover of the core discharge plenum region. The separator assembly is bolted to the chimney flange by long hold down bolts which, for ease of removal, extend above the separators. During installation, the separator base is aligned on the chimney flange with guide rods and finally positioned with locating pins. The objective of the long-bolt design is to provide direct access to the bolts during reactor refueling operations with minimum-depth underwater tool manipulation during the removal and installation of the assemblies. It is not necessary to engage threads in mating up the shroud head. A tee-bolt engages in the chimney flange and its nut is tightened to only nominal torque. Final loading is established through differential expansion of the bolt and compression sleeve.

The fixed axial flow type steam separators have no moving parts and are made of stainless steel. In each separator, the steam-water mixture rising through the standpipe impinges on vanes which give the mixture a spin to establish a vortex wherein the centrifugal forces separate the water from the steam in each of three stages. Steam leaves the separator at the top and passes into the wet steam plenum below the dryer (Figure 3-4). The separated liquid exits from the lower end of each stage of the separator and enters the pool that surrounds the standpipes to join the downcomer annulus flow.

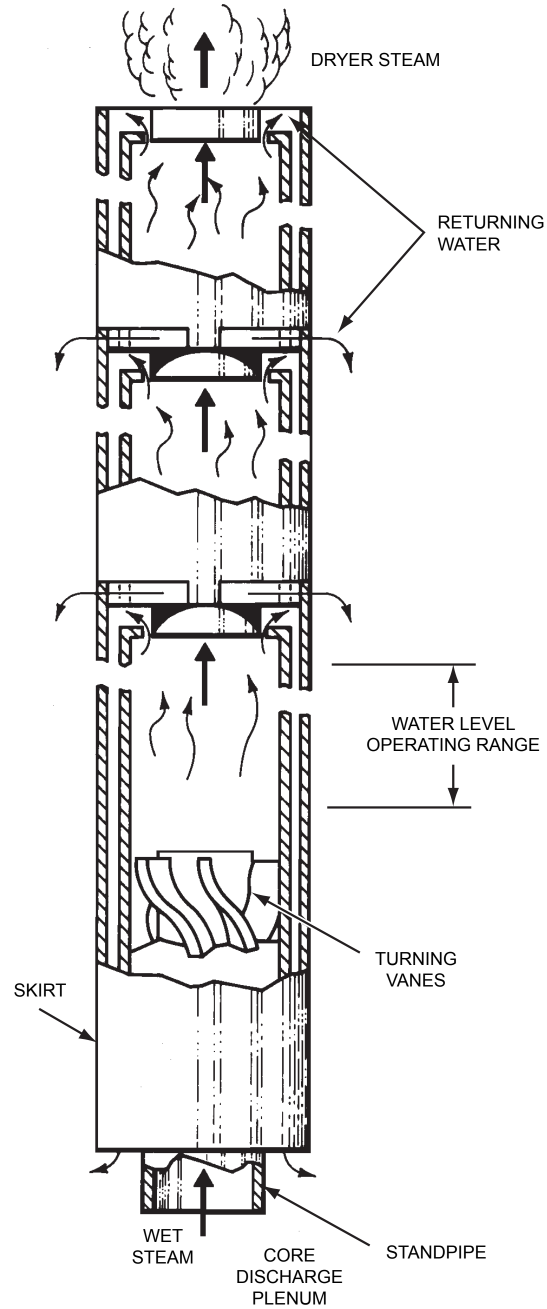


Figure 3-4. Schematic of Steam Flow Through Separator

Steam Dryer Assembly

The steam dryer assembly consists of multiple banks of dryer units mounted on a common structure which is removable from the RPV as an integral unit. The assembly includes the dryer banks, dryer supply, and discharge ducting, drain collecting troughs, drain ducts and a skirt which forms a water seal extending below the separator reference zero elevation. Steam from the steam separators flows upward to the steam

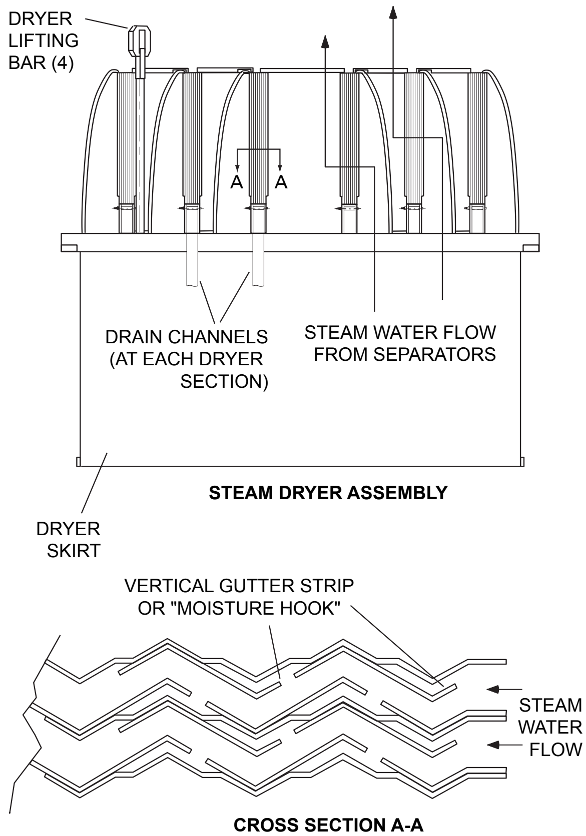


Figure 3-5. Schematic of Steam Flow Through Dryer

dryer and outward through the drying vanes (Figure 3-5). These vanes are attached to a top and bottom supporting member forming a rigid, integral unit.

Moisture is removed and carried by a system of troughs and drains to the pool surrounding the separators and then into the downcomer annulus between the chimney and reactor vessel wall. Upward and radial movement of the dryer assembly under the action of blowdown and seismic loads is limited by support brackets on the vessel shell and hold down brackets inside the RPV closure head. The assembly is arranged for removal from the vessel as an integral unit on a routine basis.

DPV/IC Outlet and IC Return

There are four 450-mm nozzles spaced around the RPV for connection to each of the four Isolation Condenser (IC) subsystems and the eight Depressurization Valves (DPVs). The IC return line nozzles are 200 mm diameter.

GDCS Inlet

There are eight 150-mm nozzles spaced around the RPV for connection to each of the four divisions of the Gravity Driven Cooling System (GDCS) injection lines. In addition, flow limiters with a venturi shape are designed into each nozzle to limit the flow in the event of a postulated GDCS line break.

GDCS Equalizing Line Inlet

There are four 150-mm nozzles spaced around the RPV for connection to each of the four divisions of the Gravity Driven Cooling System (GDCS) injection lines. In addition, flow limiters with a venturi shape are designed into each nozzle to limit the flow in the event of a postulated GDCS line break.

RWCU/SDC Outlet

There are two 300-mm nozzles provided for connection to each of the trains of the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system.

Control Rod Drive System

The Control Rod Drive (CRD) System controls changes in core reactivity during power operation by movement and positioning of the neutron absorbing control rods within the core in fine increments in response to control signals from the Rod Control and Information System (RC&IS). The CRD System provides rapid control rod insertion in response to manual or automatic signals from the Reactor Protection System (RPS). Figure 3-6 shows the basic system configuration and scope.

When scram is initiated by the RPS, the CRD System inserts the negative reactivity necessary to shut down the reactor. Each control rod is normally controlled by an electric motor unit. When a scram signal is received, high-pressure water stored in nitrogen charged accumulators forces the control rods into the core. Simultaneously, the control rod drives are inserted via the electric motor units. Thus, the hydraulic scram action is backed up by an electrically energized insertion of the control rods.

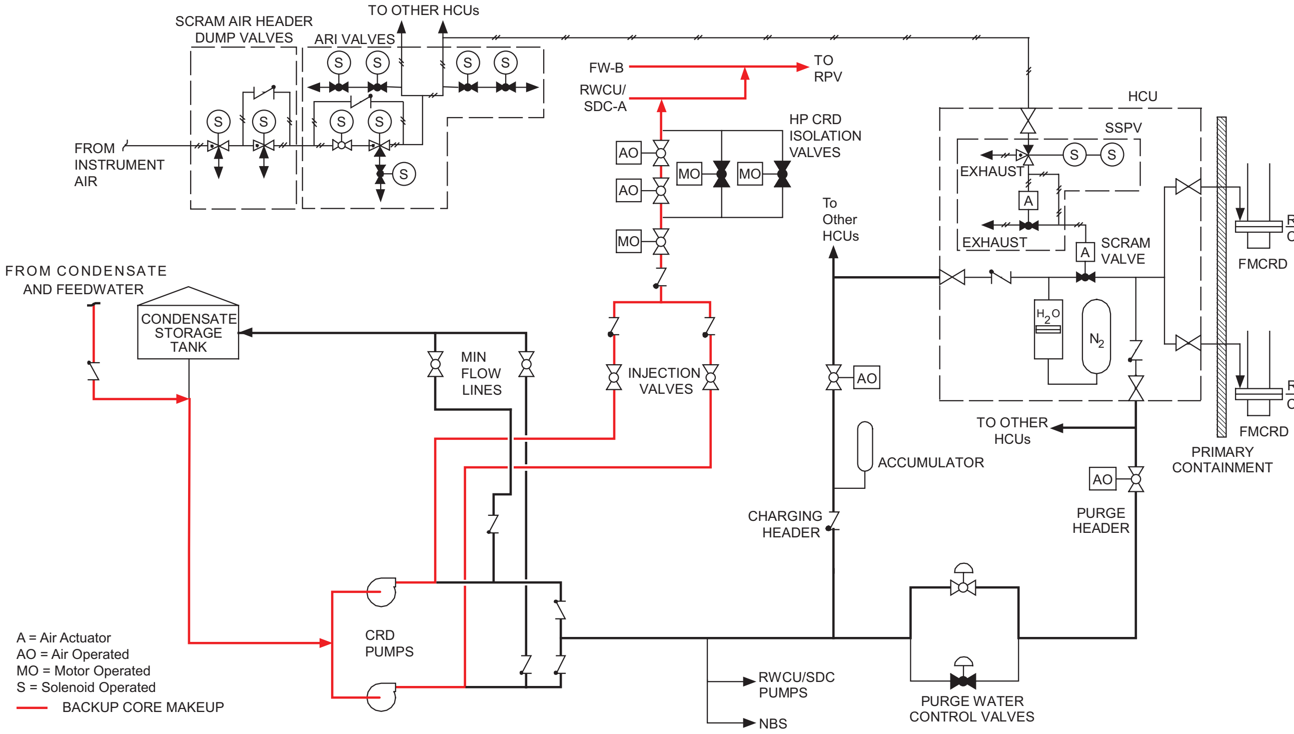


Figure 3-6. Control Rod Drive System Schematic

The CRD System consists of three major elements:

- Electrohydraulic Fine Motion Control Rod Drive (FMCRD) mechanisms.
- Hydraulic Control Unit (HCU) assemblies
- Control Rod Drive Hydraulic System (CRDHS)

The FMCRDs provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion for abnormal operating conditions. Simultaneous with scram, the FMCRDs also provide electric-motor-driven run-in of control rods as a path to rod insertion that is diverse from the hydraulic-powered scram. The hydraulic power required for scram is provided by high pressure water stored in the individual HCUs. An HCU can scram two FMCRDs. It also provides the flow path for purge water to the associated drives during normal operation. The CRDHS supplies pressurized water for charging the HCU scram accumulators and purging to the FMCRDs. In addition, the CRDHS supplies high pressure makeup water to the RPV during certain transients.

Fine Motion Control Rod Drives (FMCRD)

The ESBWR FMCRDs are distinguished from the locking piston CRDs (which are in operation in almost all current GE/GEH-designed plants) in that the control blades are moved electrically during normal operation. This feature permits small power changes, improved startup time and improved power maneuvering. The FMCRD, as with current drives, is inserted into the core hydraulically during emergency shutdown. Because the FMCRD has the additional electrical motor, it drives the control blade into the core even if the primary hydraulic system fails to do so, thus providing an additional level of protection against Anticipated Transient Without Scram (ATWS) events. The FMCRD design is an improved version of similar drives that have been in operation in European BWRs since 1972, and is basically the same drive design that is in use in ABWR.

Figure 3-7 shows a cross-section of the FMCRD as used in the ESBWR. The FMCRD consists of four major subassemblies: the drive, spool piece, brake and motor/synchros. The spool piece and motor may be removed without disturbing the drive, allowing maintenance with low personnel exposure.

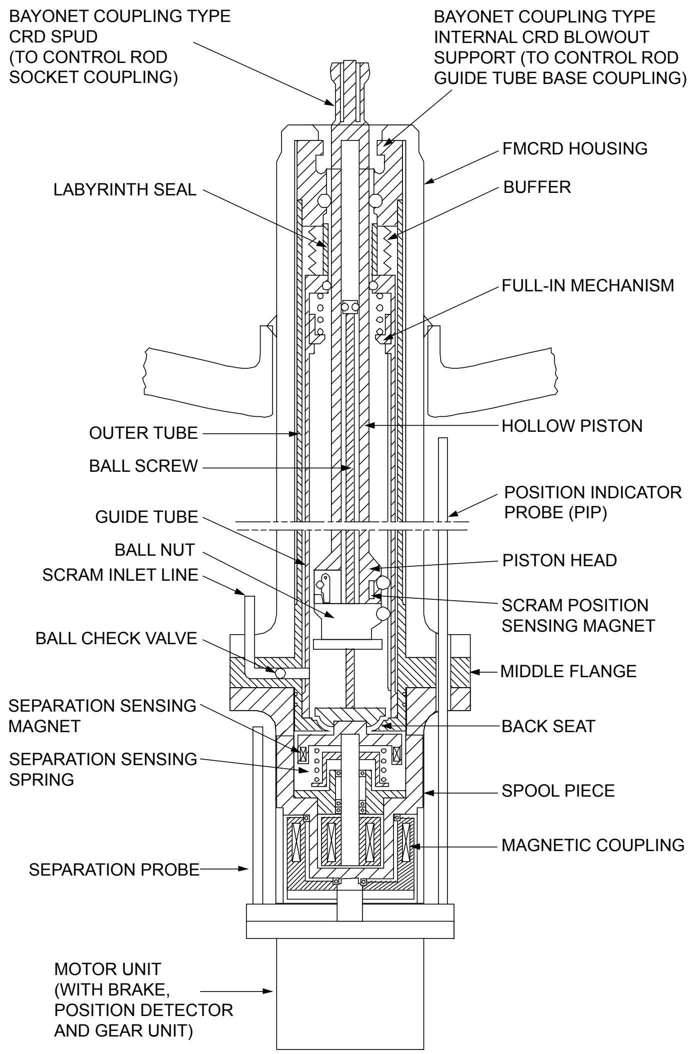


Figure 3-7. Fine Motion Control Rod Drive Cross-Section

The drive consists of the outer tube, hollow piston, guide tube, buffer, labyrinth seal, ball check valve, ball nut and ball screw shaft.

The coupling is a bayonet-type configuration which, when coupled with the mating coupling on the control rod blade, precludes separation of the blade and the hollow piston.

The hollow piston is a long hollow tube with a piston head at the lower end. The hollow piston is driven into the reactor during scram by the pressure differential that is produced by the high scram flow from the HCU accumulator. The labyrinth seal, which is contained inside the buffer, at the top end of the outer tube restricts the flow from the drive to the reactor, thereby maximizing the pressure drop which enhances scram performance. Additionally, it allows the purge flow during normal operation to preclude entrance of reactor water and associated crud into the drive. The piston head contains latches that latch into notches in the drive guide tube after scram. The scram buffering action is provided by an assembly of Belleville washers in the buffer and is supplemented by hydraulic damping as the buffer assembly parts come together.

The outer tube performs several functions, one of which is to absorb the scram pressure, preventing its application to the CRD housing, which is part of the reactor coolant pressure boundary (RCPB). The outer tube top end is a bayonet connection similar to that employed on the hollow piston which couples with a similar bayonet connection on the control rod guide tube, sandwiching the CRD housing end cap between the two. The outer tube lower end is a middle flange which bolts to the CRD housing flange. The bolts allow the drive to remain in place when the motor and spool piece are removed. The combination of the positive coupling of the control rod guide tube and the drive and the flange on the lower end of the outer tube form a positive means of preventing ejection of the FMCRD/control rod for any postulated housing break. Protection against the postulated failure of the housing to stub tube weld is provided by the same features, with the shootout load being transferred to the core plate by the flange at the top end of the control rod guide tube. These internal CRD blowout support features allow the elimination of the external support structure of beams, hanger rods, grids, and support bars used to prevent rod ejection as in previous GE BWR product lines.

The latches on the hollow piston are designed so that with only one being engaged, it is sufficient to hold the control rod in place under all loading, including the ejection load caused by a scram line break.

In normal operation, the hollow piston rests on the ball nut and is raised and lowered by translation of the ball nut resulting from rotation of the ball screw. The latches are held in a retracted position by the ball nut. During scram, the hollow piston is lifted off the ball nut by the hydraulic pressure.

The spline arrangement between the ball screw lever coupling and the middle flange back seat provides a back seat-type anti-withdrawal gear that automatically engages whenever the spool piece is lowered. This prevents the ball screw from rotating and withdrawing the rod.

The spool piece contains the magnetic coupling between the motor and the ball screw drive shaft. The magnetic coupling is employed to achieve seal-less leak-free operation of the control rod drive mechanism. The magnetic coupling consists of an inner and an outer rotor. The inner rotor is located inside the spool piece pressure boundary and the outer rotor is located on the outside. Each rotor has permanent magnets mounted on it. As a result, the inner and outer rotors are locked together by the magnetic forces acting through the pressure boundary and work as a synchronous coupling. The outer rotor is coupled with the motor unit and driven by the motor. The inner rotor is keyed to the drive shaft and follows the rotation of the outer rotor.

The spool piece also contains a weighing device, which is a spring-loaded platform with two magnets located on it. In normal service, the weight of the hollow piston and control rod is transferred to the weighing device. If, during withdrawal, the weight of the rod or hollow piston is removed from the device, then the device will move upwards and trigger two external reed switches, called separation switches. If either is opened, withdrawal motion is inhibited. There are two separation switch probes which are directly opposite each other. Each probe contains one switch. Thus, the reactivity addition from a postulated Rod Drop Accident is limited to a few cents, and this event is no longer analyzed in Safety Analysis Reports.

The spool piece is bolted to the CRD housing by bolts which pass through the middle flange. As mentioned above, the middle flange is also bolted to the CRD housing. The double bolting arrangement, combined with the back seat-type lock feature discussed above, allows spool piece servicing without disturbing the drive.

The motor unit bolts to the bottom of the spool piece. The motor unit consists of the induction motor, position signal generators and holding brake.

There are two resolver-type position signal generators located within the motor unit. The resolvers provide a continuous analog readout signal of control rod position during normal operation and are driven by gears from the motor shaft.

The holding brake located in the motor unit serves to restrain the rod against withdrawal in the unlikely event that the scram line breaks. The brake is redundant with the ball check valve in mitigating the scram line break. It should be noted at this point that the check valve on the FMCRD has no function other than to mitigate the scram line break and to limit leakage during drive replacement.

The balance of the FMCRD System includes the scram position probes which are mounted on the outside of the CRD housing. The scram probe provides a position signal at 10%, 40%, and 60% insertion, as well as continuous full-in. The continuous full-in signal prevents the loss of position indication that would otherwise occur while the hollow piston is held by the scram latches at the top latched position.

The probes use reed switches similar to the Locking Piston Control Rod Drive (LPCRD), as do the separation switch probes that are mounted on the side of the spool piece. The separation probes and associated circuits and equipment are considered important to safety and are therefore categorized as safety-related.

In addition to the FMCRD and probes, other items in the system include the power supply to motor, the Hydraulic Control Unit (HCU), scram piping, wiring and the CRD pump and its associated equipment.

Induction Motor Controller (IMC) equipment in the Rod Control & Information System (RC&IS) provides the control power to the FMCRDs for performing normal control rod movements. The IMC equipment provides for AC phase direction change of the three phase AC power provided to each AC induction motor so that both insertion and withdrawal movements can be accomplished.

The Rod Brake Controllers (RBCs) within the RC&IS provide the control power for operation of the holding brakes. The holding brake is normally de-energized and engaged by spring force when the FMCRD is stationary. The RBC provides the power to energize and disengage the brake when the FMCRD is commanded to move.

Hydraulic Control Units (HCU)

The HCU consists of a gas bottle and accumulator, which are mounted on a frame. The HCU also includes the scram and scram pilot valves. In an ESBWR there is one HCU for every two FMCRDs, similar to the ABWR. The use of the paired arrangement allows savings in space and maintenance without sacrificing reliability or safety. The two FMCRDs on a given HCU are widely separated in the core so that there is no additional loss of shutdown margin if an HCU fails.

Control Rod Drive Hydraulic System

The ESBWR Control Rod Drive Hydraulic System (CRDHS) supplies clean, demineralized water, which is regulated and distributed to provide charging of the HCU scram accumulators and purge water flow to the FMCRDs. The CRDHS is also the source of pressurized water for purging the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System pumps and for providing keep-fill flow to the RPV water level reference leg instrument lines. In addition, the CRDHS provides high pressure makeup water to the reactor vessel following the loss of the normal feedwater makeup supply (red pathway in Figure 3-6).

The CRD pump is basically the same as that used in ABWR (i.e., a multistage centrifugal pump). The filtration system is basically also the same as that used in ABWR.

Nuclear Boiler System

The purpose of the Nuclear Boiler System (NBS) is to transpot steam flow from the RPV steam outlet nozzles to the main turbine , transport preheated feedwater from the condensate & feedwater system back to the RPV, provide overpressure protection of the reactor coolant pressure boundary (RCPB), and provide primary nuclear boiler instrumentation (level, pressure, temperature and flow) signals to the plant process computer system. A main steam line flow restrictor is provided in each steam outlet nozzle. It is designed to limit the flow rate in the event of a postulated steam line break. The system incorporates provisions for relief of overpressure conditions in the RPV. Also included in the NBS is the Nuclear Island portion of the Feedwater System.

Main Steam Subsystem (MS)

In the ESBWR design, four 750-mm steam lines transport steam from the steam outlet nozzles on the RPV through Reinforced Concrete Containment Vessel (RCCV) penetrations and then through the steam tunnel to the turbine. Main steam isolation valves (MSIVs) are installed in each steam line inboard and outboard of the RCCV penetrations. Ten safety/relief valves (SRVs) are installed vertically on the main steam lines. These valves provide the Automatic Depressurization System (ADS) function during an accident condition, and the discharge from each SRV is routed through the associated SRV discharge line to quenchers located in the suppression pool. Eight additional safety valves (SVS) are spring-actuated only to provide overpressure protection in the case of a postulated Anticipated Transient Without Scram (ATWS) event. The discharge from each of these valves is routed into vertically oriented discharge stacks located in the drywell and equipped with rupture disks at their ends.

In addition, the MS is equipped with eight Depressurization Valves (DPVs). These valves are actuated during postulated LOCAs and discharge directly into the drywell. Two valves are located on each of four stub tubes, which also supply steam to the four Isolation Condensers (ICs). Figure 3-8 is a simplified piping diagram of the MS.

The MS is composed of several components and subsystems in addition to the above, which are necessary for proper operation of the reactor under various operating, shutdown and accident conditions. Some of these subsystems include: main steam bypass/drain subsystem, reactor head vent subsystem, and system instrumentation.

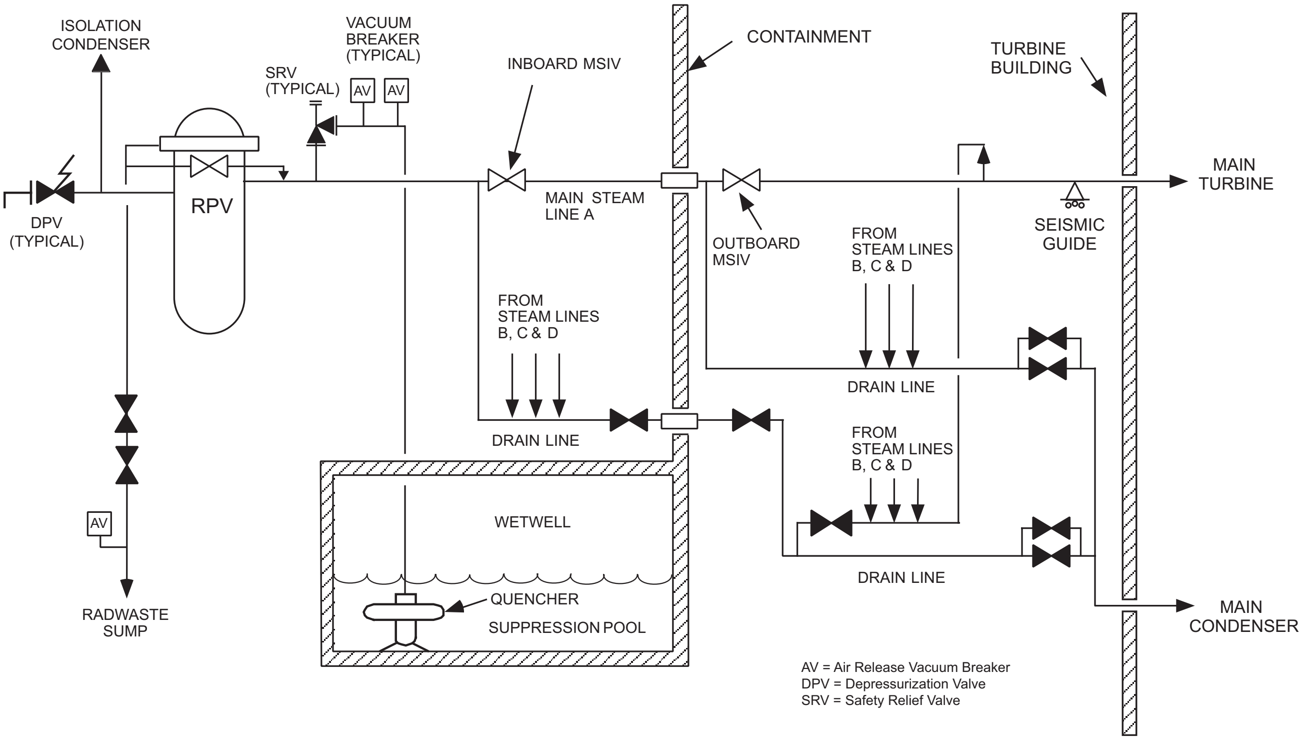


Figure 3-8. Main Steam Subsystem

Main Steam Isolation Valves (MSIV)

The main steam line isolation system is a fail-safe system that isolates the main steam lines during normal, upset, and accident conditions under the full range of reactor pressures and flow conditions. The system consists of eight MSIV assemblies mounted in four tandem pairs in the main steam lines with one valve of each pair installed inboard of the containment penetration and one valve of each pair installed outboard of the containment penetration. The MSIVs provide isolation of the main steam lines for high-energy line breaks, for containment isolation, and when required during plant shutdown condition. The MSIVs are designed to pass rated steam flow within a design pressure drop, and to limit steam line LOCA inflow to protect containment until the valves are closed. The system is shown schematically in Figure 3-8.

The MSIVs are designed to a pressure and temperature consistent with the RPV maximum design conditions. MSIVs are installed welded-in to the main steam lines to maximize the reactor coolant pressure boundary (RCPB) and containment penetration integrity. Each MSIV is designed to accommodate saturated steam at plant operating conditions. The MSIV assemblies and associated supports are designed to Seismic Category I requirements. The MSIVs form part of the RCPB and are therefore Quality Group A, and designed and fabricated to ASME B&PV Code Section III, Class 1 requirements.

The MSIVs are designed for a minimum life at the specified operating conditions. In addition to minimum wall thickness required for the design pressure, a corrosion allowance is added for the minimum design life.

MSIV-type is a gate pattern with reducing venturi inlet and outlet nozzles to fit the steam line diameter to the valve. The design uses removable internals for all wear parts and surfaces to permit ease of replacement, or allow refurbishment maintenance outside the valve body. The actuator is a high-pressure piston cylinder type. The valve actuator is capable of developing sufficient force for opening or closing the valve against a differential up to reactor design pressure, and to close against worst-case break flow. Operating power comes from process-medium integral actuation (preferred design) or high-pressure nitrogen (or nitrogen-spring) yoke-mounted actuation (alternate design).

The MSIVs are designed to close under peak accident environmental radiation, pressure and temperature conditions. In addition, they are designed to remain closed under long-term post-accident environmental conditions. Pressure drop is adjusted by the sizing of valve flow orifice diameter of the inboard and outboard MSIVs to meet design requirements. The closed MSIV leak rate is sufficiently low to provide a margin for wear and degradation during operating service so that total leakage remains within the design allowable for the cumulative leak rate through all four main steam lines.

Valve closure occurs when both of two automatic control pilot solenoid-operated valves are deenergized. Speed is controlled by cylinder inlet and exhaust path orifice sizing and factory set to provide the design stroke speed under rated operating and accident flow conditions. The MSIV actuates at two closing speeds, including a fast isolation closure by the automatic pilots, and a slow-closure speed for exercise. A separate solenoid-operated pilot valve, manually operated from the control room, is provided for a slow-closure partial- or full-stroke exercise cycle testing.

Safety/Relief Valves and Safety Valves

An SRV is in principle a dual function, direct-acting valve and is classified as safety-related . In the ESBWR, ten valves (SRVs) include the accumulators and pneumatic actuators necessary for automatic or manual actuation in addition to opening on spring pressure (dual function). Eight valves (SVs) do not include the extra actuators and open on spring pressure only (direct acting). The SRVs and SVs are considered part of the Reactor Coolant Pressure Boundary (RCPB) because the inlet side of the valves are connected to the steam line prior to the inboard MSIV (Figure 3-9). The ADS-SRV logic and three of the solenoids are also classified and qualified as safety-related per the IEEE Standards. This classification is also applied to the ADS function and other associated systems. The fourth solenoid is classified as nonsafety-related and is initiated from the Diverse Protection System.

Due to the use of the IC System in ESBWR, SRVs and SVs are not normally needed to maintain primary system pressure below the ASME Code design limits. These valves provide two main protection functions:

- **Overpressure Safety Operation:** The 18 valves function as spring-loaded safety valves and open to prevent RCPB overpressurization. The valves are self-actuated by inlet steam pressure. This will occur only during certain ATWS events.
- **Automatic Depressurization System (ADS) Operation:** The ten SRV valves are opened using a pneumatic actuator upon receipt of an automatic or manually initiated signal at the solenoid valve located on the pneumatic actuator assembly, allowing inlet steam to discharge through the SRV. The SRV pneumatic operator is so arranged that, if it malfunctions, it does not prevent the SRV from opening when steam inlet pressure reaches the spring lift setpoint. They are opened automatically or manually in the power actuated mode when required during a LOCA. The ADS designated SRVs open automatically as part of the Emergency

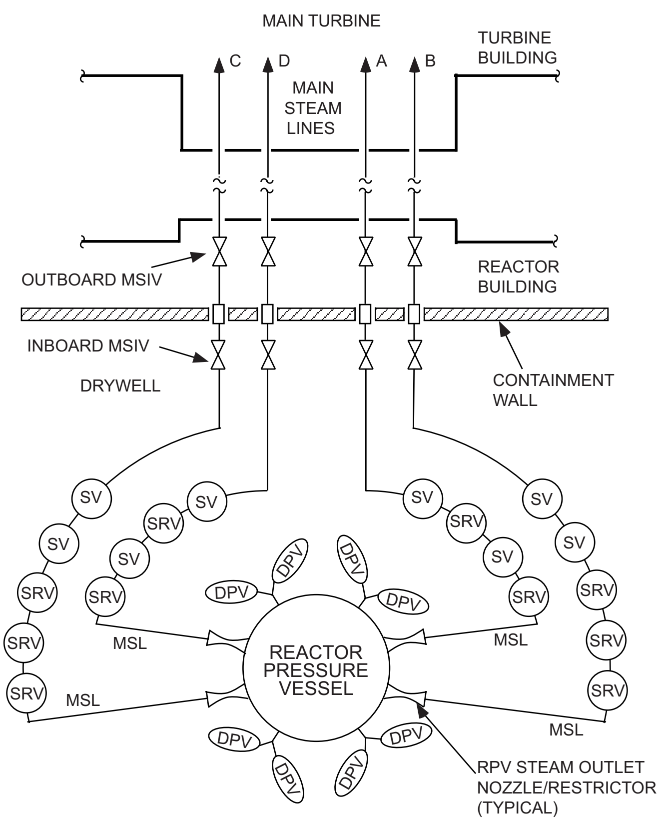


Figure 3-9. MSIV, SRV/SV and DPV Configuration

Core Cooling System (ECCS) as required to mitigate a LOCA when it becomes necessary to reduce RCPB pressure to admit low pressure ECCS coolant flow to the reactor.

The SRVs can be operated individually in the power-actuated mode by remote manual switches located in the main control room. They are provided with position sensors, which provide positive indication of SRV disk/stem position.

Each SRV has its own discharge line with two vacuum breakers. The discharge lines are sized so that the critical flow conditions occur through the valve. This prevents the conditions in the discharge lines of water hammer and pressure instability. For the ADS-SRVs, the SRV discharge lines terminate at the quenchers located below the surface of the suppression pool (SP).

The eight SVs discharge lines are routed to vertically oriented discharge stacks located in the drywell and equipped with rupture disks at their ends. The SVs discharge steam through the rupture disks into the drywell against a blast shield. Each stack also has a drain line that drains condensed steam leakage to the suppression pool

Depressurization Valves (DPV)

There are eight DPVs, located on stub lines (Figure 3-9), whose sole purpose is to aid the ADS subsystem in rapidly reducing RCPB pressure during a LOCA in order for the low pressure ECCS to add water to the RPV.

The DPVs are of a non-leak/non-simmer/non-maintenance design (Figure 3-10). They are straight-through, squib-actuated, non-reclosing valves with a metal diaphragm seal. The valve size provides about twice the depressurization capacity as an SRV. Each DPV is closed with a cap covering the inlet chamber. The cap shears off when pushed by a valve plunger that is actuated by the explosive initiator-booster. This opens the inlet hole through the plug. The sheared cap is hinged such that it drops out of the flow path and does not block the valve. The DPVs

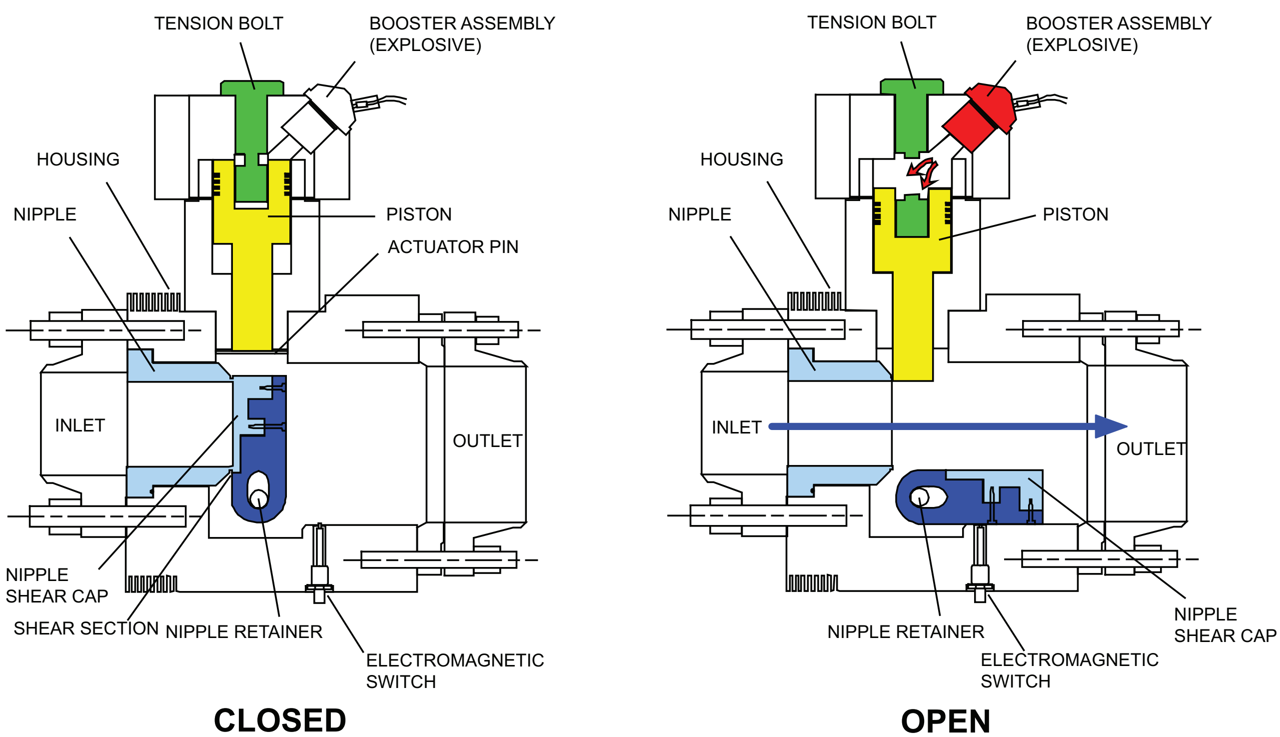


Figure 3-10. Depressurization Valve (DPV) - Typical Configuration

are designed so that there is no leakage across the cap throughout the life of the valve.

Three safety-related initiators and one non-safety-related initiator (Diverse Protection System), singly or jointly, actuate a booster, which actuates the shearing plunger. The boosters (squibs) are initiated by independent firing circuits, with each circuit incorporating triple redundancy driving three load drivers in series to provide protection against inadvertent activation. The DPV has undergone engineering development testing using a prototype to demonstrate the proper operability, reliability, and

Figure 3-11. DPV Under Test

flow capability of the design. Figure 3-11 shows the DPV test facility, located at Wyle Laboratories, Huntsville, Alabama.

Feedwater Subsystem (Nuclear Island)

Two 550-mm feedwater lines transport feedwater from the feedwater pipes in the steam tunnel through RCCV penetrations to horizontal headers in the upper drywell, which have three 300-mm riser lines that connect to nozzles on the RPV (Figure 3-12). Isolation valves are installed upstream and downstream of the RCCV penetrations. Two other valves are installed to provide rapid isolation of a postulated feedwater line break to limit the amount of additional water added to containment. The valves will be either power- and/or process-operated. Also shown in the figure are the interconnections from the Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDC), CRD and Fuel and Auxiliary Pools Cooling System (FAPCS).

Isolation Condenser System (ICS)

There are several functions that the Isolation Condenser System (ICS) provides. The ICS limits reactor pressure and suppresses pressure increase transients before they reach the Safety Relief Valve (SRV) lift setpoint, minimizing the number of SRV operating cycles following an isolation of the main steam lines. The ICS, together with the water stored in the RPV, conserves sufficient reactor coolant volumes to avoid automatic depressurization caused by low reactor water level during transients or station blackout. The ICS removes excess sensible and core decay heat from the reactor, in a passive way and with minimal loss of coolant inventory from the reactor, when the normal heat removal system is unavailable. The ICS is designed as a safety-related system to remove reactor decay heat following reactor shutdown and isolation. This avoids the need for reactor de-pressurization and operation of ECCS, which can also perform this function.

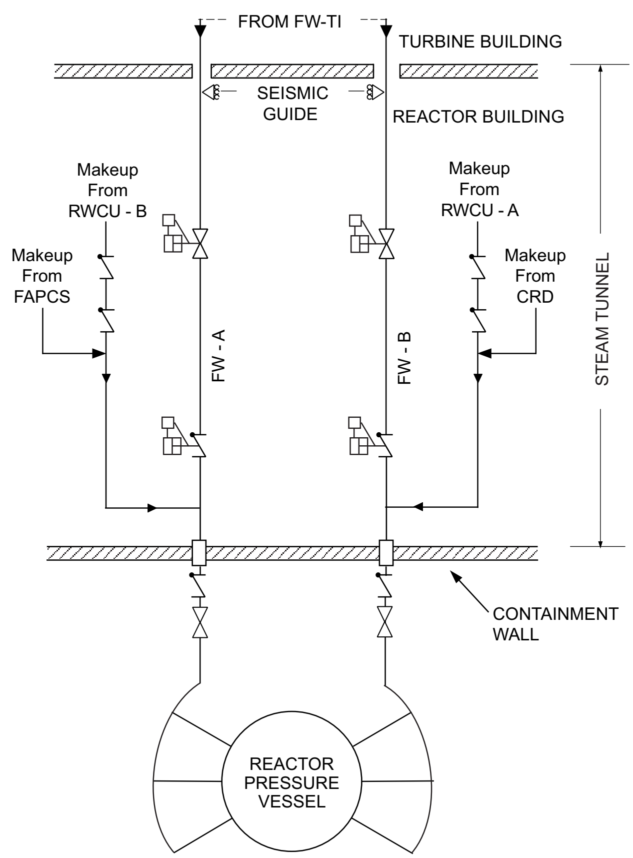


Figure 3-12. Feedwater Configuration (Nuclear Island)

The ICS consists of four totally independent trains, each containing an Isolation Condenser (IC) that condenses steam on the tube side and transfers heat to a large Isolation Condenser/Passive Containment Cooling System (IC/PCCS) pool positioned immediately outside the containment, which is vented to the atmosphere as shown on the ICS schematic (Figure 3-13). The IC, connected by piping to the reactor pressure vessel, is placed at an elevation above the source of steam (vessel) and, when the steam is condensed, the condensate is returned to the vessel via a condensate return pipe. The steam-side connection between the vessel and the IC is normally open and the condensate line is normally closed. This allows the IC, drain piping and an in-line water storage tank to fill with condensate, which is maintained at a subcooled temperature by the pool water during normal reactor operation. The IC is started into operation by opening condensate return valves and draining the condensate to the reactor, thus causing steam from the reactor to fill the tubes which transfer heat to the cooler pool water. Each IC consists of two identical modules.

The steam supply line (properly insulated and enclosed in a guard pipe which penetrates the containment roof slab) is vertical and feeds two horizontal headers through four branch pipes. Each pipe is provided with a built-in flow limiter, sized to allow natural circulation operation of the IC at its maximum heat transfer capacity while addressing the concern of IC breaks downstream of the steam supply pipe. Steam is condensed inside vertical tubes and condensate is collected in two lower headers. Two pipes, one from each lower header, take the condensate to the common drain line which vertically penetrates the containment roof slab.

A vent line is provided for both upper and lower headers to remove the noncondensable gases away from the unit, during IC operation. The vent lines are routed to the containment through a single penetration.

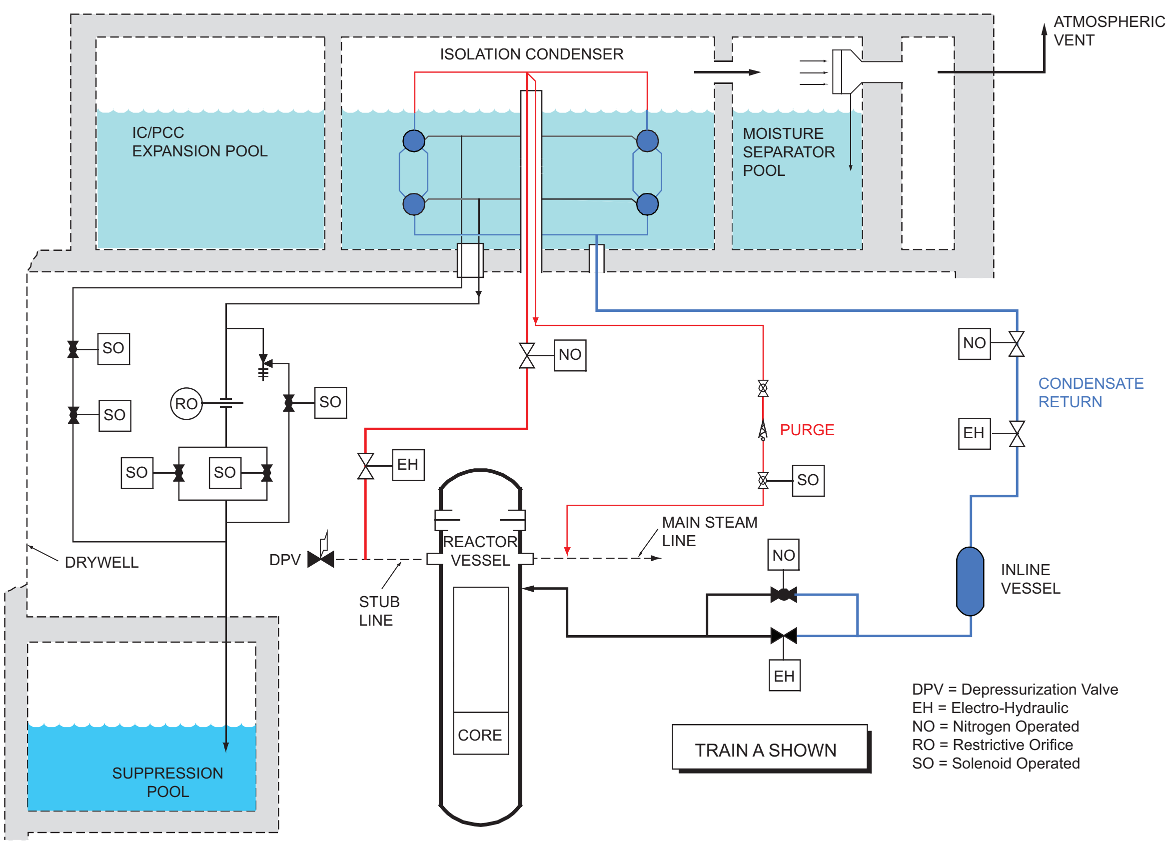


Figure 3-15. Isolation Condenser System (Standby Mode)

A purge line is provided to assure that, during normal plant operation (IC system standby conditions), the excess of hydrogen (from hydrogen water chemistry control additions) or air from the feedwater does not accumulate in the IC steam supply line, thus assuring that the IC tubes are not blanketed with noncondensables when the system is first started. The purge line penetrates the containment roof slab.

Containment isolation valves are provided on the steam supply piping and the condensate return piping.

Figure 3-14. Isolation Condenser Test Module

Located on the condensate return piping just upstream of the reactor entry point is a loop seal and a parallel-connected pair of valves: (1) a condensate return valve (electrohydraulic-operated, fail as is) and (2) a condensate return bypass valve (nitrogen piston operated, fail open). These two valves are closed during normal station power operations. Because the steam supply line valves are normally open, condensate forms in the IC and develops a level up to the steam distributor, above the upper headers. To start an IC into operation, the nitrogen motor-operated condensate return valve and the condensate return bypass valve are opened, whereupon the standing condensate drains into the reactor and the steam-water interface in the IC tube bundle moves downward below the lower headers to a point in the main condensate return line.

The loop seal assures that condensate valves do not have hot water on one side of the disk and ambient temperature water on the other side during normal plant operation, thus affecting leakage during system standby conditions. Furthermore, the loop seal assures that steam continues to enter the IC preferentially through the steam riser, irrespective of water level inside the reactor, and does not move countercurrent back up the condensate return line.

During ICS normal operation, noncondensable gases collected in the IC are vented from the IC top and bottom headers to the suppression pool after a time delay. During ICS standby operation, discharge of hydrogen excess or air is accomplished by a purge line that takes a small stream of gas from the top of the isolation condenser and vents it downstream of the RPV on the main steam line upstream of the MSIVs.

Radiation monitors are provided in the IC/PCC pool steam atmospheric exhaust passages for each IC loop. The radiation monitors are used to detect IC loop leakage outside the containment and cause either alarms or automatic isolation of a leaking IC.

The IC has undergone engineering development testing using a prototype to demonstrate the proper operability, reliability, and heat removal capability of the design over a range of pressures and temperatures. Figure 3-14 shows one of the modules (half of a heat exchanger) under test at the SIET Laboratory facility in Italy.

Chapter 4
Safety Systems

Overview

The ESBWR Safety Systems design incorporates four redundant and independent divisions of the passive Gravity Driven Core Cooling System (GDCS), the Automatic Depressurization System (ADS), and a Passive Containment Cooling System (PCCS). Refer to Figure 4-1. Inventory addition is also provided by the Isolation Condenser System (ICS) and the Standby Liquid Control System (SLCS). The ADS and ICS Systems were discussed in Chapter 3.

The RPV has no external recirculation loops or large pipe nozzles below the top of the core region. This, together with a high capacity ADS, allowed the incorporation of an ECCS driven solely by gravity, with no need for pumps. The water source needed for the ECCS function is stored in the containment upper drywell, with sufficient water to ensure core coverage to 1 meter above the top of active fuel as well as flooding the lower drywell.

The PCCS heat exchangers are located in pools above and immediately outside of containment. There is sufficient water in the external pools to remove decay heat for at least 72 hours following a postulated design basis accident, and provisions exist for external makeup beyond that, if necessary.

As a result of these passive safety simplifications in the ESBWR safety systems, there is an increase in the calculated safety performance margin of the ESBWR over earlier BWRs. This has been confirmed by a Probabilistic Risk Assessment (PRA) for the ESBWR, which shows that the ESBWR is a calculated factor of about 5 lower than ABWR and 50 better than BWR/6 in avoiding possible core damage from degraded events.

In addition to the systems mentioned above, there are other important safety-related components in the ESBWR, including the Emergency Filter Units of the Control Room Habitability HVAC Subsystem (CRHAVS).

Emergency Core Cooling Systems (ECCS)

Gravity Driven Core Cooling System

General

The GDCS is composed of four mechanical trains and four electrical divisions. A single train of the GDCS consists of three independent subsystems: a short-term cooling (injection) system, a long-term cooling (equalizing) system, and a deluge line. The short-term and long-term systems provide cooling water under force of gravity to replace RPV water inventory lost during a postulated LOCA and subsequent decay heat boil-off. The deluge line connects the GDCS pool to the lower drywell. Refer to Figure 4-2.

Each train of the GDCS injection system consists of one 200-mm pipe exiting from the GDCS pool. A 100-mm deluge line branches off and is terminated with three 50-mm squib valves and deluge line tailpipes to flood the lower drywell. The injection line continues after the deluge line connection from the upper drywell region through the drywell annulus where the line branches into two 150-mm branch lines each containing a biased-open check valve and a squib valve.

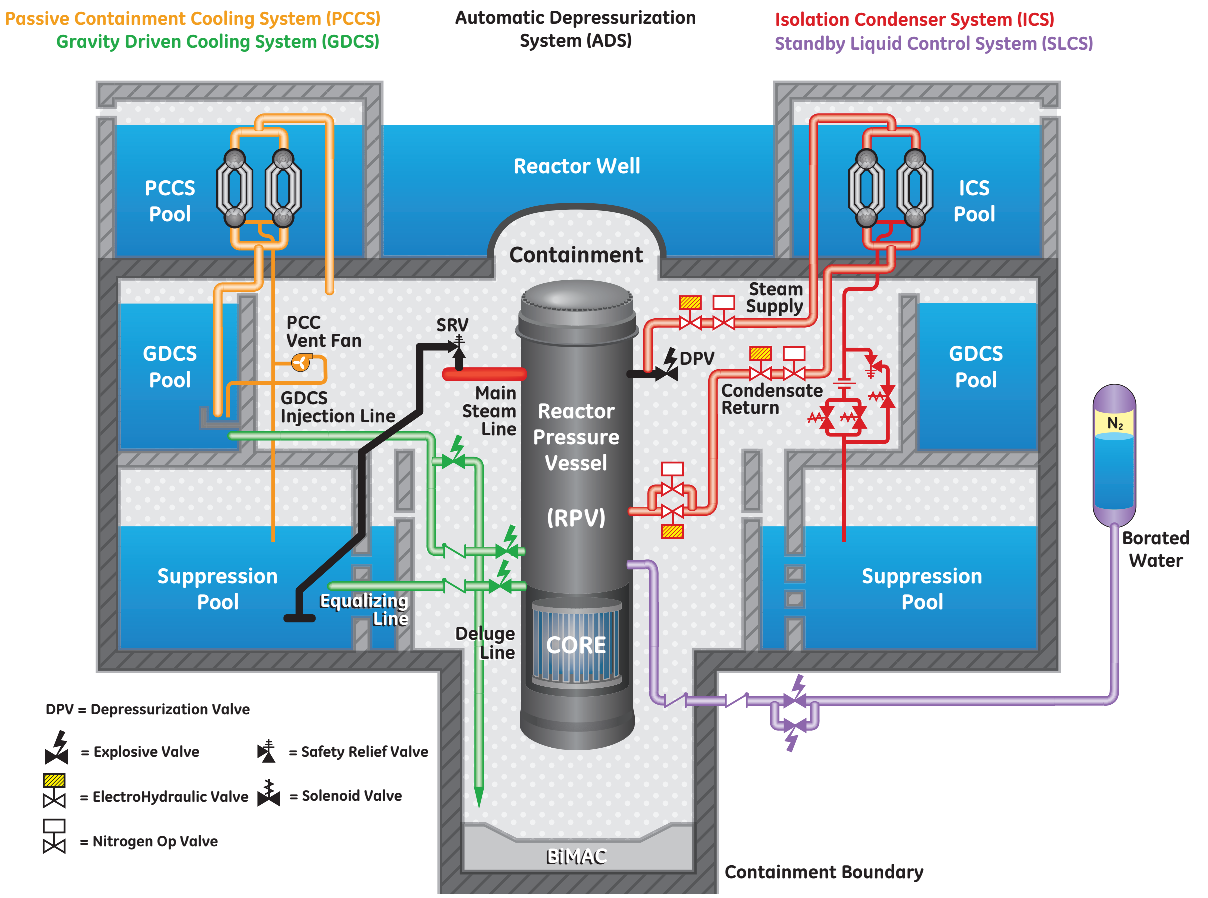


Figure 4-1. ESBWR Key Safety Systems

Each train of the long-term system consists of one 150-mm equalizing line with a check valve and a squib valve, routed between the suppression pool and the RPV. All piping is stainless steel and rated for reactor pressure and temperature. The RPV injection line nozzles and the equalizing line nozzles all contain integral flow limiters.

In the injection lines and the equalizing lines there exists a biased-open check valve located upstream of the squib-actuated valve. The GDCS squib valves are gas propellant type shear valves that are normally closed and which open when a pyrotechnic booster charge is ignited. During normal reactor operation, the squib valve is designed to provide zero leakage. Once the squib valve is actuated it provides a permanent open flow path to the vessel.

The check valves mitigate the consequences of spurious GDCS squib valve operation and minimize the loss of RPV inventory after the squib valves are actuated and the vessel pressure is still higher than the GDCS pool pressure plus its gravity head. Once the vessel has depressurized below GDCS pool surface pressure plus its gravity head, the differential pressure opens the check valve and allows water to begin flowing into the vessel. Refer to Figure 4-3 for a 3D simulation of the GDCS process.

The GDCS deluge lines provide a means of flooding the lower drywell region with GDCS pool water in the event of a postulated core melt sequence, which causes failure of the lower vessel head and allows the molten fuel to reach the lower drywell floor. A core melt sequence would result from a common mode failure of the short-term and long-term systems, which prevents them from performing their intended function. Deluge line flow is initiated by thermocouples, which sense high lower drywell region basemat temperature indicative of molten fuel on the lower drywell floor. Squib-type valves in the deluge lines are actuated upon detection of high basemat temperatures. The deluge lines do not require the actuation of squib-actuated valves on

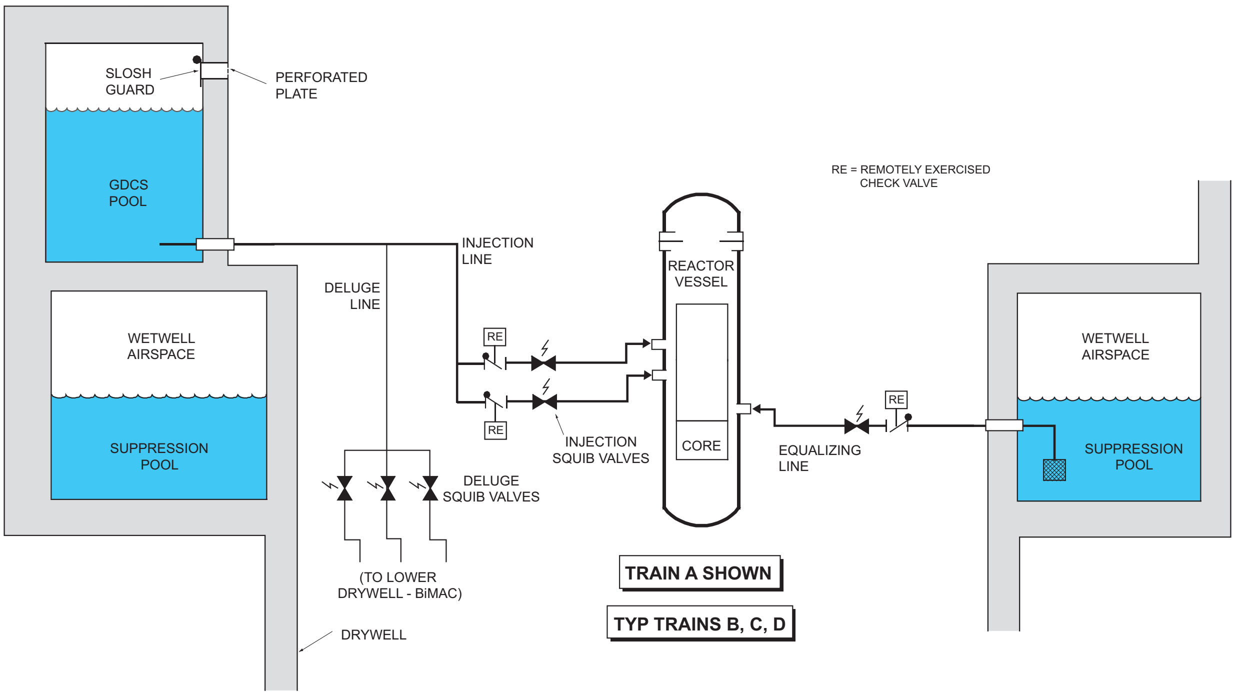


Figure 4-2. Gravity Driven Cooling System Schematic

the injection lines of the GDCS piping to perform their function.

The deluge valves are opened based on very high temperatures in the lower drywell, indicative of a severe accident. Once the deluge valve is actuated it provides a permanent open flow path from the GDCS pools to the lower drywell region. Flow then drains to the lower drywell via permanently open drywell lines. This supports the BiMAC core catcher function (see Chapter 8).

The GDCS check valves remain fully open when zero differential pressure exists across the valve. This is to minimize the potential for sticking in the closed position during long periods of non use.

Suppression pool equalization lines have an intake strainer to prevent the entry of debris material into the system that might be carried into the pool during a large break LOCA. The GDCS pool airspace opening to DW is covered by a perforated plate or equivalent to prevent debris from entering the pool and potentially blocking the coolant flow through the fuel. A slosh guard is added to the opening to minimize any sloshing of GDCS pool water into the drywell following dynamic events.

The GDCS equalizing lines perform the RPV inventory control function in the long term. By closing the loop between the suppression pool and RPV, inventory, which is transferred to the suppression pool either by PCCS condensation shortfall, or by steam condensation in the drywell (which eventually spills back to the suppression pool), can be added back to the RPV.

Equipment and Component Description

The following describes the GDCS squib valve, and deluge valve, which are unique system components that were not used in previous BWR designs.

Squib Valve

The function of the squib valve is to open upon an externally applied signal and to remain in its full open position without any continuing external power source in order to admit reactor coolant makeup into the reactor pressure vessel in the event of a LOCA. These valves also function in the closed position to maintain RPV backflow leak-tight and maintain the reactor coolant pressure boundary during normal plant operation. The valve is a horizontally mounted, straight through, long duration submersible, pyrotechnic-actuated, non-reclosing valve with

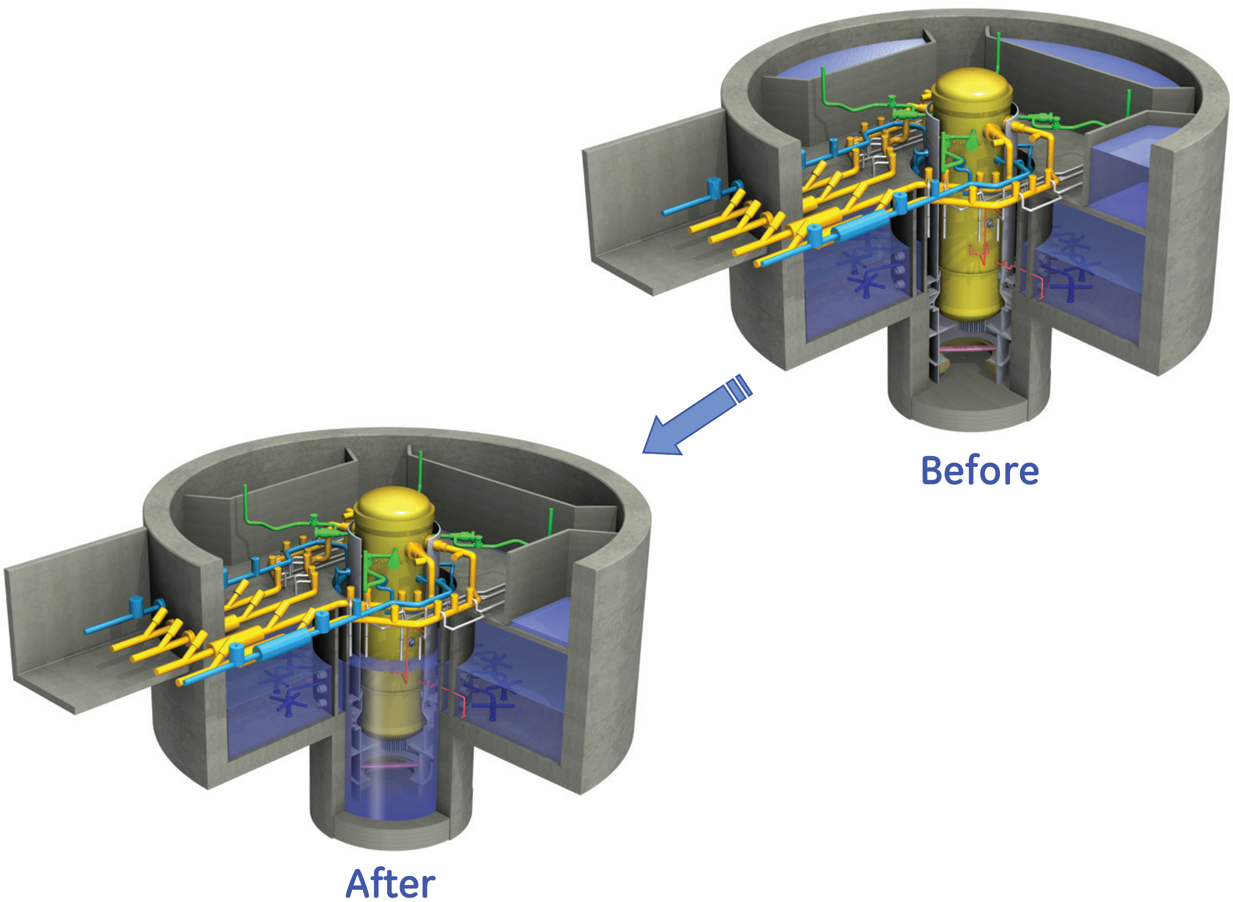


Figure 4-3. GDCS 3D Layout and Action after a Design Basis Accident

metal diaphragm seals and flanged ends. The valve design is such that no leakage is possible across the diaphragm seals throughout the 60-year life of the valve. The squib valve is classified as Quality Group A, Seismic Category I, and ASME Section III Class 1. The valve diaphragm forms part of the reactor pressure boundary and as such is designed for RPV service level conditions.

Illustrated in Figure 4-4 is a typical squib valve design that satisfies GDCS system requirements. This valve has similar design features to the ADS depressurization valve. Valve actuation initiates upon the actuation of any of redundant squib valve initiators, a pyrotechnic booster charge is ignited, and hot gases are produced. When these gases reach a designed pressure, a tension bolt holding a piston breaks allowing the piston to travel downward until it impacts the ram and nipple shear caps. Once the piston impacts the ram and nipple shear caps, the nipples are sheared. The ram and shear caps are then driven forward and are locked in place at the end of stroke by an interference fit with the nipple retainer. This lock ensures that the nipples cannot block the flow stream and provides a simple means of refurbishment by simply unthreading the plug. A switch located on the bottom of the valve provides a method of indication to the control room of an actuated valve. The shear nipple sections are designed to produce clean shear planes. The piston is allowed to backup after shearing the nipples, but, in any case, its forward motion is limited by the housing so that it will not create flow resistance. Standard metal seals are installed on the piston to reduce the potential of ballistic products from entering the flow stream. The squib valve can be completely refurbished once fired. The squib valve housing, nipples, adapter flanges, actuator housing, indicator

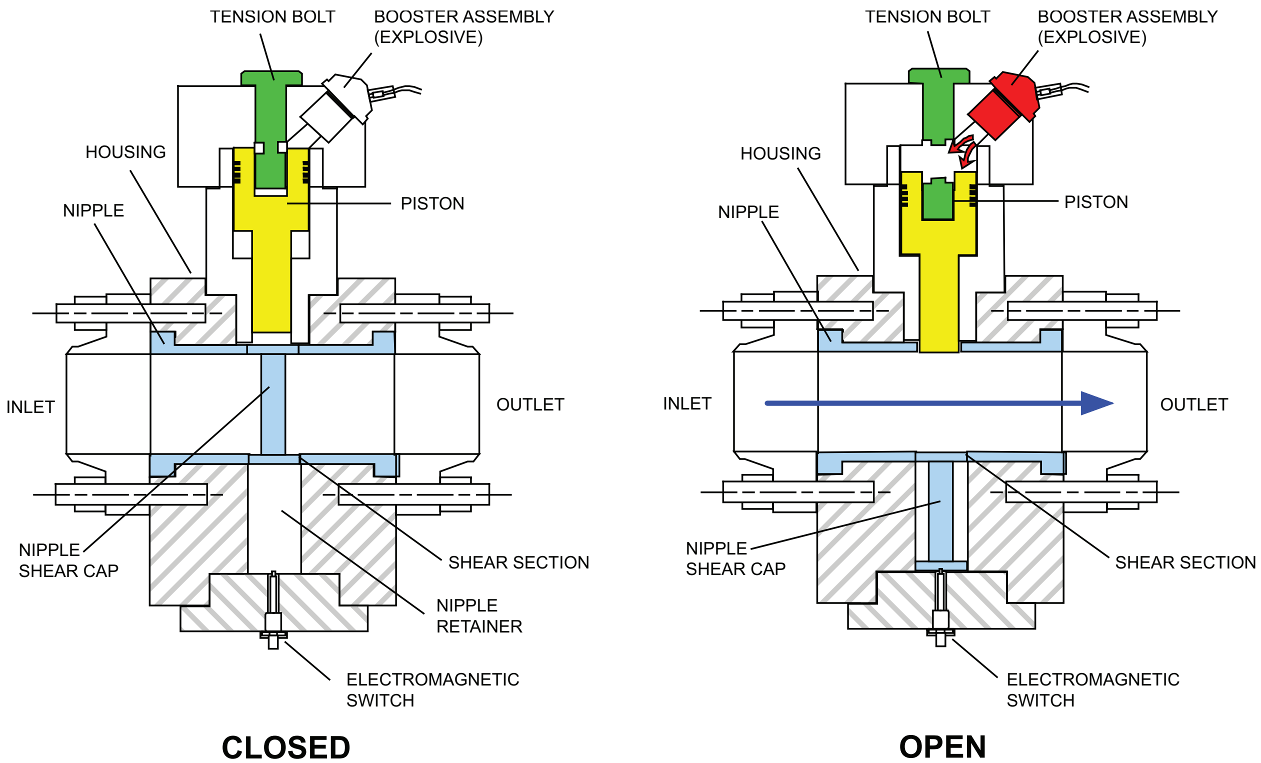


Figure 4-4. Gravity Driven Cooling System (GDCS) Squib Valve - conceptual design

switch body, indicator plunger, head cap, coupling, collar, and adapter are machined. The piston, ram, and tension bolt is made from heat-treated material for necessary strength.

GDCS Check Valve

The GDCS check valves are designed such that the check valve is fully open when zero differential pressure is applied across the check valve. The full open position is accomplished by valve design and installation. The check valve is a long duration submersible valve. The valve meets the minimum flow requirements for a valve stuck in the open position.

Remote check valve position indication is provided in the main control room by position indication instrumentation.

Deluge Valve

The deluge valve is a 50-mm squib valve similar in design to the SLCS squib valves or ADS depressurization valves. To minimize the probability of common mode failure, the deluge valve pyrotechnic booster material is different from the booster material in the other GDCS squib valves.

Automatic Depressurization System (ADS)

The ADS logic is automatically initiated if an RPV low water level signal is sustained for 10 seconds. The ADS logic is also automatically initiated on high drywell pressure that is sustained for one hour.

ADS initiation is accomplished by redundant trip channels arranged in divisionally separated logics. Each SRV or DPV receives initiation signals from three safety-related divisions and also the nonsafety-related Diverse Protection System. This allows safety-related initiation of all valves assuming one division is unavailable due to maintenance and another division is failed. Each SRV is initiated by a solenoid valve controlling a pneumatic actuator. Each DPV is initiated by a squib located on the valve. The ADS valve openings are staggered in time to control the blowdown rate and prevent excessive level swell and reduce suppression pool dynamic loads. Description of the ADS valves can be found in Chapter 3.

For ATWS mitigation, the ADS has an automatic and manual inhibit of ADS initiation to prevent ADS actuation during an ATWS. Automatic initiation of the ADS is inhibited if there is a coincident low reactor water level signal and an Average Power Range Monitor (APRMs) ATWS permissive signal (APRM high), or a coincident high RPV pressure and APRM/ATWS permissive signal. There are also main control room switches for the manual inhibit of automatic initiation of the ADS

GDCS Qualification Tests

Qualification tests of the GDCS were performed in a full-height, scaled volume test facility at GE. Figure 4-5 is a picture of the GDCS Integral System Test (GIST).

Passive Containment Cooling System (PCCS)

The Passive Containment Cooling System (PCCS) maintains the containment within its pressure limits for Design Basis Accidents (DBAs). The PCCS consists of six, low-pressure, totally independent loops, each containing a steam condenser (Passive Containment Cooling Condenser), as shown in Figure 4-6. Each PCCS condenser loop is designed for 7.8 MWt capacity and is made of two identical modules. Together with the pressure suppression containment, the PCCS condensers limit containment pressure to less than its design pressure for at least 72 hours after a LOCA without makeup to the IC/PCC pool, and beyond 72 hours with pool makeup and PCCS vent fan operation (described below). The PCCS condensers are located in a large pool (IC/PCC pool) positioned above, and outside, the ESBWR containment.

Each PCCS condenser loop is configured as follows. A central steam supply pipe is provided which is open to the containment at its lower end, and it feeds two horizontal headers through two branch pipes at its upper end. Steam is condensed inside vertical tubes and the condensate is collected in two lower headers. The vent and drain lines from each lower header are routed to the drywell through a single containment penetration per condenser module as shown on the diagram. The condensate drains into an annular duct around the vent pipe and then flows in a line that connects to a large common drain line, which also receives flow from the other header, ending in a GDCS pool.

The non-condensable vent line is the pathway by which drywell noncondensables are transferred to the wetwell. This ensures a low noncondensable concentration in the steam in the condenser, necessary for good heat transfer. During periods in which PCCS heat removal is less than decay heat, excess steam also flows to the suppression pool via this pathway.

After the first 72 hours of a LOCA, PCCS vent fans are operated to aid in the long-term removal of non-condensable gas from the PCCS for continued condenser efficiency. The discharge of each PCCS vent fan is submerged in the GDCS pool drain pan to prevent backflow that could otherwise interfere with the normal venting of the PCCS. The fans are

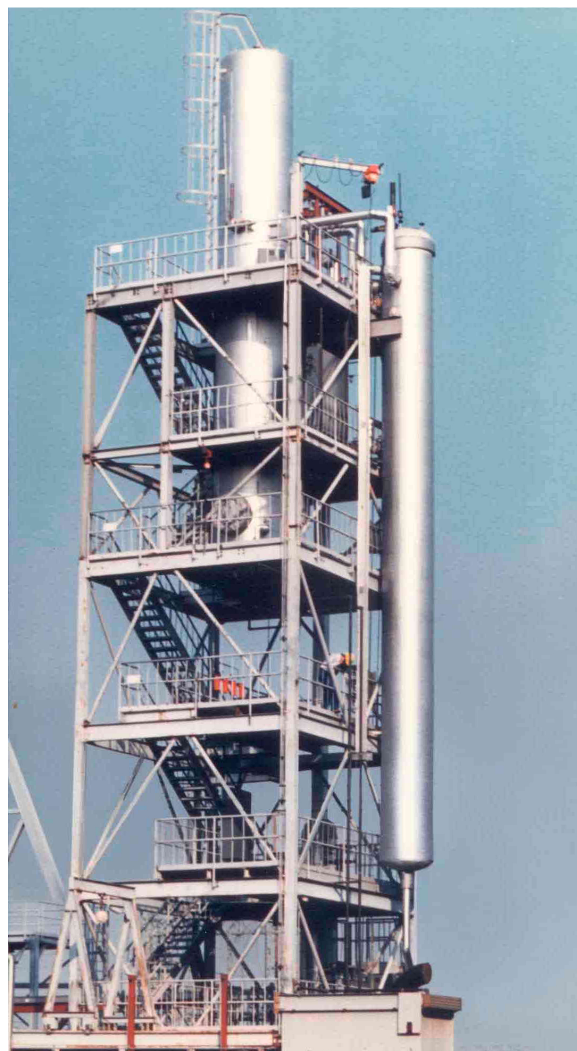


Figure 4-5. GIST Facility

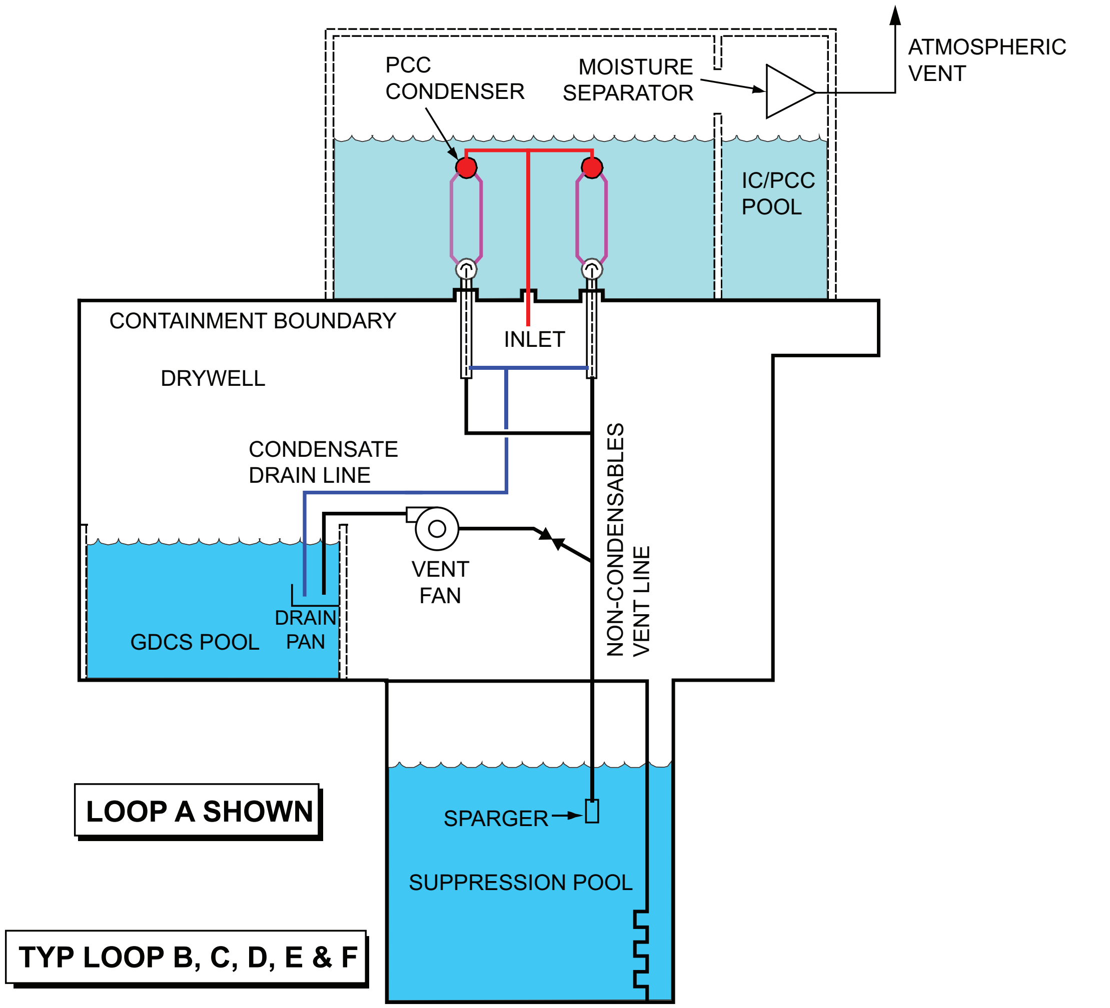


Figure 4-6. Passive Containment Cooling System Schematic

operated by operator action and are powered by a reliable power source provided by an ancillary diesel generator.

The PCCS loops receive a steam-gas mixture supply directly from the DW. The PCCS loops are initially driven by the pressure difference created between the containment DW and the suppression pool during a LOCA and then by gravity drainage of steam condensed in the tubes, so they require no sensing, control, logic, or power-actuated devices to function. The PCCS loops are an integral part of the safety-related containment and do not have isolation valves.

Spectacle flanges are included in the drain line and in the vent line to conduct post-maintenance leakage tests separately from Type A containment leakage tests. Located on the drain line and submerged in the GDCS pool, just upstream of the discharge point, is a loop seal. It prevents backflow of steam and gas mixture from the DW to the vent line, which would otherwise short circuit the flow through the PCCS condenser to the vent line. It also provides long-term operational assurance that the PCCS condenser is fed via the steam supply line.

Each PCCS condenser is located in a subcompartment of the IC/PCC pool, and all pool subcompartments communicate at their lower ends to enable full use of the collective water inventory independent of the operational status of any given IC/PCCS sub-loop. A valve is provided at the bottom of each PCC subcompartment that can be closed so the subcompartment can be emptied of water to allow PCCS condenser maintenance.

Pool water can heat up to about 101°C (214°F); steam formed, being non-radioactive and having a slight positive pressure relative to station ambient, vents from the steam space above each PCCS condenser where it is released to the atmosphere through large-diameter discharge vents. A moisture separator is installed at the entrance to the discharge vent lines to preclude excessive moisture carryover and loss of IC/PCC pool water. IC/PCC pool makeup clean water supply for replenishing level is normally provided from the Makeup Water System.

Level control is accomplished by using an air-operated valve in the makeup water supply line. The valve opening and closing is controlled by water level signal sent by a level transmitter sensing water level in the IC/PCC pool. Cooling and cleanup of IC/PCC pool water is performed by a separate cooling and cleanup subsystem of the Fuel and Auxiliary Pools Cooling System (FAPCS) that is independent from other FAPCS pool cooling and cleanup functions, in order to avoid radioactive contamination of the IC/PCC pool water (see Chapter 5). The FAPCS provides safety-related dedicated makeup piping, independent of any other piping, which provides flow paths for post-accident makeup water transfer from the Fire Protection System (FPS), as well as an attachment connection at grade elevation in the station yard outside the reactor building, whereby a post-LOCA water supply can be connected.

There have been extensive qualification tests of the PCCS, including full-scale component tests and full height scaled integral tests. Figure 4-7 shows a picture of the component testing at the SIET laboratories in Italy.

Standby Liquid Control System (SLCS)

The Standby Liquid Control System (SLCS) provides a backup method to bring the nuclear reactor to subcriticality and to maintain subcriticality as the reactor cools. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner.

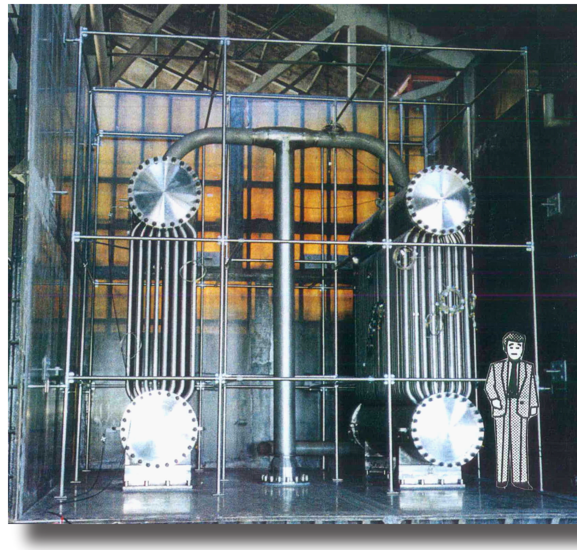


Figure 4-7 PCCS Heat Exchanger Testing

The SLCS is sized to counteract the positive reactivity effect of shutting down from rated power to cold shutdown condition. It also adds additional inventory to the RPV after confirmation of a LOCA.

The SLCS is automatically initiated in case of signals indicative of LOCA or ATWS. It can also be manually initiated from the main control room to inject the neutron absorbing solution into the reactor.

The SLCS is a safety-related passive system incorporating two identical and separate trains using pressurized accumulators to inject borated water rapidly and directly into the bypass area of the core. Each train is 50% capacity. Injection will take place after either of two squib valves in each division fires upon actuation signal from the SSLC. Figure 4-8 illustrates the SLCS configuration.

Supporting nonsafety-related equipment includes a high pressure nitrogen charging system for pressurization and to make up for losses, and a mixing and boron solution makeup system.

The boron absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the fuel. The specified neutron absorber solution is sodium pentaborate using 94% of the isotope B-10 at a concentration of 12.5%. This combination not only minimizes the quantity of liquid to be injected,

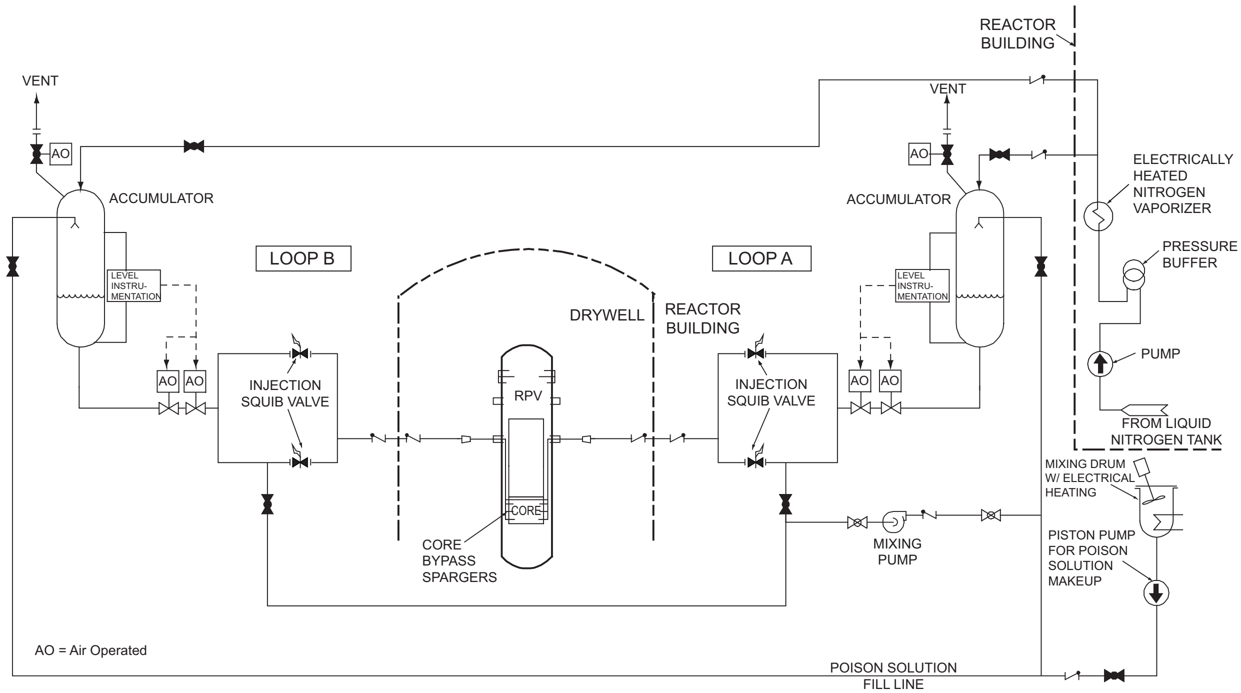


Figure 4-8. Standby Liquid Control System Schematic

but also assures no auxiliary heating is needed to prevent precipitation of the sodium pentaborate out of solution in the accumulator and piping. At all times, when it is possible to make the reactor critical, the SLCS will be able to deliver enough sodium pentaborate solution into the reactor to assure reactor shutdown.

Upon completion of injecting the boron solution, redundant accumulator level measurement instrumentation using 2-out-of-4 logic closes at least one of two injection line shut-off valves in each SLCS division. Closure of these valves prevents injection of nitrogen from the accumulator into the reactor vessel that could interfere with Isolation Condenser System operation or cause additional containment pressurization. As a backup, the accumulator vent valves are also opened at the same time.

Emergency Control Room Habitability

The Control Room Habitability Area* (CRHA) is served by a combination of individual systems that collectively provide the habitability functions. The systems that make up the habitability systems are the:

- CRHA HVAC Subsystem (CRHAVS)
- Process Radiation Monitoring Subsystem (PRMS)
- Lighting System
- Fire Protection System (FPS)

ESBWR design features are provided to ensure that the control room operators can remain in the

* The CRHA includes the plant area in which actions can be taken to operate the plant safely under normal conditions and to maintain the reactor in a safe condition during accident situations. It includes the Main Control Room (MCR) area and areas adjacent to the MCR containing operator facilities.

control room and take actions to safely operate the plant under normal conditions and to maintain it in a safe condition under accident conditions. These habitability features include missile protection, radiation shielding, radiation monitoring, air filtration and ventilation systems, lighting, personnel and administrative support, and fire protection.

The PRMS provides radiation monitoring of the CRHA environment and outside air intake. The FPS provides smoke detection and fire damper isolation. Emergency lighting is provided by the Lighting System. Storage capacity is provided in the main control room for personnel support equipment. Manual hose stations outside the CRHA and portable fire extinguishers provide fire suppression in the CRHA.

The CRHA boundary envelope structures are designed with low leakage construction. The CRHA is located in an underground portion of the Control Building (CB). The boundary walls are adjacent to underground fill or underground internal areas of the CB. The construction consists of cast-in-place reinforced concrete walls and slabs, and is constructed to minimize leakage through joints and penetrations.

Only the habitability portion of the CRHAVS is discussed here. Figure 4-9 provides a schematic diagram of the CRHAVS.

The CRHA emergency habitability portion of the CRHAVS is not required to operate during normal conditions. The normal operation of the CRHAVS maintains the air temperature of the CRHA within a predetermined temperature range. This maintains the CRHA emergency habitability system passive heat sink at or below a predetermined temperature. The normal operation portion of the CRHAVS operates during all modes of normal power plant operation, including startup and shutdown

The Emergency Filter Units (EFUs) are redundant safety-related components that supply filtered air to the CRHA for breathing and pressurization to minimize in-leakage. The EFUs and their related components form a safety-related subset of the CRHAVS.

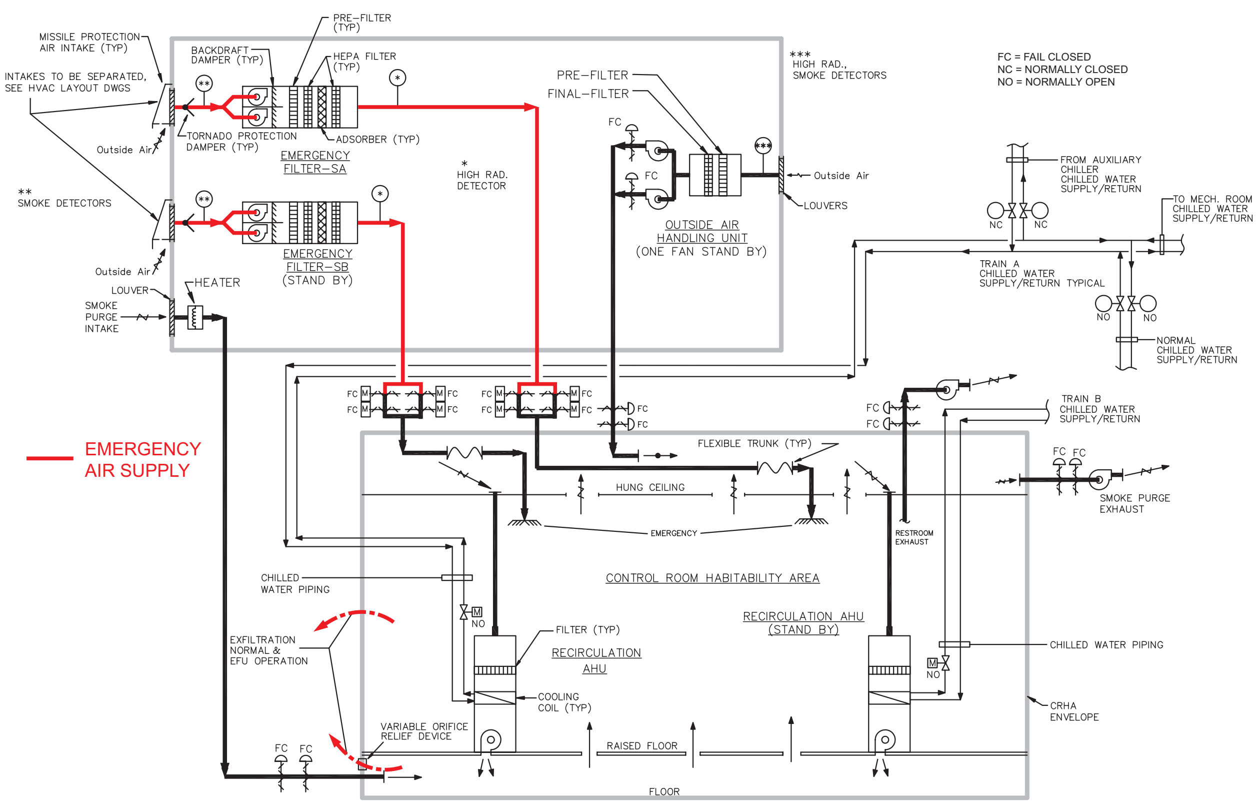


Figure 4-9. Control Room Habitability Area HVAC Subsystem (CRHAVS) Schematic

The EFU outside air supply portion of the CRHAVS is safety-related and Seismic Category I. Single active failure protection is provided by the use of two trains, which are physically and electrically redundant and separated. In the event of failure in one train, the failed train is isolated and the alternate train is automatically initiated. Both trains are 100% capacity and capable of supplying 99% credited efficiency filtered air to the CRHA pressure boundary at the required flow rate.

The EFU design utilizes a Pre-filter, HEPA filter, Carbon Filter and Post-filter to provide radiological protection of the CRHA outside air supply. The EFU design incorporates an upstream fan to maintain the entire filtration sequence and air delivery duct to the CRHA under positive pressure.

Operation of the emergency habitability portion of the CRHAVS is automatically initiated by either of the following conditions:

- High radioactivity in the main control room supply air duct, or
- Extended loss of AC power

Operation can also be initiated by manual actuation. If radiation levels in the main control room supply air duct exceed the high setpoint, the normal outside air intake and restroom exhaust are isolated from the CRHA pressure boundary by automatic closure of the isolation devices in the system ductwork. At the same time, an EFU begins to deliver filtered air from one of the two unique safety-related outside air intake locations. A constant air flow rate is maintained and this flow rate is sufficient to pressurize the CRHA boundary to at least 31 Pa (1/8-inch water gauge) positive differential pressure with respect to the surroundings. The EFU system air flow rate is also sufficient to supply the ASHRAE Standard 62 fresh air requirement by providing 10.5 liters (20 cfm) per person for up to 21 occupants (220 liters total).

With a source of AC power available, an EFU can operate indefinitely. In the event that normal AC power is not available, the safety-related battery power supply is sized to provide the required power to an EFU fan for 72 hours of operation. The CRHA isolation dampers fail closed on a loss of AC power or instrument air. Backup power to the safety-related CR EFU fans (post 72 hours) is provided by two (2) ancillary diesel generators.

Upon a loss of preferred power or Station Blackout (SBO), most of the equipment in the MCR remains powered by the nonsafety-related battery supply for the first two hours. During the first two hours the environmental conditions are maintained within the normal limits. This is accomplished via the continued operation of the CRHA recirculation Air Handling Units (AHU) and auxiliary cooling unit supplied with each recirculation AHU. The cooling function for this two-hour period is not a safety function. If this cooling function is lost, the N-DCIS components in the MCR are automatically de-energized. This is accomplished via safety-related temperature sensors with two-out-of-four logic that automatically trip the power to selected N-DCIS components in the MCR, thus removing the heat load due to these sources.

If power remains unavailable beyond two hours, the remaining CRHA equipment heat loads are dissipated passively to the CRHA heat sinks. During emergency operation, the CRHA emergency haitability system passive heat sink is designed to limit the temerature inside the CRHA to 33.9°C (93°F). The CRHA is passively cooled by conduction into the CHRA heat sink. The heat sinks consist of the following: the CHRA walls, floor, ceiling, interior walls, access corridors, adjacent Q-DCIS and N-DCIS equipment rooms and electrical chases; and the CRHA HVAC equipment rooms and HVAC chases. Sufficient thermal mass is provided in the CHRA heat sink to absorb the heat generated by the equipment, lights, and occupants.

Chapter 5
Auxiliary Systems

Overview

The main auxiliary systems in the ESBWR Nuclear Island are: Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDC), Fuel and Auxiliary Pools Cooling System (FAPCS), Reactor Component Cooling Water System (RCCWS), Plant Service Water System (PSWS), and Drywell Cooling System (DCS). There are many other Nuclear Island and non-Nuclear Island auxiliary systems, such as instrument and service air, condensate and demineralized water transfer, chilled water, HVAC, equipment drain, floor drain and other systems which are basically the same as on past BWR plants and are not covered here, since the designs are all well known.

Reactor Water Cleanup /Shutdown Cooling System (RWCU/SDC)

The Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System performs two basic functions, reactor water cleanup and shutdown cooling, which include the following major activities:

- Purify the reactor coolant during normal operation and shutdown
- Supplement reactor cooling when the reactor is at high pressure in the hot standby mode
- Assist in the control of reactor water level during startup, shutdown, and in the hot standby mode
- Induce reactor coolant flow from the reactor vessel bottom head to reduce thermal stratification during startup
- Provide shutdown cooling and cooldown to cold shutdown conditions
- Provide heated primary coolant for RPV hydrostatic testing and reactor startup
- Provide long-term post-LOCA containment cooling with cross-connection to FAPCS

System Description

The RWCU/SDC system is comprised of two independent pump-and-purification equipment trains (Figure 5-1). These trains together provide redundant cleanup capacity such that each pump train and demineralizer is designed to achieve and maintain the reactor water quality within design specifications. The system processes the water in the primary system during all modes of operation including startup, normal power generation, cooldown and shutdown operation. The capacity of each train for reactor water cleanup is 1% of the rated feedwater flow rate.

During normal plant operation, the RWCU/SDC system continuously recirculates water taking suction from the mid-vessel area of the RPV and from the reactor bottom head and returning via the feedwater line to the RPV. The reactor water is cooled by flowing through the tube-side of the Regenerative Heat Exchanger (RHX) and the Non-Regenerative Heat Exchanger (NRHX) before entering the RWCU/SDC pump suction. The pump discharges the flow to the demineralizer for the removal of impurities and returns and reheats the reactor water via the shell-side of the RHX.

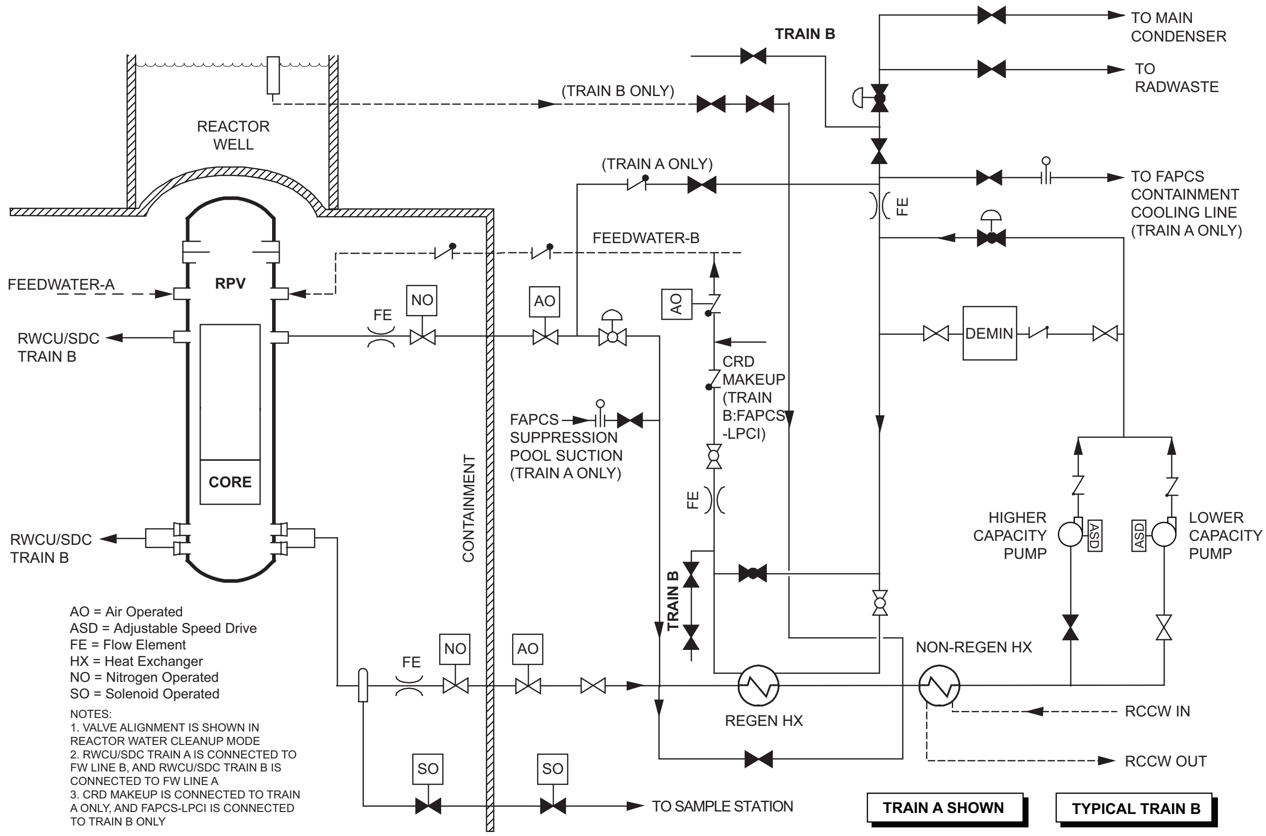


Figure 5-1. Reactor Water Cleanup/Shutdown Cooling System Schematic

Each train of the RWCU/SDC system performs the two functions of reactor water cleanup and shutdown cooling with a common piping system. The RWCU/SDC system suction line from reactor bottom head up to and including the outboard isolation valve, reactor bottom flow sample line up to and including the outboard isolation valve, pumps, demineralizer, pump suction line including suction valves up to and including the demineralizer downstream isolation valve, demineralizer bypass valve and upstream piping are constructed of stainless steel. The remaining system is constructed of carbon steel.

During reactor startup, while maintaining the flow within the cooling capacity of the NRHX, the flow from the demineralizers can be directed to the main condenser hotwell or the liquid radwaste system low conductivity tank for the removal of reactor water that thermally expands during heatup and for removal of inflow from the Control Rod Drive (CRD) system to the RPV.

For RPV hydrotesting and startup, external heating of the reactor water is required if decay heat is not available or the heatup rate from decay heat would be too slow. Feedwater (aided by an auxiliary boiler, if necessary) is used to heat the reactor and reactor water, while colder water in the vessel is overboarded by the RWCU/SDC system.

System Components

The supply side of the RWCU/SDC system is designed for the RCPB design pressure plus 10%. Downstream of the pumps, the pump shutoff head at 5% overspeed is added to the supply side design pressure.

The RWCU/SDC system includes the following major components:

- Demineralizers
- Pumps and adjustable speed motor drives
- Non-regenerative heat exchangers

- Regenerative heat exchangers
- Valves and piping

Demineralizers — The RWCU/SDC system has a mixed bed demineralizer. A full shutdown flow bypass line with a flow control valve is provided around each demineralizer unit for bypassing these units whenever necessary. Resin breakthrough to the reactor is prevented by a strainer in the demineralizer outlet line to catch the resin beads. Non-regeneration type resin beads are used, minimizing the potential for damaged beads passing through the strainer to the reactor. The demineralizer is protected from high pressure differential by a bypass valve. The demineralizer is protected from excessive temperature by automatic controls that first open the demineralizer bypass valve and then close the demineralizer inlet valve. When it is desired to replace the resin, the resin vessel is isolated from the rest of the system before old resin removal and new resin addition.

Pumps and Adjustable Speed Drives (ASD) — The RWCU/SDC pumps are each powered from an ASD. The ASDs receive power at constant AC voltage and frequency. The ASDs convert this to a variable frequency and voltage in accordance with a demand signal. The variable frequency and voltage is supplied to vary the speed of the pump motor. The ASD allows effective control of cooldown rate and reactor temperature after cooldown. The higher capacity pump is used primarily for shutdown cooling and the lower capacity pump is used primarily for reactor water cleanup.

Non-Regenerative Heat Exchanger — Each NRHX cools the reactor water by transferring heat to the RCCWS.

Regenerative Heat Exchanger — Each RHX is used to recover sensible heat in the reactor water and to reduce the closed-loop heat loss and avoid excessive thermal stresses and thermal cycles of the feedwater piping.

System Operation - Cleanup Mode

The modes of operation for the cleanup function are described below.

Power Operation — During normal power operation, reactor water flows from the reactor vessel and is cooled while passing through the tube- side of the RHXs and the tube-side of the NRHXs. The RWCU/SDC pumps then pump the reactor water through the demineralizers, and back through the RHX shell side where the reactor water is reheated and is returned to the reactor vessel via the feedwater lines.

Startup — During heatup, feedwater is introduced in the reactor to raise its temperature, while cold water is overboarded to the main condenser by the RWCU/SDC system. The system is designed to provide sufficient flow through the bottom head connections during heatup, cooldown, and startup operations to prevent thermal stratification and to prevent crud accumulation.

During reactor startup, it is necessary to remove the CRD purge water injected into the RPV and also the excess reactor water volume arising from thermal expansion. The RWCU/SDC system accomplishes these volume removals and thereby maintains proper reactor level until steam can be sent to the turbine and main condenser.

After warmup, the RPV pressure is brought to saturation by opening the vessel to the main condenser through the main steam and turbine bypass lines to promote deaeration of the reactor water. The RWCU/SDC system normally removes excess water by dumping (overboarding) to the condenser hotwell. If the demineralizer is bypassed, the radwaste system is used as an alternative flow path to avoid contaminated coolant from entering the condensate system.

Overboarding — During hot standby and startup, water entering the reactor vessel from the CRD System or water level increase due to thermal expansion during plant heatup, may be dumped (overboarded) to the main condenser to maintain reactor water level. Overboarding of reactor water is accomplished by using one of the two system trains for overboarding and the other train for the reactor water cleanup function.

The train in the overboarding mode uses a combination of RWCU/SDC pump flow and pressure control to maintain the reactor water level. A pressure control station is located downstream of the demineralizer. The pressure control station consists of a pressure control valve, a high pressure restriction orifice, an orifice bypass valve, and a main condenser isolation valve.

Reactor water level is automatically controlled by controlling the pump speed and the pressure control valve position through a combination of flow, level, and pressure control signals. During the early phases of startup, when the reactor pressure is low, the restriction orifice is bypassed. The restriction orifice bypass valve automatically closes when the pressure upstream reaches a predetermined set point to ensure the pressure drop across the pressure control valve and the orifice bypass valve are maintained within their design limits.

During overboarding, the RHX is bypassed since there is no return flow to the RPV, and the NRHX is in service to cool the reactor water to minimize flashing and two-phase flow in the pressure reducing components and downstream piping. The demineralizer is also in service to ensure the water overboarded to the condenser meets water quality specification requirements. In the event high radiation is detected downstream of the demineralizer, the overboarding flow is manually shifted to the Liquid Waste Management System (LWMS) by first opening the remote manual isolation valve to the radwaste system and then closing the remote manual system isolation valve to the main condenser.

Refueling — During refueling, when the reactor well water may have a stratified layer of hot water on the surface, the RWCU/SDC system can be used to supplement the FAPCS to cool the reactor well water.

System Operation - Shutdown Cooling Mode

In conjunction with the heat removal capacity of either the main condenser and/or the isolation condensers, the RWCU/SDC system can reduce the RPV pressure and temperature during cooldown operation from the rated design pressure and temperature to below boiling at atmospheric pressure in less than one day. The system can be connected to nonsafety-related standby AC power (diesel-generators), allowing it to fulfill its reactor cooling functions during conditions when the preferred power is not available.

The shutdown cooling function of the RWCU/SDC system provides decay heat removal capability at normal reactor operating pressure as well as at lower reactor pressures. The redundant trains of RWCU/SDC permit shutdown cooling even if one train is out of service; however, cooldown time is extended when using only one train. In the event of loss of preferred power, the RWCU/SDC system, in conjunction with the isolation condensers, is capable of bringing the RPV to the cold shutdown condition in a day and a half, assuming the most limiting single active failure, and with the isolation condensers removing the initial heat load.

The modes of operation of the shutdown cooling function are described below:

Normal Plant Shutdown — The operation of the RWCU/SDC system at high reactor pressure reduces the plant reliance on the main condenser or ICS. The entire cooldown is controlled automatically. As cooldown proceeds and reactor temperatures are reduced, pump speeds are increased and various bypass valves are opened, as described below. During the early phase of shutdown, the RWCU/SDC pumps operate at reduced speed to control the cooldown rate to less than the maximum allowed RPV cooling rate.

In order to maintain less than the maximum allowed RPV cooling rate, the RWCU/SDC pumps and system configuration is aligned to provide a moderate system flow rate. The flow rate for each train is gradually increased as RPV temperature drops. To accomplish this, in each train the bypass line around the RHX and the bypass line around the demineralizer are opened to obtain the quantity of system flow required for the ending condition of the shutdown cooling mode. In addition to the RCCWS inlet valve to each NRHX being open, at an appropriate point, the air-operated RCCWS bypass control valve to each NRHX will start to close in order to increase the cooling water supply to each NRHX.

The automatic reactor temperature control function controls the ASD, controlling the cooldown by gradually increasing the speed of the system pumps up to the maximum pump flow. Water purification operation is continued without interruption. Over the final part of the cooldown, maximum flow is developed through the RWCU/SDC pumps. After about two weeks, flow rate reduction becomes possible while maintaining reactor coolant temperatures within target temperature ranges.

CRD System flow is maintained to provide makeup water for the reactor coolant volume contraction that occurs as the reactor is cooled down. The RWCU/SDC system overboarding line is used for fine level control of the RPV water level as needed.

Hot Standby — During hot standby, the reactor is at rated pressure and shut down. The RWCU/SDC system may be used as required, in conjunction with the main condenser or isolation condensers, to maintain a nearly constant reactor temperature by processing reactor coolant from the reactor bottom head and the mid-vessel region of the reactor vessel and transferring the decay heat to the RCCWS by operating both RWCU/SDC trains and returning the purified water to the reactor via the feedwater lines.

The pumps and the instrumentation necessary to maintain hot standby conditions are connectable to the Standby AC Power supply during any loss of preferred power.

Refueling — The RWCU/SDC system can provide additional cooling of the reactor well water when the RPV head is off in preparation for removing spent fuel from the core.

Operation Following Transients — In conjunction with the isolation condensers, the system has the capability of removing the core decay heat, plus drain excess makeup due to the CRD purge flow, after one-half hour following control rod insertion.

Post-LOCA Shutdown (With Fuel Failure) — The preferred method of reaching and maintaining long-term containment cooling and cold shutdown after a LOCA is the FAPCS. In the unlikely event there has been a fuel failure, the RWCU/SDC system can be cross-connected to the FAPCS suppression pool suction and the FAPCS containment cooling line to provide containment cooling capabilities (Figure 5-1). This will allow containment cooling while maintaining the contaminated water inside the reactor building. Additionally, the RWCU/SDC system has the capability to return cooled suppression pool water to the reactor vessel through the RWCU mid-vessel suction to preclude using the feedwater injection flowpath, which exits the reactor building.

For this mode of operation, the RWCU/SDC system requires manual realignment of cross-connections with the FAPCS. Each cross-connection contains spectacle flanges and closed manual isolation valves. These provisions preclude the possibility of intersystem LOCA during normal modes of operation. There is also an intersystem cross-connection, which must be realigned for mid-vessel injection. The NRHX provides the heat removal capacity to sufficiently cool the plant from stable shutdown conditions to cold shutdown conditions.

Additional Consideration

In addition to the MS and Feedwater Systems, RWCU/SDC is the only other normally operating process system with primary system high pressure water located outside the containment. Therefore, special attention is paid to providing prompt system isolation in case of a postulated system pipe break in the Reactor Building. Inlet and outlet flows are measured and the difference, if large, will cause containment isolation valves to close. As an additional precaution, there is a third remote manual valve located outside the containment which can be used to effect isolation.

Fuel and Auxiliary Pools Cooling System (FAPCS)

The Fuel and Auxiliary Pools Cooling System (FAPCS) consists of two 100% cooling and cleaning (C/C) trains, each with a pump, a heat exchanger and a water treatment unit for cooling and cleaning of various cooling and storage pools except for the Isolation Condenser and Passive Containment Cooling System (IC/PCCS) pools (Figure 5-2). A separate subsystem with its own pump, heat exchanger, and water treatment unit is dedicated for cooling and cleaning of the IC/PCCS pools independent of the FAPCS C/C train operation during normal plant operation (Figure 5 3) in order to prevent radioactive contamination of these pools.

The primary design function of the FAPCS is to cool and clean pools located in the containment, reactor building, and fuel building during normal plant operation. The FAPCS provides flow paths for filling and makeup of these pools during normal plant operation and during post accident conditions, as necessary.

The FAPCS C/C train is also designed to provide the following accident recovery functions in addition to the spent fuel pool cooling function:

- Suppression pool cooling (SPC)
- Drywell spray
- Low pressure coolant injection of suppression pool water into the RPV
- Alternate Shutdown Cooling

During normal plant operation, at least one FAPCS C/C train is available for continuous operation to cool and clean the water of the spent fuel pool, while the other train can be placed in standby or other mode for cooling the Gravity Driven Cooling System (GDCS) pools and suppression pool. If necessary during refueling outage, both trains may be used to provide maximum cooling capacity for cooling the spent fuel pool.

Each FAPCS C/C train has sufficient flow and cooling capacity to maintain spent fuel pool bulk

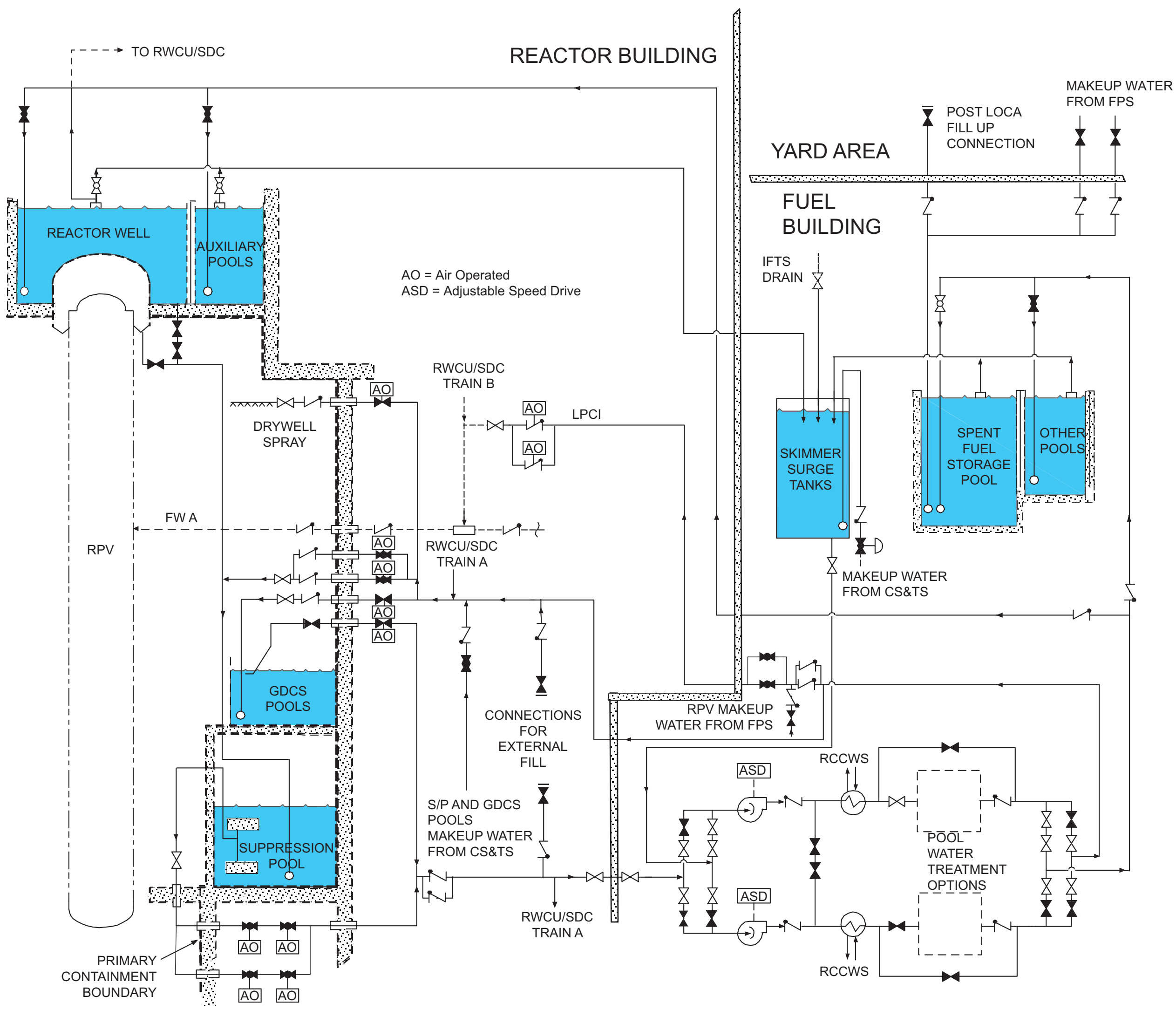


Figure 5-2. Fuel and Auxiliary Pools Cooling System Schematic

water temperature below 48.9°C (120°F) under normal spent fuel pool heat load conditions. During the maximum spent fuel pool heat load conditions of a full core off-load plus irradiated fuel in the spent fuel pool resulting from 20 years of plant operations, both FAPCS C/C trains are needed to maintain the bulk temperature below 60°C (140°F).

All operating modes are manually initiated and controlled from the main control room (MCR), except the SPC mode, which is initiated either manually or automatically on high suppression pool water temperature signal. Instruments are provided for indication of operating conditions to aid the operator during the initiation and control of system operation. Provisions are provided to prevent inadvertent draining of the pools during FAPCS operation.

System Operation

The following discuss the major design operating modes of FAPCS.

Spent Fuel Pool Cooling and Cleanup — One of the FAPCS C/C trains is continuously operated in this mode to cool and clean the water in the spent fuel pool during normal plant operation and during a refueling outage. This mode may be initiated following an accident to cool the fuel pool for accident recovery. During this mode of operation, water is drawn from the skimmer surge tanks, pumped through the heat exchanger and water treatment unit to be cooled and cleaned and then returned to the spent fuel pool. When necessary, a portion or all of the water may bypass the water treatment unit.

Fuel and Auxiliary Pool Cooling and Cleanup — During a refueling outage, one or both FAPCS C/C trains are placed in this mode of operation to cool and clean the water in the spent fuel pool and pools listed below depending on the heat load condition in these pools. During this mode of operation, water is drawn from the skimmer surge tanks, pumped through the heat exchanger and water treatment unit to be cooled and cleaned and then returned to these pools. When necessary, a portion or all of the water may bypass the water treatment unit. This applies to:

- Upper fuel transfer pool
- Buffer pool

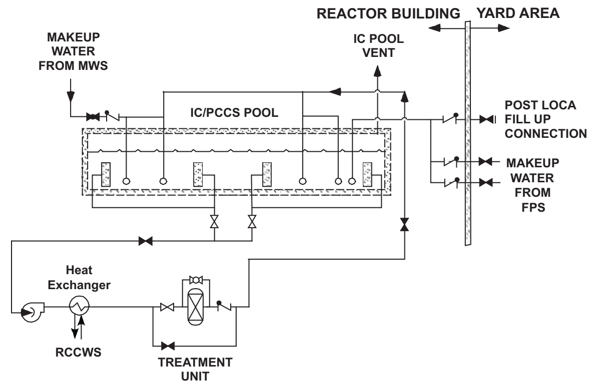


Figure 5-3. FAPCS IC Subsystem Schematic

- Reactor well
- Dryer and separator storage pool

IC/PCCS Pool Cooling and Cleanup — As necessary during normal plant operation, the separate IC/PCCS pool C/C subsystem is placed in this mode. During this mode of operation, water is drawn via a common suction header from IC/PCCS pools. Water is cooled and cleaned by the IC/PCCS pool C/C subsystem and is then returned to the pools through a common line that branches and discharges deep in the pools.

GDCS Pool Cooling and Cleanup — As necessary during normal plant operation, one of the FAPCS C/C trains that is not operating in spent fuel pool cooling mode is placed in this mode. In this mode of operation, water is drawn from GDCS pools A and D. The water is cooled and cleaned and is then returned to GDCS pool B/C. During the operation, the water level in the GDCS pool B/C rises and the water is cascaded and discharged at a submerged location in the adjacent GDCS pools A and D.

Suppression Pool Cooling and Cleanup — As necessary during normal plant operation, one of the FAPCS C/C trains that is not operating in spent fuel pool cooling mode is placed in this mode. In this mode of operation, water drawn from the suppression pool is cooled and cleaned and then returned to the suppression pool. This mode may be automatically initiated during normal operation in response to a high temperature signal from the suppression pool. This mode may be manually initiated following an accident to cool the suppression pool for accident recovery.

Low Pressure Coolant Injection (LPCI) — This mode may be initiated following an accident after the reactor has been depressurized to provide reactor makeup water for accident recovery. In this mode the FAPCS pump takes suction from the suppression pool and pumps it into the reactor vessel via RWCU/SDC loop B and then Feedwater loop A. Alternatively, a separate motor-driven pump in the fire pump enclosure can take suction from the fire protection storage tank and pump water into the reactor vessel via a tie in with the primary LPCI flow path.

Alternate Shutdown Cooling — This mode may be initiated following an accident for accident recovery. In this mode, FAPCS operates in conjunction with other systems to provide reactor shutdown cooling in the event of loss of other shutdown cooling methods. During this mode of operation, FAPCS flow path is similar to that of LPCI mode. Water is drawn from the suppression pool, cooled, and then discharged back to the reactor vessel via LPCI injection flow path. The warmer water in the reactor vessel rises and then overflows into the suppression pool via two opened safety-relief valves on the main steam lines A and B, completing the loop for this mode of operation.

Drywell Spray — This mode may be initiated following an accident for accident recovery. During this mode of operation, FAPCS draws water from the suppression pool, then cools and sprays the cooled water to drywell air space to reduce the containment pressure.

A crosstie to the RWCU/SDC System is also provided in the FAPCS suppression pool suction and the FAPCS discharge headers such that the RWCU/SDC may be used as an alternative for post-accident decay heat removal in the unlikely event of fuel failure from an accident. This allows decay heat removal while maintaining the contaminated water inside the reactor building; refer to previous discussion for RWCU/SDC

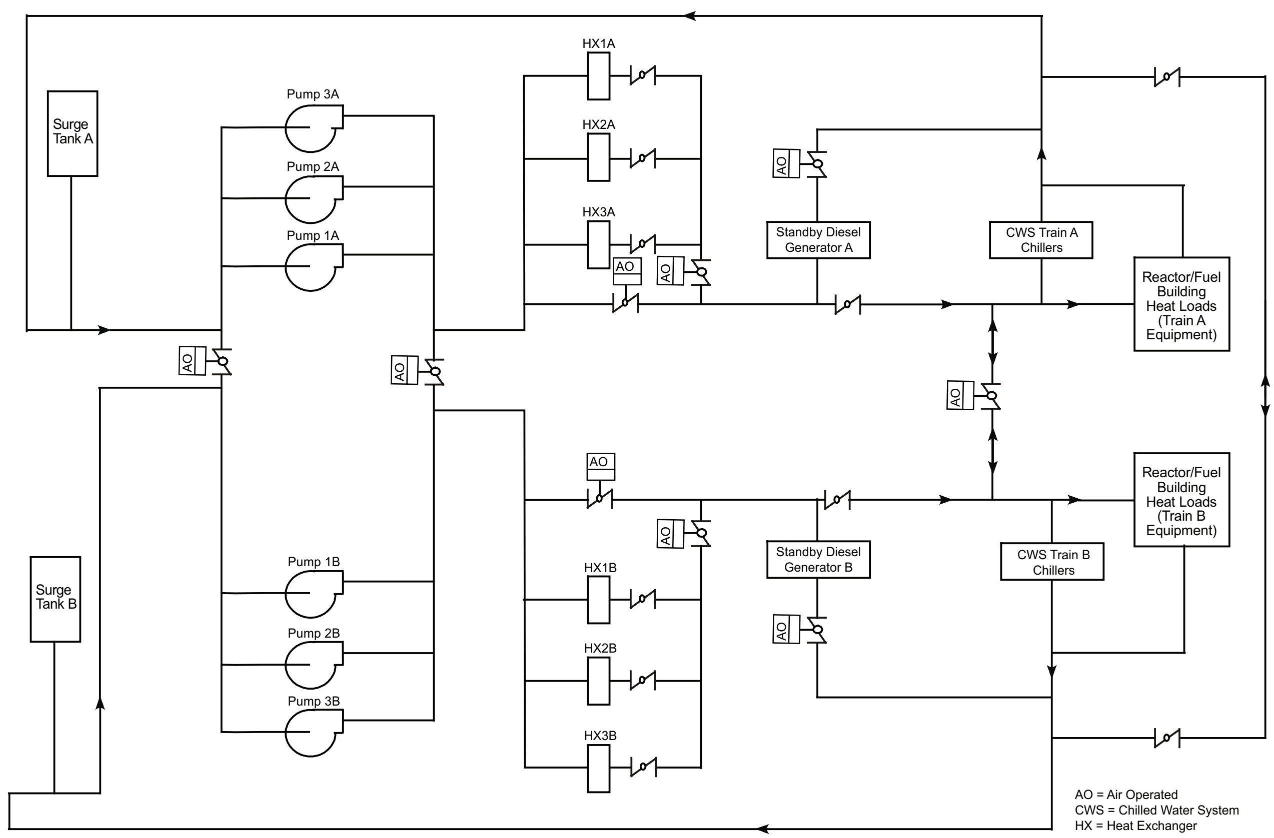


Figure 5-4. Reactor Component Cooling Water System Schematic

The FAPCS is a nonsafety-related system with the exception of piping and components required for containment isolation and refilling of the IC/PCCS pools and the spent fuel pool with emergency water supplies from offsite.

Reactor Component Cooling Water System (RCCWS)

The Reactor Component Cooling Water System (RCCWS) provides cooling water to nonsafety-related components in the Reactor, Fuel, Electrical, and Radwaste Building and provides a barrier against leakage of radioactive contamination of the Plant Service Water System (PSWS).

The RCCWS consists of two 100% capacity independent and redundant trains (Figure 5-4). RCCWS cooling water is continuously circulated through various auxiliary equipment heat exchangers and rejects the heat to the PSWS. In the event of a loss of preferred power (LOPP), the RCCWS supports the FAPCS and the RWCU/SDC in bringing the plant to cold shutdown condition in 36 hours if necessary, assuming the most limiting single active failure. In addition, RCCWS provides cooling water to the Standby Onsite Power System Diesel Generators.

Each RCCWS train consists of parallel pumps, parallel heat exchangers, one surge tank, connecting piping, and instrumentation. Both trains share a chemical addition tank. The two trains are normally connected by crosstie piping during operation for flexibility, but may be isolated for individual train operation or maintenance of either train. The pumps in each train discharge through check valves and butterfly valves to a common header leading to the RCCWS heat exchanger header. Crosstie lines between each train are provided up and downstream of the heat exchangers; at the pump suction and discharge headers; and downstream of the standby diesel generators cooling water supplies. The heat exchanger outlet isolation valves are provided. The heat exchanger flow control valves, bypass temperature control valves, and cross-tie isolation valves are pneumatically operated.

RCCWS cooling water is supplied to the following major users:

- Chilled Water System (CWS) Nuclear Island chiller-condenser
- RWCU/SDC non-regenerative heat exchanger
- FAPCS heat exchanger
- Standby Onsite AC Power Supply Diesel Generators

The flow paths to heat exchangers and coolers are provided with flow balancing features that may be fixed orifice plates and/or control or manual valves (that can also be used for isolation). The major heat exchangers and coolers have motor-operated isolation valves for operator convenience.

The RCCWS pumps and heat exchangers are located in the Turbine Building.

The pumps in each train are powered from separate busses. During a loss of preferred power, the pumps in either train can be powered from the two non-safety related, standby diesel generators.

The RCCWS utilizes plate-type heat exchangers. Leakage through holes or cracks in the plates is not considered credible based on industry experience with plate-type heat exchangers. In addition, the heat exchangers are designed such that any gasket leakage from either RCCWS or PSWS will drain to the Equipment and Floor Drain System. This design mitigates cross-contamination of RCCWS by PSWS or PSWS by RCCWS. Pressure- and air-relief valves are provided as necessary.

Surge tanks provide a constant pump suction head and allow for thermal expansion of the RCCWS inventory. The tanks are located above the highest point in the system. Makeup to the RCCWS inventory is from the Makeup Water System (MWS) through an automatic level control valve to the surge tank. A manual valve provides a backup source of makeup from the Fire Protection System.

System Operation

The RCCWS operates during startup, normal power, hot standby, normal, and extended cooldown, shutdown/refueling and LOPP.

RCCWS heat exchanger operation is coordinated with PSWS flow. RCCWS cooling water flow through a RCCWS heat exchanger is only allowed if there is a corresponding PSWS water flow to absorb the heat load.

Plant Service Water System

The Plant Service Water System (PSWS) rejects heat from nonsafety-related components in the reactor and turbine buildings to the environment. It consists of two independent and 100% redundant trains that continuously circulate water through the RCCWS and the Turbine Component Cooling Water System (TCCWS) heat exchangers. The heat removed is rejected to either the normal power heat sink (NPHS) or to the auxiliary heat sink (AHS), for example, by mechanical draft cooling towers (site-specific). In the event of a LOPP, the PSWS supports the RCCWS in bringing the plant to cold shutdown condition in 36 hours, assuming the most limiting single active failure or a passive component failure.

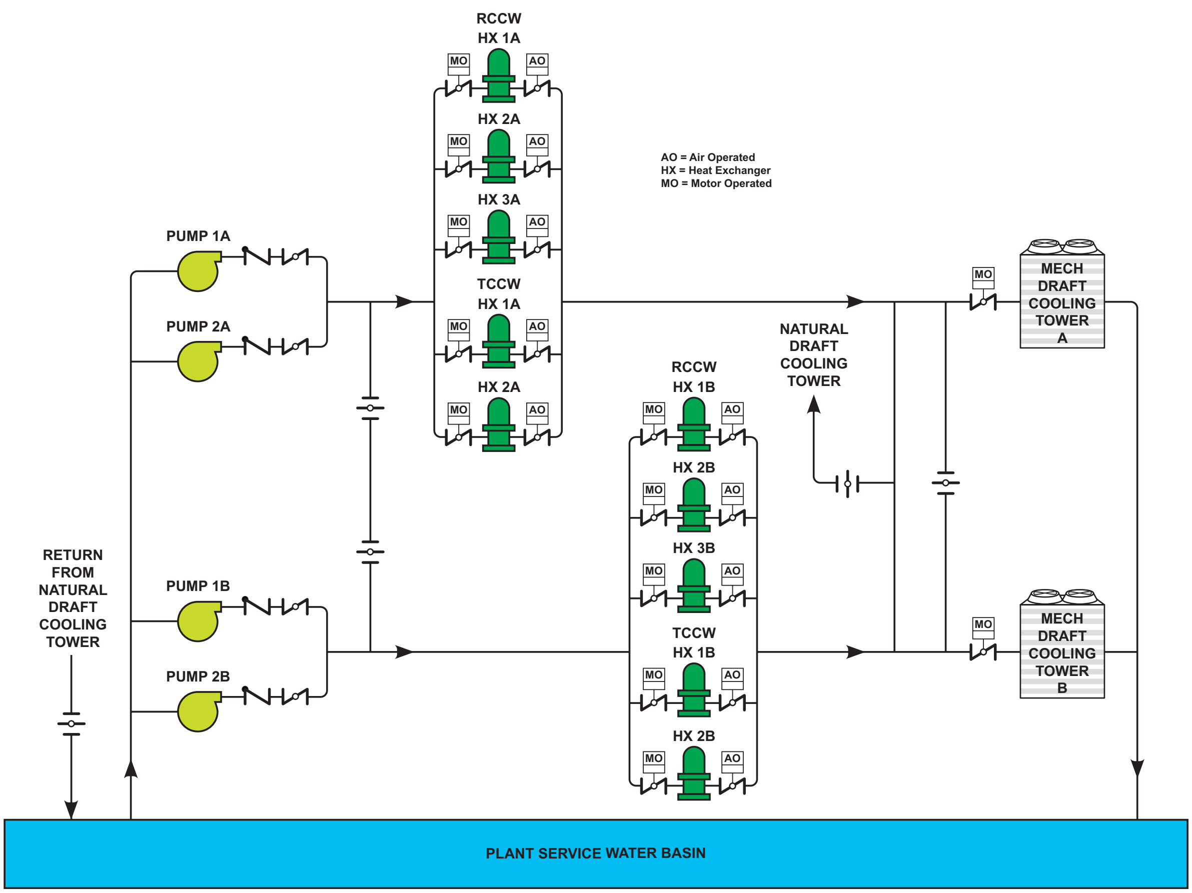


Figure 5-5 Plant Service Water System Schematic

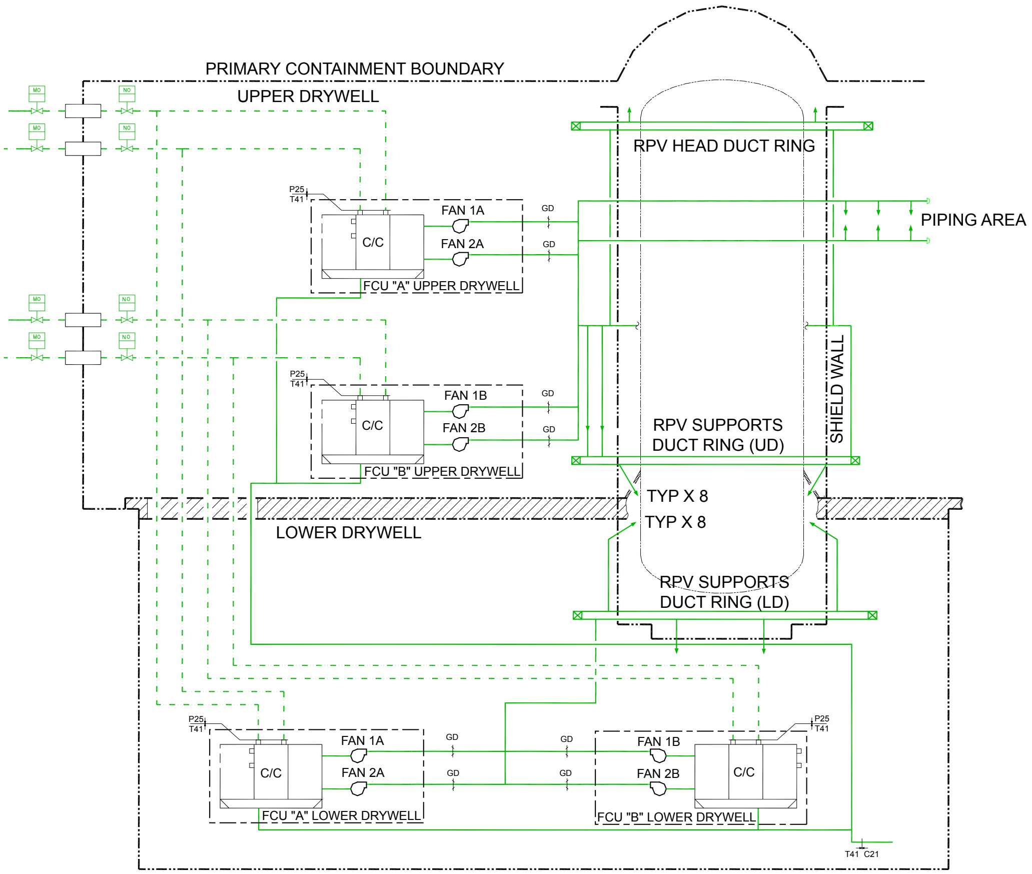


Figure 5-6 Drywell Cooling System Schematic

Each PSWS train consists of two 50% capacity vertical pumps taking suction in parallel from a plant service water (PSW) basin (Figure 5-5). Discharge is to a common header. Each common header supplies PSW to each RCCWS and TCCWS heat exchanger train arranged in parallel. The PSW is returned via a common header to the mechanical draft cooling tower in each train. Remotely-operated isolation valves and a crosstie line permit routing of the PSW to either the NPHS or AHS cooling tower. The TCCWS heat exchangers are provided with isolation valves for remote operation. Manual balancing valves are provided at each heat exchanger outlet.

The PSWS pumps are located at the plant service water basins. Each pump is sized for 50% of the train flow requirement for normal operation. The pumps are low-speed, vertical wet-pit designs with allowance for increase in system friction loss and impeller wear. Normally, the pumps in each train are powered from redundant electrical buses. During a LOPP, the pumps are powered from the two nonsafety-related standby diesel-generators.

Valves are provided with hard seats to withstand erosion caused by raw water. The valves are arranged for ease of maintenance, repair, and in-service inspection. During a LOPP, the motor-operated valves are powered from the two nonsafety-related standby diesel-generators.

The AHS provided for each PSWS train is a separate, multi-celled mechanical draft cooling tower with 50% of the cell fans supplied by one of the redundant electrical buses. During a LOPP, the fans are powered from the two nonsafety-related standby diesel-generators. The adjustable-speed, reversible motor fan units can be controlled for cold weather conditions to prevent freezing in the basin. The mechanical and electrical isolation of the cooling towers allows maintenance, including complete disassembly, during full-power operation. Makeup, for blowdown, drift, and evaporation losses to the basin is from the station water system. Anti-fouling treatment of the PSWS is provided.

System Operation

The PSWS operates during startup, normal power operation, hot standby, cooldown, shutdown/ refueling, and LOPP.

During normal power operation, the crosstie valves in the PSWS pump discharge header are open, allowing two of the four 50% capacity PSWS pumps to supply water to both PSWS trains. Heat removed from the RCCWS and TCCWS is rejected to the normal power heat sink or to the auxiliary heat sink.

Operation of any two of the four PSWS pumps is sufficient for the design heat load removal in any normal operating mode. During normal and LOPP cooldown mode, three pumps can be used for operational convenience to bring the plant to cold shutdown condition in 24 hours.

During a LOPP, the running PSWS pumps restart automatically using power supplied by the nonsafety-related standby diesel-generators.

Drywell Cooling System (DCS)

The Drywell Cooling System (DCS) is a closed-loop recirculating air/nitrogen cooling system with no outside air/nitrogen introduced into the system except during refueling. The system uses direct-drive-type Fan Cooling Units (FCUs) to deliver cooled air/nitrogen to various areas of the upper and the lower drywell. Ducts distribute the cooled, recirculated air/nitrogen through diffusers and nozzles. The drywell heat loads are transferred to the Nuclear Island subsystem of the Chilled Water System (CWS) circulating through the cooling coils of the FCUs. The DCS consists of four FCUs, two located in the upper drywell and two in the lower drywell (Figure 5-6).

Each upper drywell FCU has a cooling capacity of 50% of the upper drywell design heat load during normal plant operating conditions. Both FCUs are normally operating. Each FCU is comprised of a cooling coil and two fans downstream of the coil. Nuclear Island subsystem of CWS train A supplies one FCU, and Nuclear Island subsystem of CWS train B supplies the other. One of the fans operates while the other is on standby status. The fan on standby automatically starts upon loss of the lead fan. Cooled air/nitrogen leaving the FCUs enters a common plenum and is distributed to the various zones in the upper drywell through distribution ducts. Return ducts are also provided. The FCUs draw air/nitrogen directly from the upper drywell.

Each lower drywell FCU has a cooling capacity of 50% of the lower drywell design heat load. Each FCU is comprised of a cooling coil and two fans downstream of the coil. One of the fans operates while the other is on standby status. The fan on standby automatically starts upon loss of the lead fan. Nuclear Island subsystem of CWS train A supplies one FCU, while Nuclear Island of subsystem CWS train B supplies the other. Cooled air/nitrogen is supplied below the RPV and in the RPV support area through supply ducts. Return ducts are also provided. The FCUs draw air/nitrogen directly from the lower drywell.

Each FCU has a condensate collection pan. The condensate collected from the FCUs in the upper and the lower drywell is piped to a Leak Detection and Isolation System (LD&IS) flowmeter to measure the condensation rate contribution to unidentified leakage.

The piping for train A and train B of the Nuclear Island subsystem of CWS independently penetrate the containment. The cooling coils of one FCU in the upper drywell and one FCU in the lower drywell are piped in parallel to Nuclear Island subsystem of CWS train A and the remaining two are piped in parallel to Nuclear Island subsystem of CWS train B. The system is designed so both FCUs in the upper drywell and both FCUs in the lower drywell are always operating during normal plant operation, assuming the loss of a single electrical group or failure of any single FCU motor or fan. Upon failure of one FCU, the two fans of the remaining

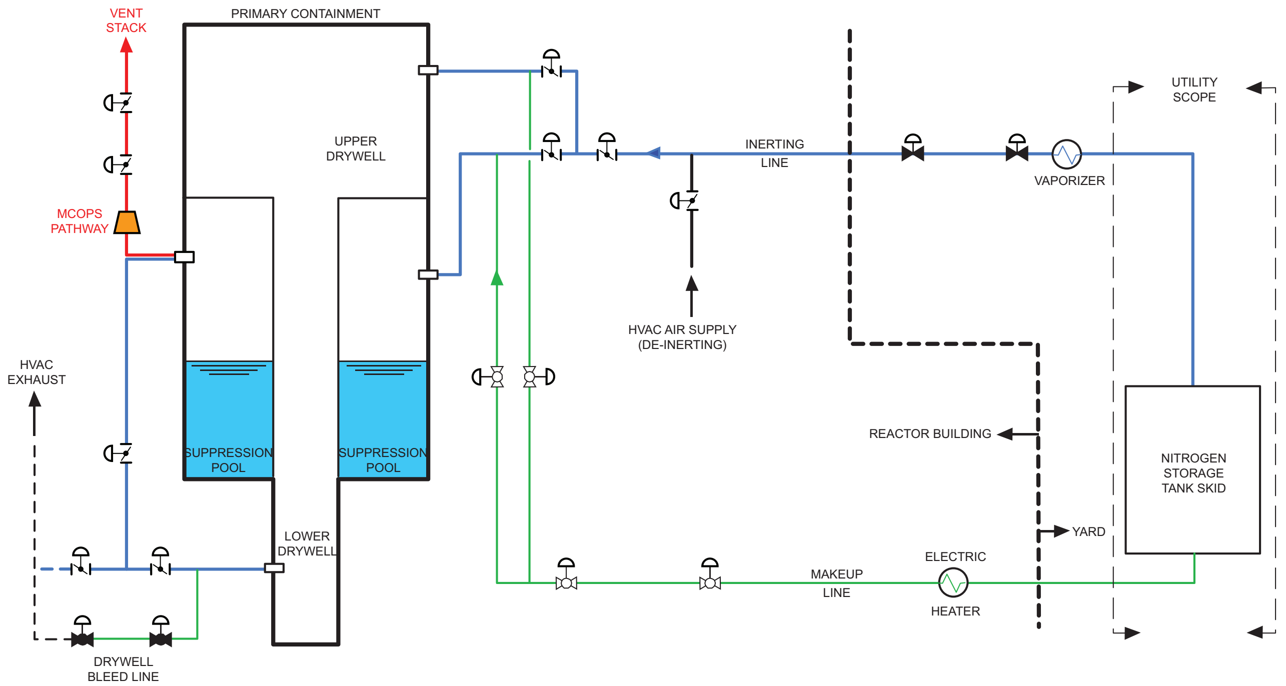


Figure 5-7. Containment Inerting System Schematic

FCU are in service. One FCU with two fans in operation maintains the drywell temperature below the maximum allowed.

System Operation

During normal plant operating condition, two FCUs in the upper drywell and two FCUs in the lower drywell are continuously operating, with one fan in service per FCU to maintain the required ambient conditions.

During plant refueling conditions, one FCU in the upper drywell and one FCU in the lower drywell continuously operate with two fans in service to maintain a habitable environment in the drywell for maintenance activities.

Nonsafety-related onsite diesel generators power the FCUs during a LOPP.

Containment Inerting System (CIS)

The Containment Inerting System (CIS) is designed to establish and maintain an inert atmosphere (nitrogen) within the primary containment volume (PCV) (Figure 5-7). An inert atmosphere is maintained in all operating modes except plant shutdown for refueling and/or maintenance. The CIS is sized to reduce containment oxygen concentrations from atmospheric to <4% by volume in less than 4 hours and < 2% in the next 8 hours in order to assure the limit of <3% during operation. After shutdown, the system also permits de-inerting of the containment for safe operator access without breathing apparatus within 12 hours.

The CIS consists of a pressurized liquid storage tank, a steam-heated main vaporizer for large nitrogen flow, electric heater for vaporizing makeup flow, two injection lines, an exhaust line, a bleed line and associated valves, controls and instrumentation. All CIS components are located inside the Reactor Building except the liquid nitrogen storage tank and the steam-heated main vaporizer that are located in the yard.

The first of the injection lines is used only for makeup. It includes an electric heater to vaporize the nitrogen and to regulate the nitrogen temperature to acceptable injection temperatures. Remotely-operated valves, together with a pressure-reducing valve, enable the operator to accomplish low rates of nitrogen injection into the drywell and suppression pool airspace.

The second injection line is used for the inerting function where larger flow rates of nitrogen are required. This line provides the flow path for vaporized nitrogen at an appropriate temperature from the steam-heated main vaporizer to be injected into the containment through remotely-operated valves and a pressure-reducing valve to injection points common with the makeup supply. The inerting and makeup lines converge to common injection points in the upper drywell and suppression pool airspace.

The CIS includes an exhaust line from the lower drywell on the opposite side of containment from the injection points. The discharge line connects to the Reactor Building HVAC system exhaust before being diverted to the Reactor Building/Fuel Building vent stack. A small bleed line bypassing the main exhaust line is also provided for manual pressure control of the containment during normal reactor operation.

System Operation

During plant startup, liquid nitrogen from the storage tanks is vaporized and injected into the wetwell and drywell regions of the containment. The nitrogen is mixed with the PCV atmosphere by the Drywell Cooling System (DCS) fans. Once inerting is complete, the CIS provides nitrogen makeup to maintain the required oxygen concentration and maintain a slightly positive pressure within the PCV to preclude air in-leakage from the reactor building.

In case of a severe accident where containment failure by overpressure is threatened, it is possible for operators to use the CIS to manually vent the wetwell (MCOPS). For more information see Chapter 11.

Chapter 6
Core and Fuel Design

Introduction and Summary

The design of the ESBWR core and fuel is based on the proper combination of many design variables and operating experience. These factors contribute to the achievement of high reliability, excellent performance, and improved fuel cycle economics. GE Hitachi (GEH) has worked together with Global Nuclear Fuel (GNF), a joint venture of GE, Toshiba and Hitachi, to achieve these objectives for the ESBWR core and fuel.

The core and fuel design methods employed for design analyses and calculations have been verified by comparison with data from operating plants, gamma scan measurements, testing facilities, and Monte Carlo neutron transport calculations. GNF continually implements advanced core and fuel design technology, such as control cell core, spectral shift operation, axially varying gadolinia and enrichment zoning, fuel cladding with improved corrosion resistance, part-length fuel rods, interactive channels, and wider water gaps in the ESBWR core. As these technological improvements are added, the core and fuel design parameters are optimized to achieve better fuel cycle economics, while improving fuel integrity and reliability, and while maintaining overall reactor safety.

The reactor lattice configuration and fuel element design for the ESBWR are basically the same as employed in previous GE-designed plants operating around the world. Key features of the ESBWR reactor core design are summarized in the following paragraphs:

- The ESBWR core mechanical design is based on conservative application of stress limits, operating experience, and experimental test results. The moderate pressure levels characteristic of a direct-cycle reactor, approximately 1,000 psia (6,900 kPa), reduce cladding temperatures and stress levels
- The low coolant saturation temperature, high heat transfer coefficients, and neutral water chemistry of the ESBWR are significant, advantageous factors in minimizing Zircaloy clad temperature and associated temperature-dependent corrosion and hydride buildup. This results in improved cladding performance at high burnup
- The basic thermal and mechanical criteria applied in the ESBWR design have been proven by irradiation of statistically significant quantities of fuel. The design heat fluxes and linear heat generation rates are similar to values proven in fuel assembly irradiation in the large fleet of operating BWRs
- In-reactor experience of fuel components acquired in the existing fleet is applicable to the ESBWR
- Because of the large negative moderator density (void) coefficient of reactivity, the ESBWR has a number of inherent advantages, including:
- (1) self-flattening of the radial power distribution, (2) spatial xenon stability, and (3) ability to override xenon in order to follow load. The inherent spatial xenon stability of the ESBWR is particularly important for large-sized plants, and permits daily load following over a large range of core power levels

- The moderate power density and the power distributions used in sizing the ESBWR core include margins providing for operational flexibility
- The ESBWR fuel assembly pitch is 0.1 inch more than the conventional BWR fuel assembly pitch, a feature it shares with the ABWR, so that it can accommodate more water in the bypass gaps between the fuel assemblies, which improves cold shutdown margin and core thermal-hydraulic stability and results in milder response for pressurization transients.

Core Configuration

The reactor core of the ESBWR is arranged as an upright cylinder containing a large number of fuel assemblies (1,132) located within the reactor vessel. The coolant flows upward through the core. The core arrangement (plan view) and the lattice configuration are shown in Figures 6-1 and 6-2, respectively. Important components of this arrangement are described in the following pages.

As can be seen from Figure 6-1, the ESBWR reactor core is comprised of fuel assemblies, control rods and nuclear instrumentation. The fuel assembly and control rod mechanical designs are basically the

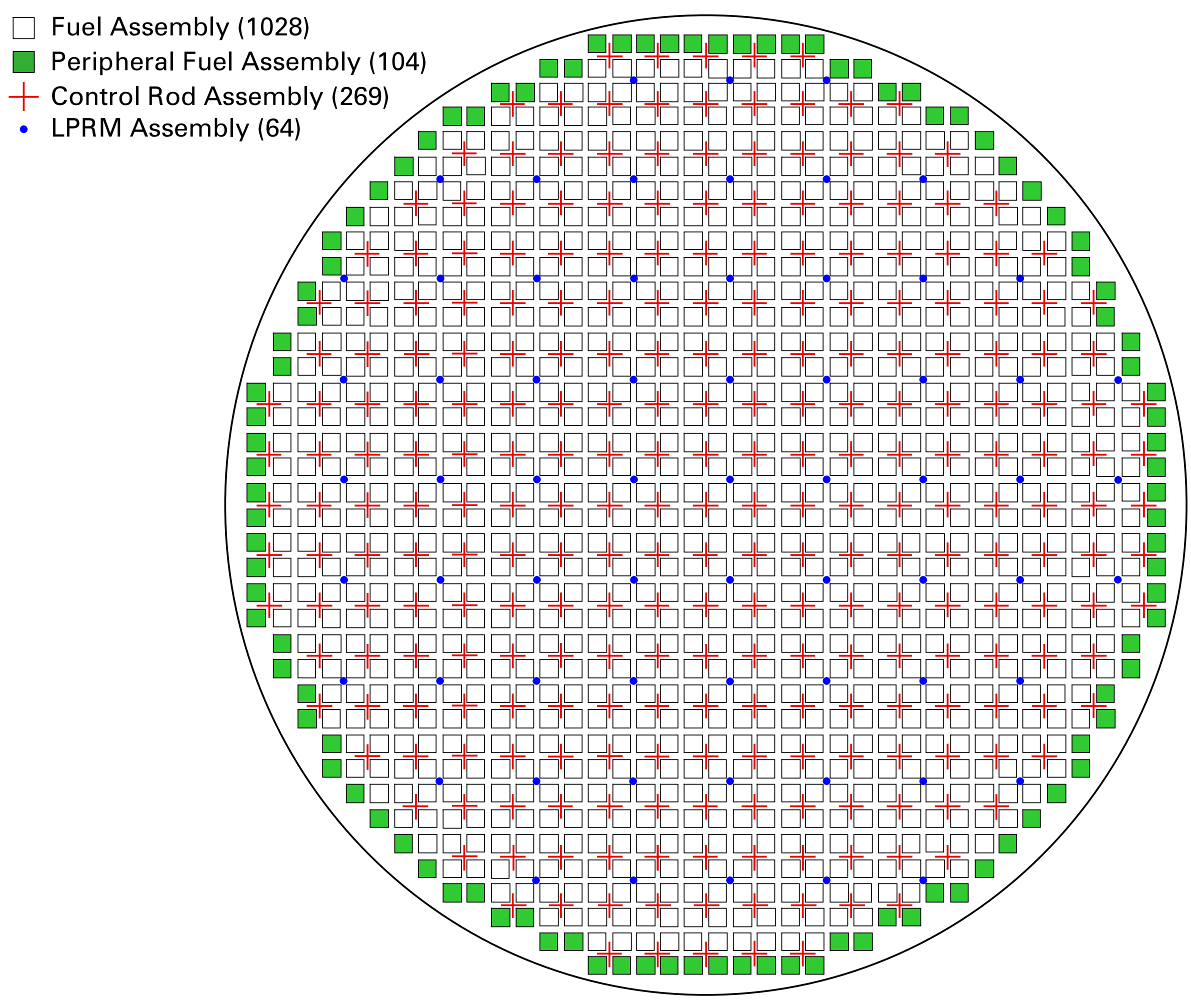


Figure 6-1. ESBWR Core Configuration

same as used in all but the earliest GE boiling water reactors; however, evolutionary improvements have been made to these components throughout the history of the GE BWR. The current generation of these components will be described below for application to the ESBWR.

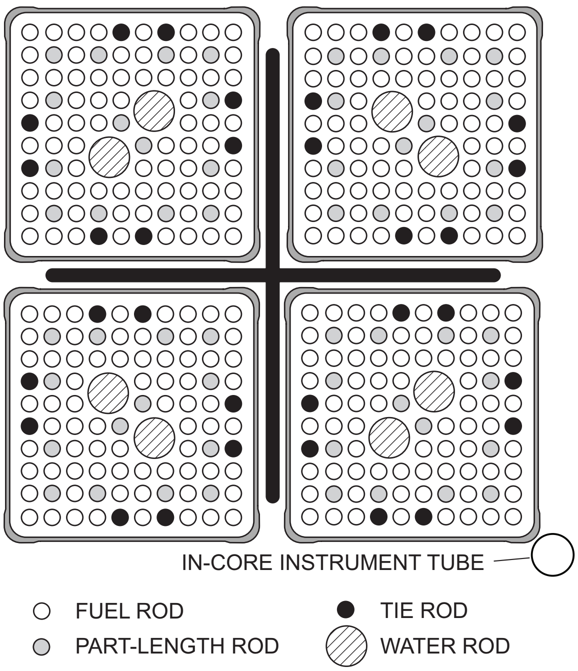


Figure 6-2. Four Bundle Fuel Module (Cell)

Fuel Assembly Description

The BWR fuel assembly consists of a fuel bundle and a channel. The fuel bundle contains the fuel rods and the hardware necessary to support and maintain the proper spacing between the fuel rods. The channel is a Zircaloy box which surrounds the fuel bundle to direct the core coolant flow through the bundle and also serves to guide the movable control rods.

The current ESBWR fuel design is based on GNF's GE14 product line. The GE14 design contains a 10x10 array of 78 full-length fuel rods, 14 part-length rods which span roughly two-thirds of the active core and two large central water rods. The design is adapted for ESBWR by shortening the active fuel length, which aids in promoting natural circulation flow. The shortened ESBWR version is denoted as GE14E.

Figure 6-3 shows the GE14 design with the major components identified. The cast stainless steel lower tie plate includes a conical section which seats into the fuel support and a grid which maintains the proper fuel rod spacing at the bottom of the bundle. The cast stainless steel upper tie plate maintains the fuel rod spacing at the top of the bundle and provides the handle that is used to lift the bundle.

The fuel bundle assembly is held together by eight tie rods located around the periphery of the fuel bundle. Each tie rod has a threaded lower end plug which screws into the lower tie plate and a threaded upper end plug which extends through a boss in the upper tie plate and is fastened with a nut. A lock tab washer is included under the tie rod nut to prevent rotation of the tie rod and nut. The part-length rods also have lower end plugs which are threaded into the lower tie plate to prevent movement of the rods during shipping or handling with the bundle oriented horizontally. The upper end plugs of the full-length fuel rods and water rods have extended shanks that protrude through bosses in the upper tie plate to accommodate the differential growth expected for high exposure operation. Expansion springs are also placed over each upper end plug shank to assure that the full-length fuel rods and water rods are properly seated in the lower tie plate.

High-performance Zircaloy ferrule spacers are located axially to maintain the proper rod spacing along the length of the fuel bundle, to prevent flow-induced vibration, and to enhance the critical power performance. These spacers are captured in the correct axial locations by pairs of tabs welded to one of the two water rods. The water rod with tabs is placed through the spacers and then rotated to capture the spacers. Once assembled, rotation of the water rod with tabs is prevented by a square lower end plug which fits into a square hole in the lower tie plate.

The fuel assembly includes a Zircaloy-2 interactive fuel channel which channels flow vertically through the fuel bundle, provides lateral stiffness to the fuel bundle, and provides a surface to support

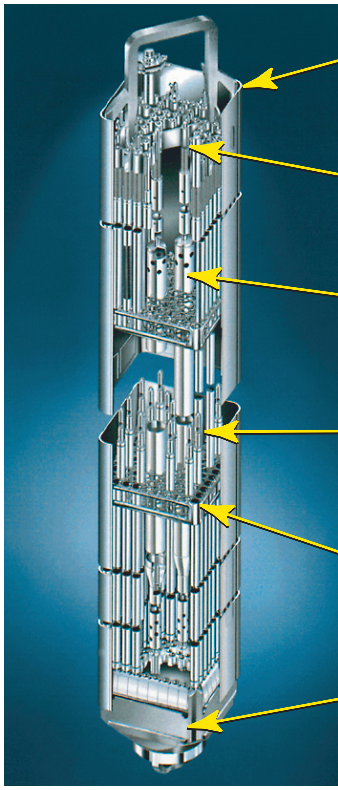


Figure 6-3. GE14 Fuel Assembly

the control rods as they are inserted. To channel the fuel bundle, the channel is lowered over the upper tie plate, spacers and lower tie plate. At the bottom end, the channel fits tightly over Inconel alloy X-750 finger springs, which seal the passage between the channel and lower tie plate to control leakage flow.

The channel and channel fastener are attached to the fuel bundle by the channel fastener cap screw which extends through a hole in the clip (or gusset) welded to a top corner of the channel and is threaded into a post on the upper tie plate. Figure 6-4 shows the channel fastener assembly.

The fuel rod design includes annealed, fully recrystallized Zircaloy-2 cladding tubing, UO_2 fuel pellets, a retainer spring assembly, and lower and upper end plugs. The fuel rods are loaded with UO_2 or $(U, Gd)O_2$ fuel pellets as required for shutdown margin control and power shaping. A plenum spring is used to apply a preload to the fuel column to prevent fuel from shifting and being damaged inside the fuel rod during shipping and handling. This plenum spring is also shown in Figure 6-4.

The lower end plug is welded to the lower end of the cladding before loading any of the internal fuel rod components mentioned above. After loading all internal components, the fuel rod is evacuated, then backfilled with helium. The upper end plug is inserted into the top end of the fuel rod, compressing the retainer spring, and welded to the cladding.

GE14 Key Fuel Design Features

The GE14 design utilizes several key design features, including part-length fuel rods, high performance spacers, low pressure drop upper tie plate, lower tie plate with debris filter, large central water rods, and interactive channels. These key design features are individually discussed below.

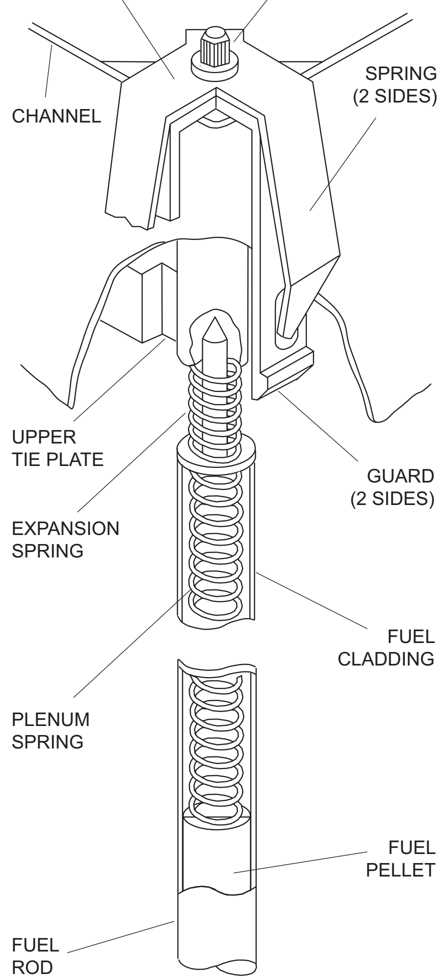


Figure 6-4. Channel Fastener Assembly

Part-Length Rods

Part-length fuel rods (PLRs) were introduced with the GE11 fuel design and have been used in all subsequent GE/GNF fuel designs. For GE14, the 14 PLRs terminate approximately at two-thirds the height of the active fuel length to provide increased flow area and reduce the two-phase pressure drop. This reduction in two-phase pressure drop leads to an improvement in core and channel stability and allows for an increase in the cladding diameter to maximize the fuel weight for a given overall pressure drop. In addition, the PLRs increase the moderator-to-fuel ratio in the top of the core to improve cold shutdown margins and fuel efficiency.

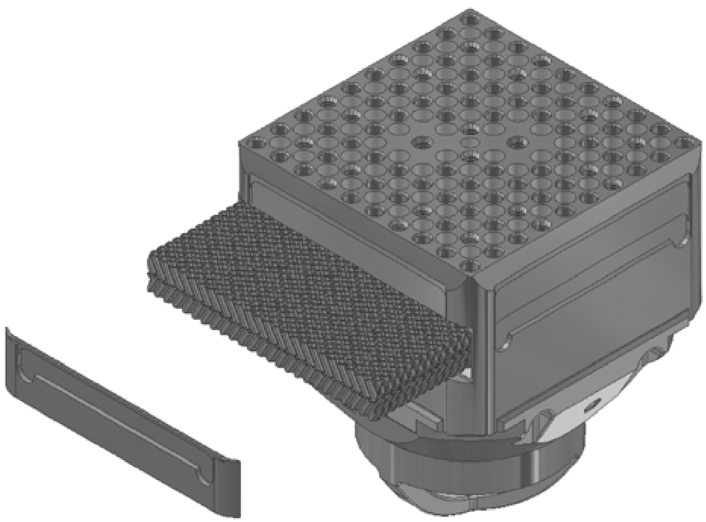


Figure 6-5. Lower Tie Plate Incorporating Defender (IM) Debris Filter

High Performance Spacers

The high-performance Zircaloy ferrule spacer provides excellent critical power performance with acceptable pressure drop characteristics. This spacer concept is also used in a number of previous products. Six spacers are used to maintain rod bow and flow-induced vibration margins for the 10x10 fuel rods of the GE14E design, while at the same time providing excellent critical power capability.

Upper Tie Plate (UTP)

The Upper Tie Plate (UTP) is designed to minimize two-phase pressure drop to improve fuel stability performance and reduce the pumping power required to drive core flow.

Lower Tie Plate (LTP) with Debris Filter

As discussed previously, the use of part-length rods and the low pressure drop upper tie plates to reduce two-phase pressure drop allows for retaining adequate single-phase pressure drop at the lower tie plate for thermal-hydraulic stability performance. In addition, it allows for the use of restricted flow paths in the lower tie plate (LTP), which serve to effectively filter debris. Figure 6-5 shows a top view of the Defender (IM) debris filter LTP, which are standard with GE14E fuel designs.

Large Central Water Rods

One of the basic characteristics of a BWR is that it is under-moderated at operating temperatures. In order to improve moderation and fuel efficiency, fuel rods are removed from the center of the fuel bundle and replaced with water rods to provide a zone for non-boiling water flow. The GE14 design includes two large central water rods to replace eight fuel rod locations and provide improved moderation.

Interactive Channels

The interactive fuel channel design has an optimized cross section, as illustrated in Figure 66, which includes thick corners where stresses are highest and thinner flat sides where stresses are low. This design minimizes the amount of Zircaloy-2 material in the channel in order to improve nuclear efficiency, increases the moderator in the bypass region for improved reactivity and hot-to-cold swing and increases the control rod clearance.

ESBWR Advanced Fuel Design

The advanced fuel design for ESBWR takes advantage of technology developments made since the introduction of GE14, including spacer manufacturing and performance enhancements to improve critical power performance and improve fuel reliability. Optimized axial fuel length, and part-length rod radial and axial positioning balanced the trade-offs between fuel assembly pressure drop, stability and nuclear efficiency in the natural circulation reactor. Features, such as total bundle uranium mass, were also considered in optimizing fuel cycle costs. The latest technology in debris filtration is applied to the advanced bundle to insure maximum fuel durability

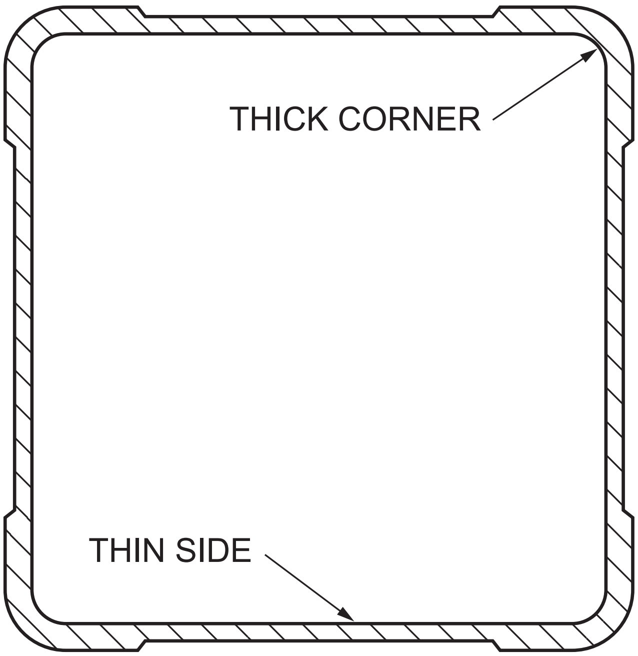


Figure 6-6. Cross Section of Interactive Channel

and reliability. Component design will continue to be shared between the ESBWR and the standard BWR fleet, thereby providing irradiation experience for components developed for the advanced fuel design.

Control Rod Description

As shown in Figures 6-1 and 6-2, cruciform control rods are configured for insertion between every four fuel assemblies comprising a module or cell. The four assemblies in a cell provide guidance for insertion and withdrawal of the control rods.

The control rods perform dual functions of power distribution shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of selected patterns of rods. The rods, which enter from the bottom of the reactor, are positioned in such a manner as to maintain the core in a critical state, and to control the radial power distribution. These groups of control elements, which are inserted during power operation, experience a somewhat higher duty cycle and neutron exposure than the other rods, which are used mainly for reactor shutdown.

The reactivity control function requires that all rods be available for either reactor "scram" (prompt shutdown) or reactivity regulation. Because of this, the control elements are mechanically designed to withstand the dynamic forces resulting from a scram. In the ESBWR they are connected to bottom-mounted drive mechanisms which provide electric motor-driven fine motion axial positioning control for reactivity regulation, as well as a hydraulically actuated rapid scram insertion function. The design of the rod-to-drive connection permits each control rod to be attached or detached from its drive during refueling without disturbing the remainder of the control functions. The bottom-mounted drives permit the entire control function to be left intact and operable for tests with the reactor vessel open.

The ESBWR has adapted the Marathon control rod blade (CRB) design, which has been applied to existing BWRs. The Marathon CRB, shown in Figure 6-7, consists of 'square' stainless steel absorber tubes, edge welded together to form the control rod wings, and welded to individual tie rod segments to form the cruciform assembly shape. The square absorber tubes are filled with a combination of boron carbide (B4C) capsules, empty capsules, and spacers. Since the active fuel height of the ESBWR design is shorter than past BWRs, the active absorber zone for the CRB is also shorter.

The control rod is dimensionally compatible with the fuel assemblies (unirradiated and irradiated). The control rod is guided, rotationally restrained and laterally supported by the adjacent fuel assemblies. The control road is designed and constructed to establish and maintain the alignment of the control rod drive line (CRDH, CRGT, and fuel assemblies) so that control rod insertion and withdrawal is predictable. The top of the active absorber of a fully withdrawn control rod is below the Bottom of the Active Fuel (BAF). Absorber gap requirements are placed on the control rod in the operating condition to be compatible with the core nuclear design requirements.

The ESBWR Marathon CRB is designed to be compatible with the Control Rod Guide Tube (CRGT) cylindrical boundary, to provide a seat

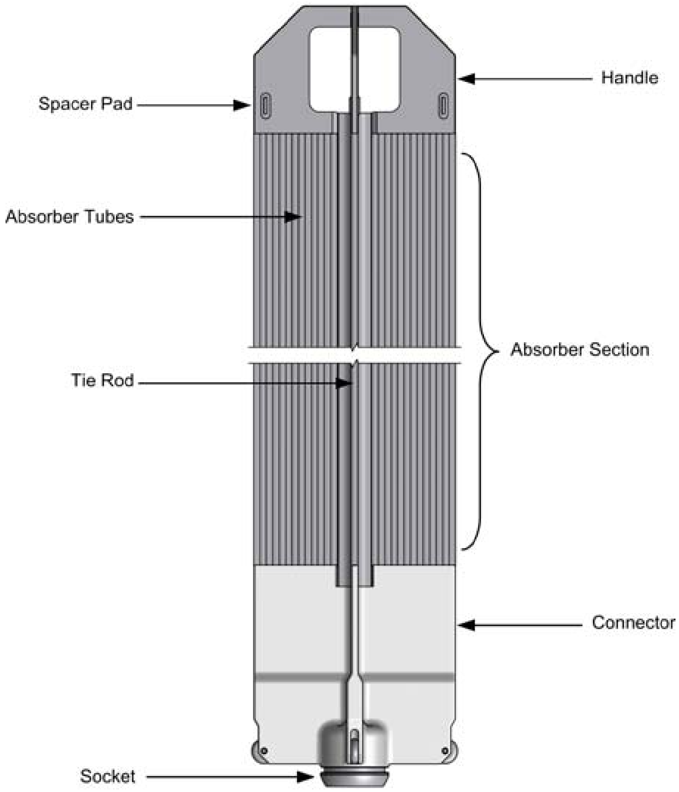


Figure 6-7. ESBWR Marathon Control Rod

with the guide tube base during FMCRD removal, to provide lower guide rollers for smooth transitions, and to have clearance with the orificed fuel support for insertion and withdrawal from the core. The control rod coupling socket provides a compatible interface with the FMCRD. The coupling engages the FMCRD by rotating one-eighth turn (45º). With the FMCRD, Control Road Drive Housing (CRDH), and CRGT positively assembled, any orientation of the cruciform control rod between the fuel assemblies is a coupled position, and rotation to an uncoupled position is not possible during reactor operation.

The structure of the ESBWR Marathon control rod has been evaluated during all normal and upset conditions, and has been found to be mechanically acceptable. The Marathon control rod is analyzed for Safe Shutdown Earthquake (SSE) events. The fatigue usage of the control rod has also been found to be well below lifetime limits. For all cases, the mechanical lifetime exceeds the nuclear lifetime.

Core Orificing

Control of the core flow distribution among the fuel assemblies is accomplished by fixed orifices. These orifices are located in the fuel support pieces and are not affected by fuel assembly removal and replacement. The core is divided into two orifice zones. The fuel assemblies, have more restrictive orifices than the inner zone peripheral, Figure 6-1. Thus, flow to the higher power fuel assemblies is increased. The orificing of all fuel assemblies increases the thermal-hydraulic stability margin of both the core and individual fuel channels.

Other Reactor Core Components

In addition to fuel assemblies and control rods, there are also in-core monitoring components and neutron sources located in the reactor core.

SRNM Assembly

There are 12 Startup Range Neutron Monitoring (SRNM) assemblies, each consisting of a fixed position in-core regenerative fission chamber sensor located slightly above the midplane of the fuel region. The sensors are contained within pressure barrier dry tubes located in the core bypass water region between fuel assemblies and distributed evenly throughout the core. The signal output exits the bottom of the dry tube under the vessel.

LPRM Assembly

There are 64 Local Power Range Monitoring (LPRM) assemblies evenly distributed throughout the reactor core. Each assembly extends vertically in the core bypass water region at every fourth intersection of the fuel assemblies and contains four fission chamber detectors evenly spaced at four axial positions adjacent to the active fuel. Detector signal cables are routed within the assembly toward

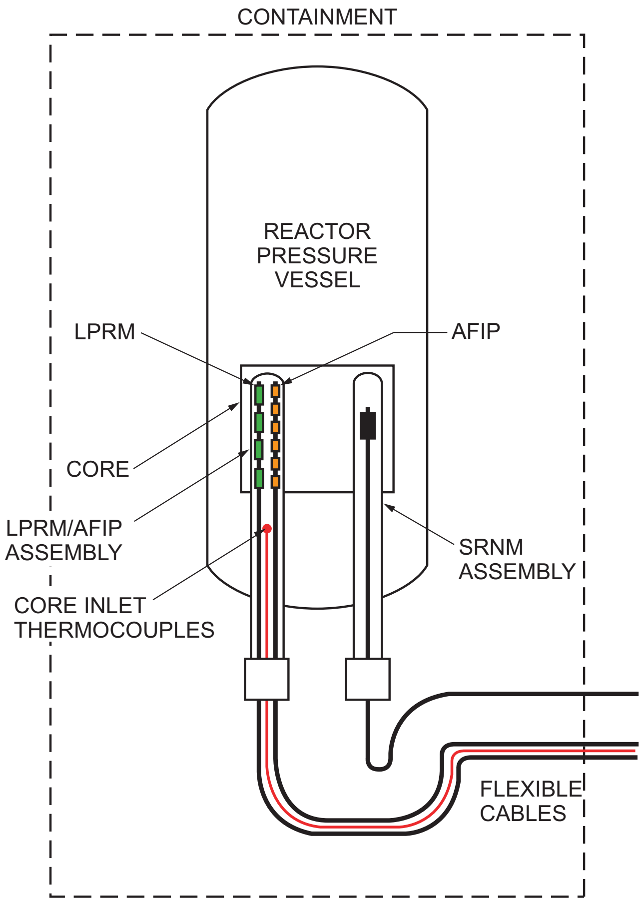


Figure 6-8. AFIP, LPRM and SRNM Schematic

the bottom of the reactor pressure vessel where the assembly penetrates the vessel pressure boundary. Below the vessel bottom, the pressure boundary is formed by an extended portion of the in-core instrument housing tube that houses the assembly.

The LPRM assembly enclosing tube also houses the Automatic Fixed In-Core Probe (AFIP) subsystem, which are gamma thermometer calibration devices. A schematic of the AFIP, LPRM, and SRNM assemblies is shown in Figure 6-8.

The LPRM assembly contains two thermocouples located just below the core plate. These thermocouples are used to calculate core inlet enthalpy, core flow by means of a heat balance.

Neutron Sources

Several californium-252 startup sources are located within the core. They are positioned vertically in the reactor by fit-up in a slot (or pin) in the upper grid and a hole in the lower core support plate (Figure 6-9). The compression of a spring at the top of the housing exerts a column-type loading on the source. Though anchored firmly in place, the sources can easily be removed, but they need not be disturbed during refueling.

The active portion of each source consists of a stainless steel sleeve enclosing two californium-252 sources. The resulting neutron emission strength is sufficient to provide indication on the source range neutron detectors for all reactivity conditions equivalent to the condition of all rods inserted prior to initial operation.

The active source material is entirely enclosed in a stainless steel cladding. The source is cooled by natural circulation of the core leakage flow in the annulus between the stainless steel sleeve and the californium sources.

Core Nuclear Design

The reactor core is designed to operate at rated power without any limitations, while delivering the total cycle length and energy desired by the utility. These design goals are achieved by designing with sufficient margin to thermal and reactivity limits to accommodate the types of uncertainties encountered in actual operation. Based on its extensive experience in BWR core design, GNF has developed a consistent set of design margins to ensure meeting these objectives without compromising overall efficiency due to the use of undue conservatism.

Core Configuration

The ESBWR core map is illustrated in Figure 6-1. There are 1,132 fuel assemblies, 269 control rods and 64 LPRM assemblies. Also the core periphery zone with more restrictive inlet flow orifices is shown.

As an option, ESBWR can employ the Control Cell Core (CCC) operating strategy in which control rod movement to offset reactivity changes during power operations is limited to a fixed group of control rods. Each of these control rods and its

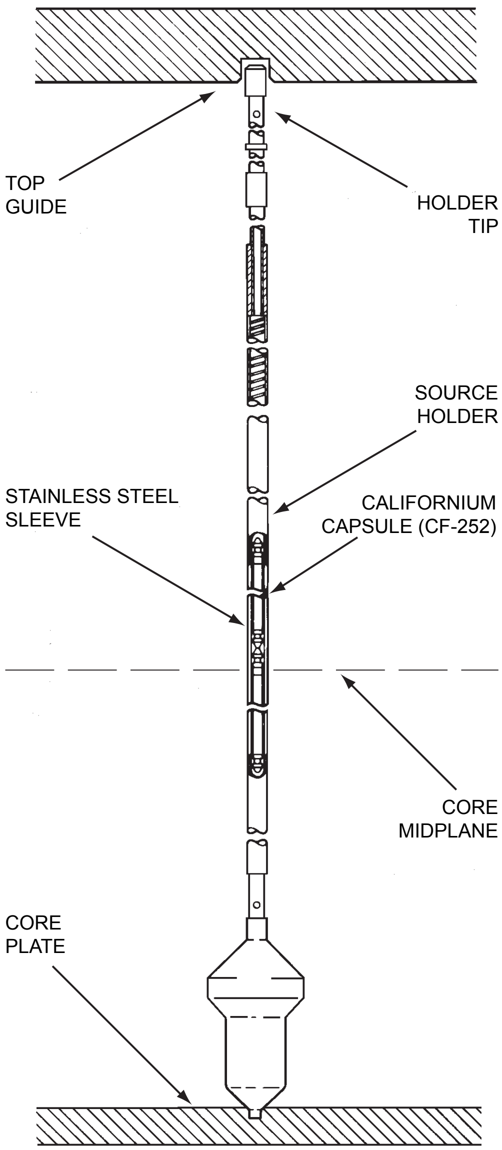


Figure 6-9. Neutron Source Schematic

four surrounding fuel assemblies comprise a control cell. All other control rods are normally withdrawn from the core while operating at power.

Low reactivity fuel assemblies are placed on the core periphery and in the control cells, to reduce neutron leakage and provide for control rod motion adjacent to low power fuel, respectively. For an initial core, the low reactivity fuel is comprised of natural uranium or low enrichment fuel. For a reload core, the low reactivity fuel is typically the high exposure fuel; fresh and low exposure fuel are scatter loaded in the remaining core fuel assembly locations.

Core Nuclear Characteristics

Reactivity Coefficients: In a boiling water reactor, two reactivity coefficients are of primary importance: the fuel Doppler coefficient and the moderator density reactivity coefficient. The moderator density reactivity coefficient may be broken into two components: that due to temperature and that due to steam voids.

- **Fuel Doppler Reactivity Coefficient:** As in all light-water-moderated and low-enrichment reactors, the fuel Doppler reactivity coefficient is negative and prompt in its effect, opposing reactor power transients. When reactor power increases, the UO2 temperature increases with minimum time delay and results in higher neutron absorption by resonance capture in the U-238
- **Moderator Density Reactivity Coefficient:** During normal plant operations, the steam void component of the moderator density reactivity coefficient is of prime importance. The steam void component is large and negative at all power levels. This steam void effect results in the following operating advantages:

  - **Xenon Override Capability:** Since the steam void reactivity effect is large compared with xenon reactivity, the ESBWR core has the capability of overriding the negative reactivity introduced by the build-up of xenon following a power decrease

  - **Xenon Stability:** The steam void reactivity is the primary factor in providing the high resistance to spatial xenon oscillations in a boiling water reactor. Xenon instability is an oscillatory phenomenon of xenon concentration throughout the reactor that is theoretically possible in any type of reactor. These spatial xenon oscillations give rise to local power oscillations which can make it difficult to maintain the reactor within its thermal operating limits. Since these oscillations can be initiated by reactor power level changes, a reactor which is susceptible to xenon oscillations may be restricted in its load-following capability. The inherent resistance of the ESBWR to xenon instability permits significant flexibility in load-following capability

- **Load Changing by Control Rod Positions and Feedwater Temperature:** The ESBWR is capable of daily load following between 100% and 50% power by adjusting feedwater temperature and control rod density within the core

Reactivity Control

Reactor shutdown control in BWRs is assured through the combined use of the control rods and burnable poison in the fuel. Only a few materials have nuclear cross sections that are suitable for burnable poisons. An ideal burnable poison must be essentially depleted in one operating cycle so that no residual poison exists to penalize the cycle length. It is also desirable that the positive reactivity from poison burnup match the almost linear decrease in fuel reactivity from fission-product buildup and U-235 depletion. A self-shielded burnable poison consisting of gadolinium(III) oxide (Gd2O3), called gadolinia, dispersed in selected fuel rods in each fuel assembly provides the desired characteristics. The gadolinia concentration is selected such that the poison is essentially depleted during the operating cycle. Gadolinia has been used in BWRs since the early 1970's, and has proven to be an effective and efficient burnable poison. In addition to its use for reactivity control, gadolinia is also used to improve axial power distributions by axial zoning of the burnable poison concentration.

The core is designed so that adequate shutdown capability is available at all times. To permit margin for credible reactivity changes, the combination of control rods and burnable poison has the capability to shut down the core with the maximum worth control rod pair fully withdrawn at any time during the fuel cycle. This capacity is experimentally demonstrated when reactivity alternations are made to the reactor core, such as during the initial core startup, and during each startup after a refueling outage.

Fuel Management

The flexibility of the ESBWR core design permits significant variation of the intervals between refueling. The first shutdown for refueling can occur anywhere from one to two years after commencement of initial power operation. Thereafter, the cycle length can be varied up to 24 months with GE14 fuel. The desired cycle length can be obtained by adjusting both the refueling batch size and the average enrichment of the reload bundles. A wide range of batch average discharge exposures can be supported depending on licensing limits and uranium supply considerations. While GNF can recommend operating margins that have been proven adequate, utility specifications on operating margins can be readily introduced into the ESBWR core.

The average bundle enrichments and batch sizes are a function of the desired cycle length. The initial ESBWR core has an average enrichment ranging from approximately 1.7 wt% U-235 to approximately 3.2 wt% U-235 for cycle lengths ranging from one to two years. For ESBWR reload cores using GE14 fuel, the average bundle enrichment is roughly 4.2 wt% U-235 with a reload batch fraction of 35% for a two-year cycle.

Neutron Monitoring System

The Neutron Monitoring System (NMS) is a system of in-core neutron detectors and out-of-core electronic monitoring equipment. The system provides indication of neutron flux, which can be correlated to thermal power level for the entire

range of flux conditions that can exist in the core. There are four subsystems in the NMS: the Startup Range Neutron Monitoring (SRNM) Subsystem, the Power Range Neutron Monitoring (PRNM) Subsystem [comprised of the Local Power Range Monitors (LPRM) and Average Power Range Monitors (APRM)], the Automated Fixed In-Core Probe (AFIP) Subsystem and the Multi-Channel Rod Block Monitoring (MRBM) Subsystem.

The NMS design has been greatly simplified for ESBWR application. Key simplification features include the SRNM, period-based trip logic and the Automated Fixed In-Core Probe (AFIP) System. The SRNMs replace the separate Source Range

Monitor (SRM) and Intermediate Range Monitor (IRM) found in conventional BWRs. Use of these fixed in-core SRNM detectors eliminates the drive mechanism and the associated control systems for the moveable SRM and IRM detectors. IRM range switches have been eliminated by incorporating a period-based trip design in the startup power range. Hence, operability is greatly improved and accidental trips due to manual range switching are eliminated. The AFIP uses fixed in-core gamma thermometers for automatic core flux mapping and calibrating the power range monitors in the ESBWR design, eliminating reactor building space for the old TIP system, enhancing operability, and reducing personnel radiation dosage.

Startup Range Neutron Monitoring (SRNM) Subsystem

The SRNM Subsystem monitors the neutron flux from the source range to approximately 100% of the rated power. The wide range (11 decades) makes the SRNMs suitable for U.S. NRC Regulatory Guide 1.97 flux monitoring. The SRNM Subsystem provides neutron flux related trip inputs (flux level and period) to the Reactor Protection System (RPS), including a non-coincident trip function for refueling operations and a coincident trip function for other modes of operation. The SRNM Subsystem has 12 channels where each channel includes one detector installed at a fixed position within the core.

Power Range Neutron Monitoring (PRNM) Subsystem

The PRNM Subsystem provides flux information for monitoring the average power level of the reactor core. It also provides information for monitoring the local power level. The PRNM Subsystem monitors local thermal neutron flux up to 125% of rated power and overlaps with part of the SRNM range.

The PRNM Subsystem consists of two subsystems:

- Local Power Range Monitoring (LPRM) Subsystem
- Average Power Range Monitoring (APRM) Subsystem

The LPRM Subsystem continuously monitors local core neutron flux. It consists of 64 detector assemblies with 4 detectors per assembly. The 256 LPRM detectors are separated and divided into four groups to provide four independent APRM signals. The APRM Subsystem averages the readings of the assigned LPRM detectors and provides measurement of reactor core power. Individual LPRM signals are also transmitted through dedicated interface units to various systems such as the RCIS and the plant process computer.

An Oscillation Power Range Monitor (OPRM) is also part of the APRM. Each OPRM receives identical LPRM signals from the corresponding APRM as inputs, and forms many OPRM cells to monitor the neutron flux behavior of all regions of the core. The LPRM signals assigned to each cell are summed and averaged to provide an OPRM signal for this cell. The OPRM trip protection algorithm detects thermal-hydraulic instability and provides trip output to the RPS if the trip setpoint is exceeded.

Automatic Fixed In-Core Probe (AFIP) Subsystem

The Automatic Fixed In-Core Probe (AFIP) subsystem is comprised of AFIP sensors and their associated cables, as well as the signal-processing electronic unit. The AFIP sensors are gamma thermometer in design (Figure 6-10). A gamma thermometer consists of a stainless steel rod that has short sections of its length thermally insulated from the reactor coolant. The insulation, normally a chamber of argon gas, allows the temperature to rise in the insulated section in response to gamma energy deposition. A two junction thermocouple measures the temperature difference between the insulated and non-insulated sections of the rod. The thermocouple reading is thus related in a straight forward way to the gamma flux. When properly adjusted for the number and spectrum of the gamma rays produced from fission and neutron capture, the fission density in the surrounding fuel can be inferred from the gamma flux and therefore, indirectly from the thermocouple reading.

The AFIP gamma thermometer sensors are installed permanently within the LPRM assemblies. In each LPRM assembly in the core, there are seven AFIP gamma thermometer sensors with four gamma thermometers installed next to each LPRM detector and three spaced between LPRMs. Consequently, there are AFIP sensors at all LPRM locations. The AFIP sensor cables are routed within the LPRM assembly and then out of the reactor pressure vessel through the LPRM assembly penetration to the vessel. The AFIP subsystem generates signals proportional to the axial power distribution at the radial core locations of the LPRM detector assemblies. The AFIP signal range is sufficiently wide to accommodate the corresponding local power range that covers from approximately 1% to 125% of reactor rated power.

During core power and LPRM calibration, the AFIP signals are collected automatically to the AFIP data processing and control unit, where the data are properly amplified and compensated by applying correct sensor calibration adjustment factors. Such data are then sent to the plant computer

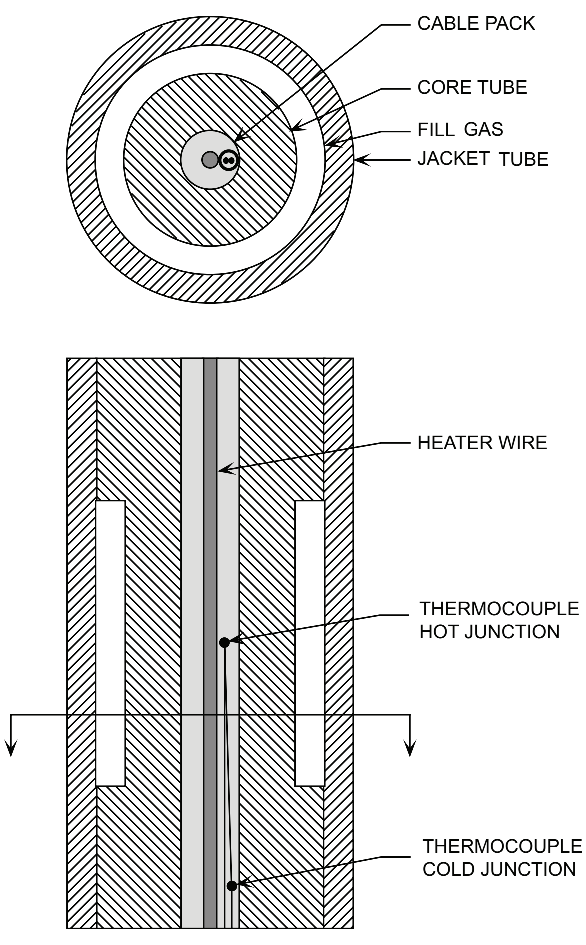


Figure 6-10. Gamma Thermometer Cross-section

function of the Nonsafety-Related Distributed Control and Information System (N-DCIS) for core local power and thermal limit calculations. The calculated local power data are then used subsequently for LPRM calibration. The AFIP data collection and processing sequences are fully automated, with manual control available. The AFIP gamma thermometer sensor has near constant, very stable detector sensitivity due to its operation principle, and its sensitivity does not depend upon fissile material depletion or radiation exposure.

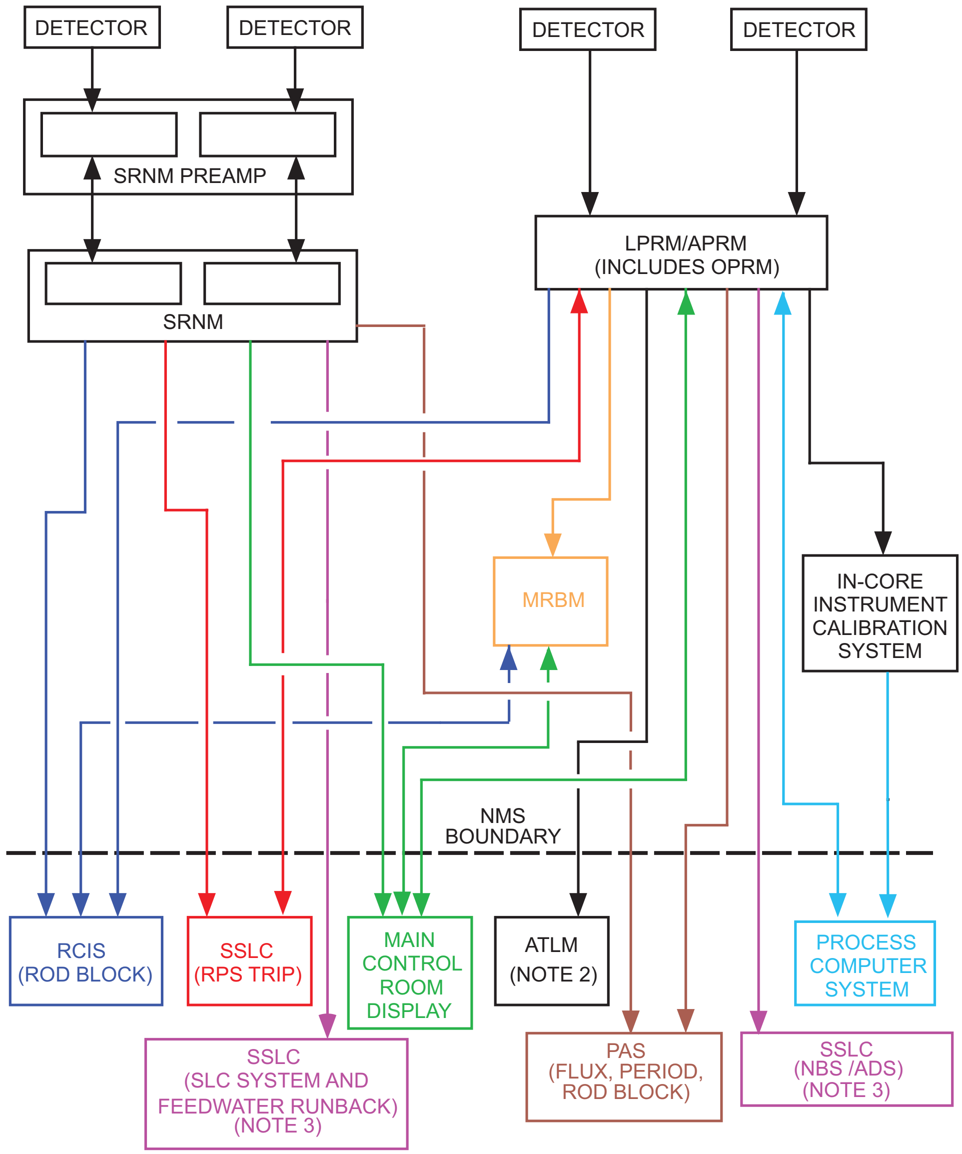
The AFIP gamma thermometer, however, can be calibrated, either manually or automatically, by using a built-in calibration device inside the gamma thermometer/LPRM assembly. The calibrated new sensitivity data of the AFIP sensors are stored in the AFIP control unit and are readily applied to the newly collected AFIP data to provide accurate local power information. The interval of the gamma thermometer calibration is to be specified in the plant technical specification.

With its stable sensitivity and rugged hardware design, the AFIP sensor has a lifetime much longer than that of the LPRM detectors. The AFIP sensors in an LPRM assembly are replaced together with the LPRM detectors when the whole LPRM assembly is replaced. The AFIP detectors within the LPRM assembly are installed such that physical separation is maintained between the LPRM detectors and the AFIP detectors. The AFIP cables are also routed separately within the LPRM assembly from the LPRM detector cables, with separate external connectors.

Multi-Channel Rod Block Monitor (MRBM) Subsystem

The MRBM Subsystem is designed to stop the withdrawal of control rods and prevent fuel damage when the rods are incorrectly being continuously withdrawn, whether due to malfunction or operator error. The MRBM averages the LPRM signals surrounding each control rod being withdrawn. It compares the averaged LPRM signal to a preset rod block setpoint and, if the averaged values exceed this setpoint, the MRBM Subsystem issues a control rod block demand to the RCIS.

Those portions of the Neutron Monitoring System that input signals to the RPS qualify as a nuclear safety system. The SRNM and the APRM Subsystems, which monitor neutron flux via in-core detectors, provide scram logic inputs to the RPS to initiate a scram in time to prevent excessive fuel clad damage as a result of overpower transients. The APRM Subsystem also generates a simulated thermal power signal. Both upscale neutron flux and upscale simulated thermal power are conditions which provide scram logic signals. A block diagram of a typical NMS division is shown in Figure 6-11.



NOTES:
1. DIAGRAM REPRESENTS ONE OF FOUR NMS DIVISIONS (MRBM IS A DUAL CHANNEL SYSTEM. THERE IS ONLY ONE IN-CORE INSTRUMENT CALIBRATION SYSTEM).
2. ATLM MONITOR IS AN RCIS FUNCTION THAT BLOCKS ROD MOTION AS THE CORE APPROACHES THERMAL LIMITS.
3. SRNM AND APRM ATWS PERMISSIVE SIGNALS TO SSLC.
4. INTERCONNECTIONS MAY BE FIBER-OPTIC OR METALLIC.

Figure 6-11. Basic Configuration of a Typical Neutron Monitoring System Division

HITACHI

Chapter 7
Instrumentation and Control

Overview

The ESBWR instrumentation and control (I&C) design (sometimes referred to as DCIS – distributed control and information system) features redundancy, diversity, fault tolerant operation, and extensive self-diagnostics while the system is in operation. This is made possible by the extensive use of advanced digital technologies.

Previous BWRs used hard-wired point-to-point connections from field equipment: Motor Operated Valves (MOVs), instrumentation, switchgear, modulating valves, etc.; to the cable spreading and control room cabinets and benchboard equipment. Essentially, there was one wire/cable per function. This traditional design scheme resulted in approximately 30,000-50,000 wires between the field to the cable spreading room and then control room. Instead, the ESBWR is designed with I&C systems that use extensive multiplexing technology that allows far fewer cables between the field and main control room and the complete elimination of the cable spreading room.

The system design comprises:

- Remote Multiplexing Units (RMUs) in the field; this equipment generally handles 200-400 signals per RMU and interfaces the I&C system with the normal field signal inputs: analog transmitters, dry contacts, thermocouples, Resistance Temperature Detectors (RTDs) and Linear Variable Differential Transformers (LVDTs); and signal outputs (typically valve position demands, switchgear and squibs)
- A distributed, networked controller layer that includes the dual and triple redundant controllers that operate the plant and generally acquire and send signals to/from the RMUs.
- A distributed, networked computer system, display, control and alarm/annunciator layer. This equipment includes all the workstations, flat panel displays, peripherals and alarms in the control room and forms the I&C interface to the operator - there is no single process computer

The instrumentation of the ESBWR is generally associated with the control of the reactor, control of the balance of plant (BOP), an extensive and intelligent alarm system, prevention of the operation of the plant under unsafe or potentially unsafe conditions, monitoring of process fluids and gases, and continuous and long-term monitoring of the performance of the plant.

Design goals of the I&C System include:

- Minimize reactor trips/system unavailability due to human errors and eliminate scrams and trips from single active component failures
- Design any systems necessary for power generation (except the electrical system) to be single-failure proof for both control and trips
- Computerized operator aids and normal/emergency procedures to reduce manual data processing and optimize the human engineered operator interface to minimize operator burden
- Provide for most I&C equipment communication and display protocols to follow internationally recognized standards
- Use standardized modular equipment and extensive self-diagnostics/fault identification to minimize operation and maintenance costs, reduce surveillance requirements and frequen-

cies, and reduce the burden on the maintenance staff

- Achieve a high degree of plant automation

Digital Measurement and Control

In general, a standardized set of microprocessor-based instrument modules is used to implement most ESBWR control and monitoring functions. However different hardware and software platforms are applied to provide diversity and defense in depth. Hence a standardized set of modules comprise the Reactor Protection System (RPS), another set comprises the Engineered Safeguard Features (ESF) and yet another set comprises the normal computer and power generation control and monitoring functions. There is a Diverse Protection System (DPS) that backs up the RPS and most ESF functions.

The standardized safety-related controllers, nonsafety-related controllers and RMUs exploit the many advantages of digital technology, including self-test, automatic calibration, user interactive Main Control Room (MCR) and Remote Shutdown Panel (RSP) operator display panels, and, where possible, use of common hardware platforms and components. These features reduce calibration, adjustment, diagnostic and repair time and reduce spare circuit card inventory requirements. The use of highly-reliable safety-related and industrial components improves DCIS system availability due to the enhanced individual component reliability, redundancy and reduced mean time to repair. Carefully chosen modern DCIS equipment also contributes to the reduction of control room monitoring and control equipment heat load and power requirements.

The various safety and nonsafety-related controller chassis and RMUs are standardized within their hardware/software platform types, only modular, plug-in, interchangeable circuit boards and modules (usually to accommodate different I/O types) differ between systems. Functional features provided in the I&C design include:

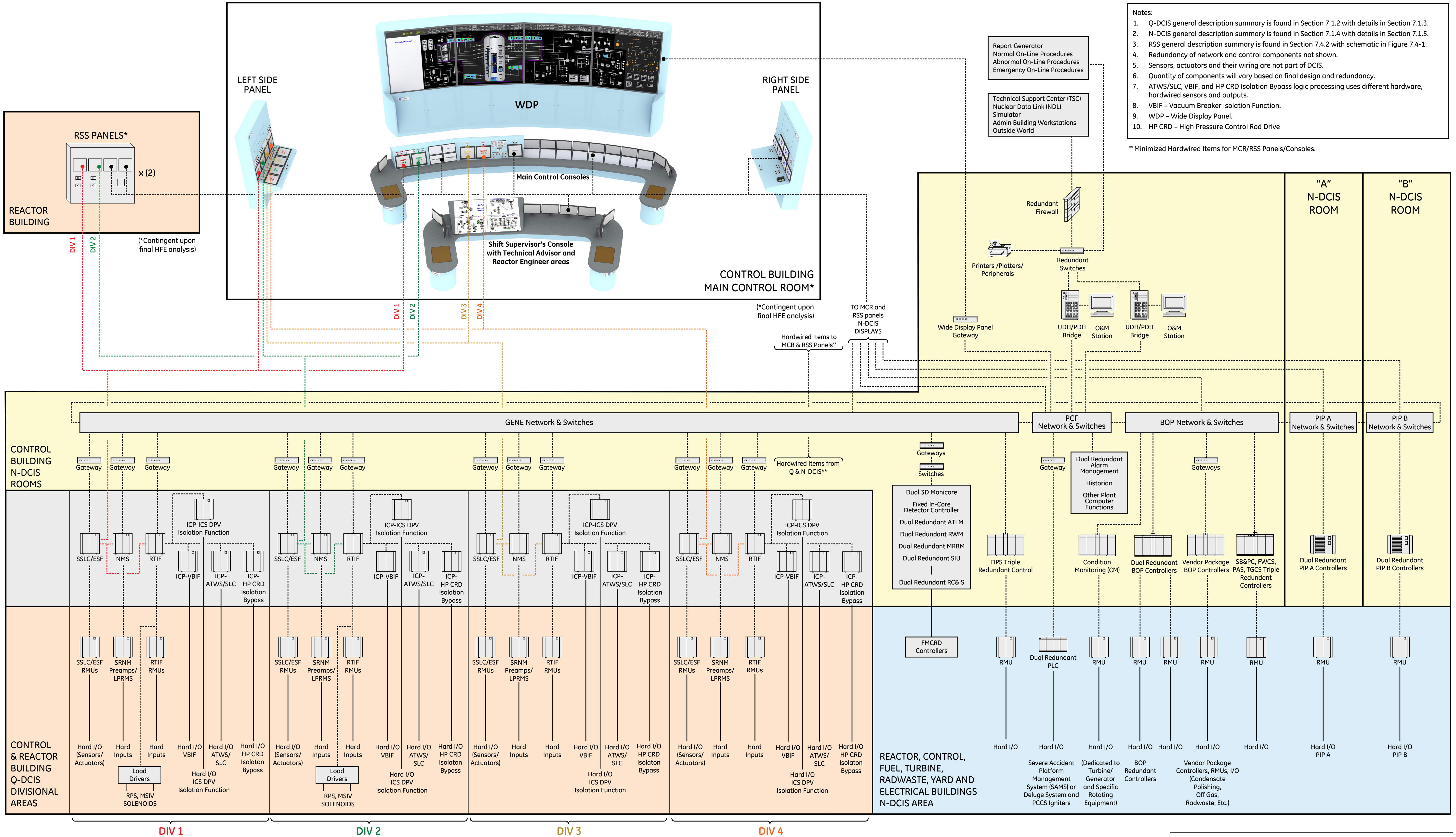
- Sensor signal processing
- Redundant sensor power supplies to meet the requirements of all sensors and controllers
- Functional microprocessors implementing data transfers, self-test functions and communications
- High-speed parallel and data busses for communication between the functional microprocessors and other modules
- Discrete and analog outputs driving external relays, actuators, logic circuits, meters and recorders
- Redundant power supplies and power feeds for both the safety and non-safety internal power supplies, controllers and RMU electronics
- Fiber-optic and other interfaces, allowing the various controllers to communicate directly with their RMUs and allowing communications between the various controllers, gateways and displays
- Menu-driven front panel displays for operator/technician interface

A simplified drawing of the ESBWR DCIS configuration appears as Figure 7-1.

Data Control Networks

The Distributed Control and Information System (DCIS) provides redundant and distributed control and instrumentation to support the monitoring and control of interfacing plant systems. The system contains safety-related (Q-DCIS) cabinets and nonsafety-related (N-DCIS) cabinets that respectively acquire and output signals to/from safety and non-safety control and monitoring systems. The system provides all electrical devices and circuitry (such as data communication and power) functions between sensors, display devices, controllers and actuators, which are defined and provided by other plant systems. The DCIS also includes the associated data acquisition and communication software, required to support its function of plant-wide data and control distribution; all data communication networks use redundant data paths and power supplies to increase reliability.

As shown on Figure 7-1, digital technology and networked fiber-optic signal transmission technology have been combined in the ESBWR design to integrate control and data acquisition for all of the plant buildings and yard. Segmentation is incorporated to provide additional redundancy so that systems or



Notes:
1. Q-DCIS general description summary is found in Section 7.1.2 with details in Section 7.1.3.
2. N-DCIS general description summary is found in Section 7.1.4 with details in Section 7.1.5.
3. RSS general description summary is found in Section 7.4.2 with schematic in Figure 7.4-1.
4. Redundancy of network and control components not shown.
5. Sensors, actuators and their wiring are not part of DCIS.
6. Quantity of components will vary based on final design and redundancy.
7. ATWS/SLC, VBIF, and HP CRD Isolation Bypass logic processing uses different hardware, hardwired sensors and outputs.
8. VBIF – Vacuum Breaker Isolation Function.
9. WDP – Wide Display Panel.
10. HP CRD – High Pressure Control Rod Drive

** Minimized Hardwired Items for MCR/RSS Panels/Consoles.

Figure 7-1. Simplified ESBWR DCIS Configuration

groups of systems can operate independently. Also as noted above and indicated on Figures 7-1 and 7-4, diversity is incorporated to mitigate common cause hardware and software failure concerns.

Signals from various plant process sensors provide input to RMUs located near those sensors. The RMUs digitize input signals and multiplex the signals via redundant fiber-optic cables to the controllers in the safety and nonsafety-related DCIS rooms. From there, the signals are sent to the various computers, controllers and display devices as needed. The process is bidirectional in that signals from the operator or plant controllers are put on the networks and multiplexed via optical fiber to the RMUs where they are directed to the various actuators (valves, switchgear, etc.

The Q-DCIS is configured into four separate divisions, there are four corresponding safety-related network (each of which is redundant), and the N-DCIS controllers incorporate either dual or triple redundancy. Whether Q-DCIS or N-DCIS, redundancy is such that a single cable, fiber, power supply or power feed can be lost without affecting the operation of any safety function or power generation DCIS function. Any RMU or controller can fail without loss of safety function, loss of power generation or affecting the operation of more than one part of a redundant mechanical system.

Each RMU is itself single-failure-proof down to a small number of signals; all single failures are self-diagnosed. The RMUs are located throughout the plant to keep plant wiring from sensors and to actuators as short as possible.

ESBWR Safety-Related DCIS Design Principles

All nuclear plant protection systems must be designed to IEEE-Std 603 standards requiring single-failure-proof design and independence between divisions; these standards have been essentially unchanged for decades. The introduction of software driven microprocessor based safety systems has not changed the requirements, but has introduced new concerns not relevant to the older analog/relay based designs. The design principles are illustrated in Figure 7-2..

Figure 7-2. Protection System Design Principles

Independence

IEEE Std 603 has always required independence and this is achieved by designing such that the four ESBWR safety-related DCIS systems are physically separated (located in different rooms/reactor building quadrants/fire areas) and electrically separated (separate power supplies, optical fiber communications, and no conductive paths between divisions or from any division to nonsafety-related components). The use of microprocessor-based systems based on two-out-of-four logic has further required that data be isolated such that software-based faults cannot adversely interfere with the safety functions of another division.

Determinacy

This is also a requirement made important by the use of microprocessor-based safety systems; relays and analog systems can generally be defined by operating times and time constants. The microprocessor-based design must require that the software must always complete all safety functions within the time required by the various transient and accident analyses. This does not require that software be written to be time invariant but instead that any randomness not extend system response past the analyses' requirements.

Redundancy

This requirement remains the same whatever the safety system design basis. IEEE Std 603 requires that the system design be such that the safety function is accomplished in the presence of any single random failure and for reactor scram systems, the design not cause inadvertent actuation given a single failure; this philosophy is usually called N-1. The ESBWR design is further improved to N-2 in that any single division may be completely out of service for any reason, any design basis event may occur with a simultaneous single failure of another division and the safety function will still be accomplished. Because of the N-2 capability, any one ESBWR safety division may be out of service indefinitely without incurring an LCO (limiting condition of operation); this feature supports lengthy surveillance tests (like battery capacity tests) without taking any associated technical specification operating penalties.

Diversity – Defense-in-Depth

This is also a requirement made important by the use of microprocessor-based safety systems that are presumed subject to common cause software- and hardware-related failures that are not assumed for relay- and analog-based designs. Although it is extremely unlikely that four asynchronous safety divisions would fail simultaneously and silently, the ESBWR is designed to assume that the scenario could happen and still provide all necessary safeguards to the public. The ESBWR addresses these concerns by configuring the DCIS into separate hardware/software platforms within each division and between the divisions and N-DCIS. Figures 7-3 and 7-4 illustrate the independence and diversity associated with the ESBWR DCIS.

Simplicity

Simplicity is a subjective criterion but, given the quality assurance, diversity, reliability and redundancy requirements, the ESBWR DCIS is designed to minimize component types, use standard software techniques and minimize DCIS supporting document types:

Figure 7-4 indicates that the safety-related hardware/software platforms performing Reactor Scram and Isolation functions (RTIF-NMS), SSLC/ESF functions (ECCS) and Independent Control Platform functions within a division are implemented on diverse platforms. Many of the safety related functions are backed up by the diverse (from all safety) Diverse Protection System (DPS) hardware/software platform. The DPS and ESBWR nonsafety-related RTNSS, plant computer and power generation functions are operated on hardware/software platforms diverse from the safety systems.

ESBWR Hardware/ Software Platforms

The ESBWR hardware/software platforms and configuration are indicated on Figure 7-4; the DCIS is configured using the following types of hardware/software platforms:

Reactor Trip and Isolation Function-Neutron Monitoring System (RTIF-NMS)

The Reactor Trip and Isolation Function (RTIF) platform supports the reactor trip (scram) and LD&IS (MSIV isolation only) systems. The equipment shuts down and isolates the reactor when predefined measured parameters move past their analytical setpoints. The Neutron Monitoring System (NMS) provides reactor flux measurements from 10-9 to 102 percent power and provides both period and absolute flux trips to the reactor trip system and rod and temperature blocks to the plant nonsafety-related control systems. The hardware/software for both systems is arranged in four divisions, supports remote acquisition of the necessary signals, remote initiation of the reactor scram and MSIV solenoids, and the trip scheme is arranged as a fail-safe design using two-out-of-four logic. Figure 7-5 provides a simplified diagram of the reactor trip and isolation function.

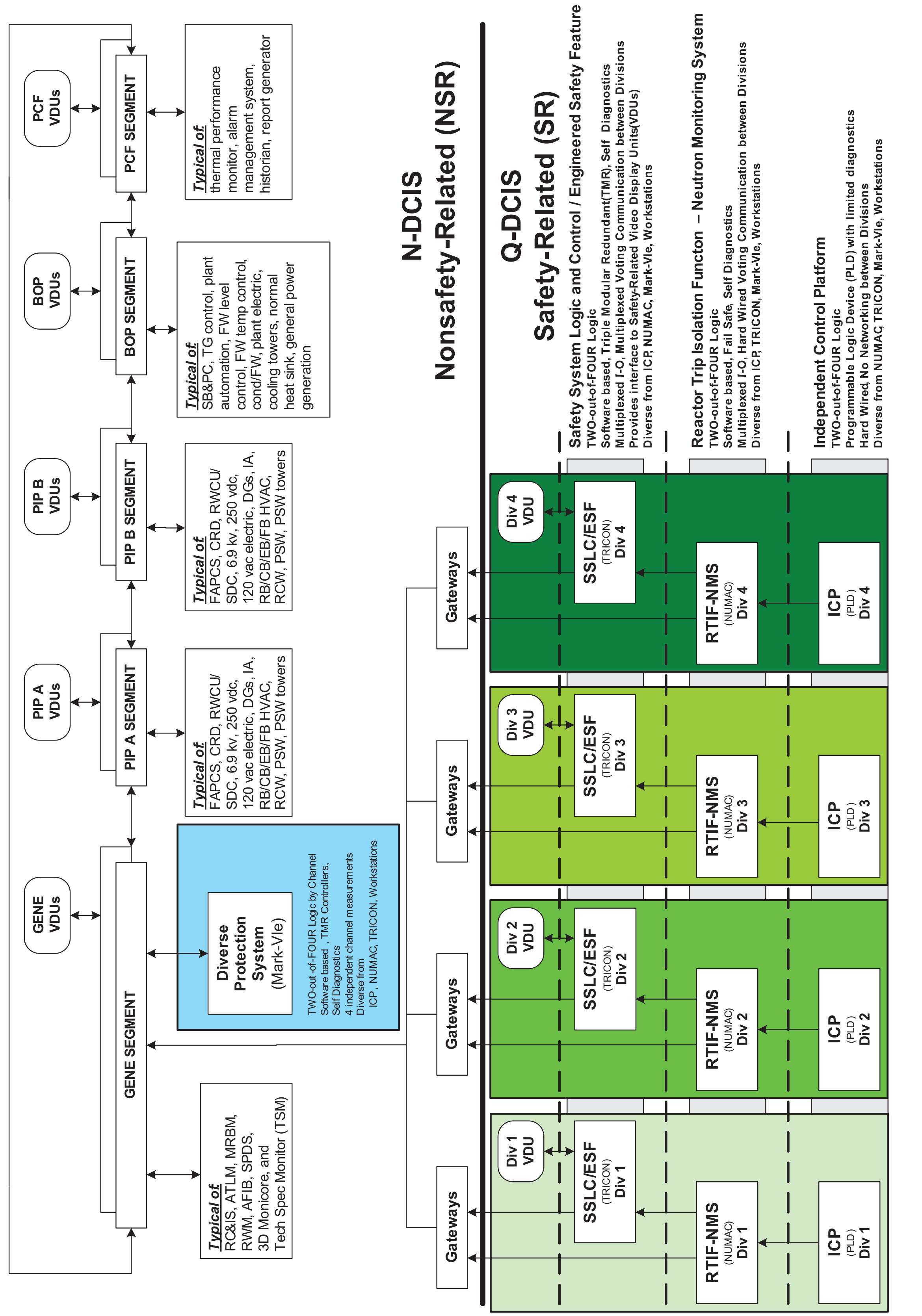


Figure 7-3. Platform, Divisional, Network, and Channel Independence

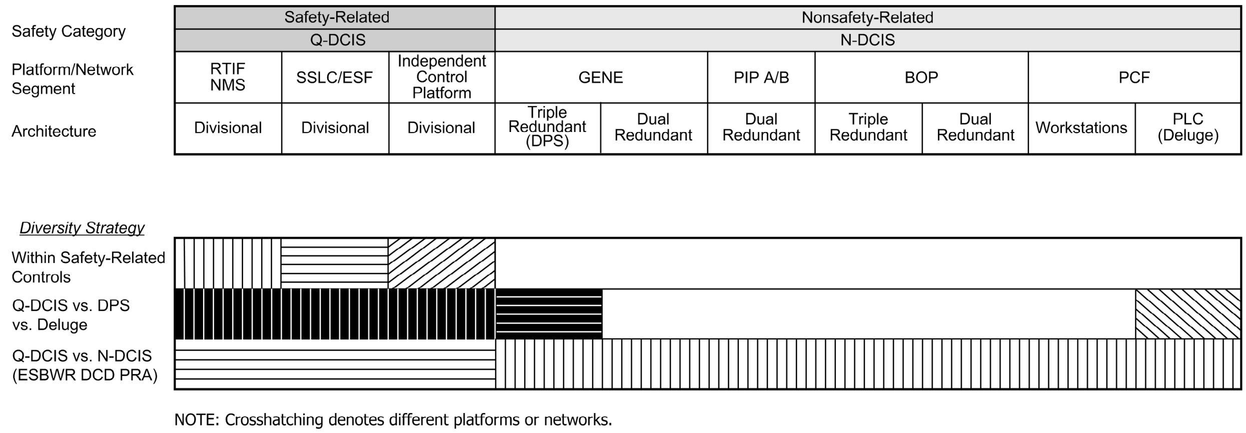


Figure 7-4. ESBWR Hardware/Software Diversity

Safety System Logic and Control/Engineered Safety Features (SSLC/ESF)

The Safety System Logic and Control/Engineered Safety Features (SSLC/ESF) initiate the ESBWR Emergency Core Cooling Systems (ECCS), Isolation Condensers, Automatic Depressurization System and Gravity Driven Cooling System. The platform is also responsible for initiating the safety-related non-MSIV isolation systems and providing safety-related monitoring of parameters and systems within each division. The hardware/software for SSLC/ESF is arranged in four divisions, supports remote acquisition of the necessary signals, remote actuation of various actuators, and the trip schemes are arranged as a non fail-safe design using two-out-of-four logic.

Independent Control Platforms (ICP)

The Independent Control Platforms (ICP) are implemented as large-scale gate arrays without operating systems and are therefore not subject to the same kinds of failures as the “programmable” RTIF-NMS and SSLC/ESF platforms. The platforms support four separate functions:

- Anticipated Transient Without Scram/Standby Liquid Control (ATWS/SLC)
- Vacuum Breaker Isolation Function (VBIF)
- High Pressure Injection Isolation Valve Bypass Function (HPCRD BYPASS)
- ICS/DPV Isolation Function (IDIF)

The independent control platforms are arranged in four divisions, are hard-wired in that no multiplexing is used and use two-out-of-four trip logic. The ICP platform does not use a cyclic real-time executive or operating system with associated controller application processor. The ICP platform is not changeable after setup and testing. The ICP are safety-related and provide backup to the highly improbable failure of the other safety-related hardware/software platforms and are used when a common cause safety concern cannot be mitigated by a nonsafety-related system like DPS; the ICP functions are required only when normal safety-related systems have failed and are therefore beyond design basis events.

Diverse Protection System (DPS)

The Diverse Protection System (DPV) is implemented as a triply-redundant hardware/ software platform running two-out-of-four trip logic to initiate backup scram and ECCS functions. The DPS will operate independently of the primary safety-related systems and does not have any sensors, actuators, hardware or software in common with RTIF-NMS or SSLC/ESF.

Reactor Trip and Isolation Function-Plant Investment Protection (RTIF/PIP)

These hardware/software platforms are dually-redundant and support the operation of systems required for long-term cold shutdown. Although they are nonsafety-related and not technically required until 72 hours after any design basis event (the ESBWR is safe indefinitely with just the safety-related systems operational) the controllers are designed for high reliability. The various RTNSS/PIP systems are mechanically redundant and the controllers and networks for each mechanical segment are themselves redundant; the redundant mechanical and control systems are organized into two segments that can operate independently from one another

Power Generation

The nonsafety-related power generation hardware/software platforms are organized into a separate segment from the RTNSS/PIP controllers and can operate independently of either RTNSS/PIP segment. These controllers are only needed for power generation and are not required for any post-accident shutdown function. The power generation controllers are either triply-redundant or dually-redundant with the former providing the reliability to make their associated transients into infrequent events. There is no single DCIS failure that will affect power generation.

Plant Computer Function

The ESBWR does not have a central plant process computer, but the various traditional computer functions (alarms, RWM, historian, SPDS, etc.) are instead distributed on singly- and dually-redundant workstations as required. The functions that involve monitoring other systems and issuing blocks as required will be implemented on hardware/software platforms different from the monitored system.

Digital Protection System Applications

Advanced Safety Systems Design

The Reactor Protection System (RPS), Neutron Monitoring System (NMS), Leak Detection and Isolation System (LD&IS) and the ECCS (ICS, GDCS and ADS) are four-channel (divisional) systems actuated by two-out-of-four logic from four-channel (divisional) sensor inputs. NMS is described in Chapter 6. Figures 7-1 and 7-3 show a more detailed interface diagram of these systems.

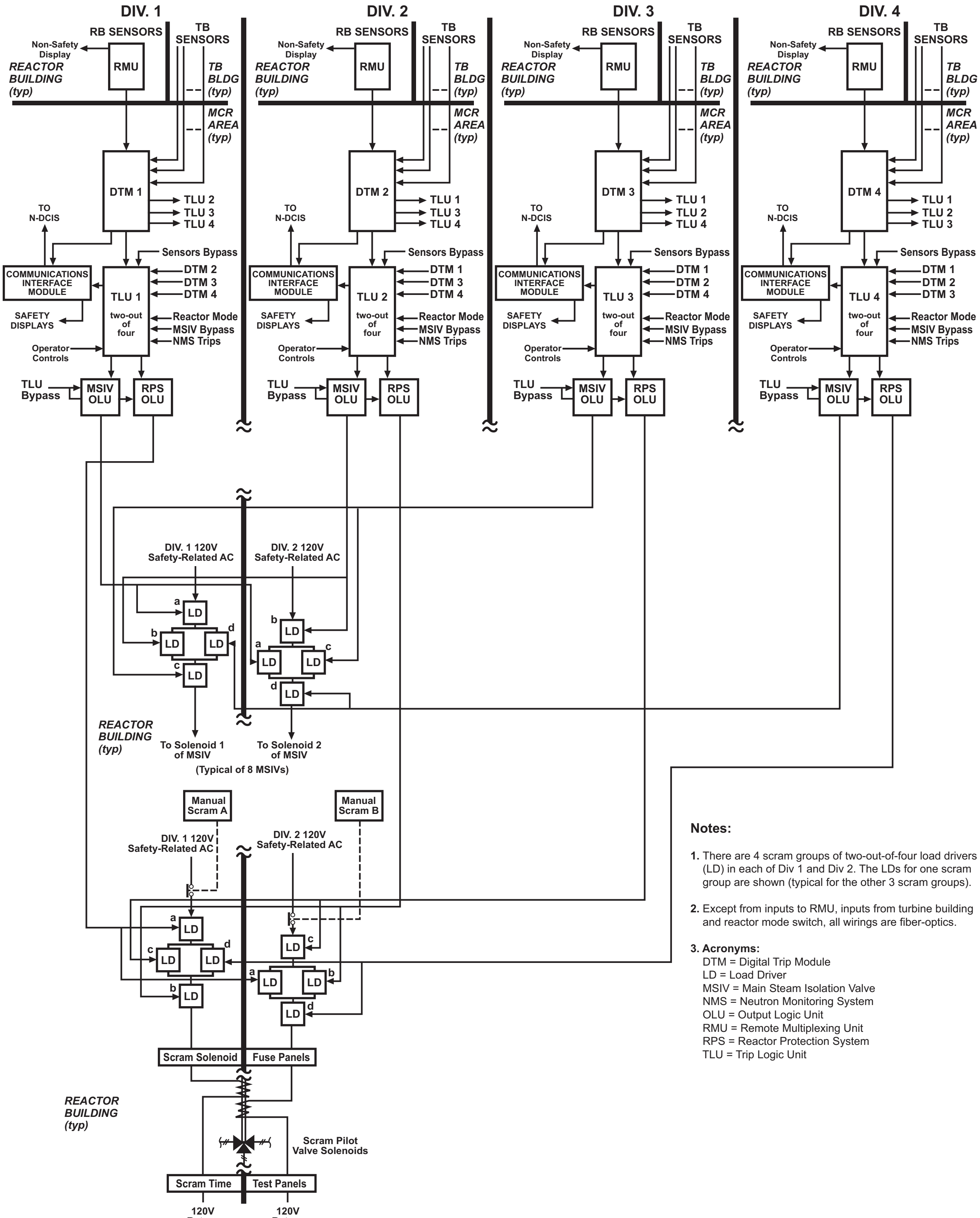
As previously described RTIF-NMS, SSLC/ESF and the ICP and their associated Q-DCIS equipment are divided into four divisions. To support independence, each division is physically and electrically separated from the other divisions. Communications between divisions is strictly limited to supporting two-out-of-four logic and is always via optical fiber; communications between divisions (Q-DCIS) and non-safety (N-DCIS) is always via optical fiber and unidirectional. There are no conductive paths between divisions or between divisions and N-DCIS; the DCIS configuration is designed to prohibit any control from nonsafety-related components to safety-related components and further designed such that divisional safety-related equipment can only be controlled from within its own division. This Q-DCIS configuration provides complete electrical isolation and prevents spreading electrical faults between safety-related system divisions and between safety- and nonsafety- related equipment. Communication between safety-related divisions and nonsafety-related equipment is through Data Gateways, although all data are isolated at the Q-DCIS source using safety-related hardware and software and therefore allow information to flow in only one direction, the nonsafety-related gateways allow the safety interface to be simple and provide protocol translation between Q-DCIS and N-DCIS.

Some control signals bypass multiplexing networks or gateways (but never isolation) when the signal design requirements are such that processing the signal through them might not satisfy the established design requirements for response time; examples are signals to the scram solenoids and signals to control rod blocking workstations.

Safety System Logic and Control/Engineered Safety Features (SSLC/ESF)

The SSLC/ESF controls the automatic actuation and operation or the following Emergency Core Cooling Systems (ECCS) during emergency operation:

- Isolation Condenser System (ICS)
- Safety Relief Valves (SRVs) and Depressurization Valves (DPVs) of the Automatic Depressurization System (ADS)
- Gravity Driven Cooling System (GDCS)



Notes:

1. There are 4 scram groups of two-out-of-four load drivers (LD) in each of Div 1 and Div 2. The LDs for one scram group are shown (typical for the other 3 scram groups).

2. Except from inputs to RMU, inputs from turbine building and reactor mode switch, all wirings are fiber-optics.

3. **Acronyms:**
   DTM = Digital Trip Module
   LD = Load Driver
   MSIV = Main Steam Isolation Valve
   NMS = Neutron Monitoring System
   OLU = Output Logic Unit
   RMU = Remote Multiplexing Unit
   RPS = Reactor Protection System
   TLU = Trip Logic Unit

Figure 7-5. Reactor Trip and Isolation Function

- The ECCS initiation of SLC (as opposed to the ATWS/SLC initiation of SLC by the ICP).

Reactor Trip and Isolation Function (RTIF)

These cabinets contain the hardware/software platforms that include the reactor protection System (RPS) and the MSIV closure function of the leak detection and isolation system (LD&IS) see Figure 7-5. The system includes the overall complex of instrument channels, trip logic, trip actuators and scram logic circuitry that initiate rapid insertion of control rods (scram) to shut down and isolate the reactor if monitored system variables exceed pre-established limits. This action avoids fuel damage, limits system pressure and thus restricts the release of radioactive material. The RPS also establishes reactor operating modes and provides status and control signals to other systems and alarms.

The RPS overrides all operator actions and process controls and is based on a fail-safe design philosophy that allows appropriate protective action by providing reliable single-failure-proof capability to automatically or manually initiate a reactor scram while maintaining protection against unnecessary scrams resulting from single failures. This is accomplished through the combination of fail-safe equipment design and a redundant two-out-of-four logic arrangement that reconfigures to a two-out-of-three logic if a channel is bypassed. The fail-safe design will initiate a channel trip based on self diagnosed critical (hardware or software) faults or loss of power. It should be noted that despite any reconfiguration actions or operator initiated bypasses, the RPS system will never degrade to a capability that does not support a reactor trip in the presence of two un-bypassed parameters exceeding their trip value.

Leak Detection and Isolation System (LD&IS)

The Leak Detection and Isolation System (LD&IS) is a four-division system consisting of temperature, pressure, flow and fission-product sensors with associated instrumentation, alarm and isolation functions. This system detects and alarms leakage and provides signals to close containment isolation valves, as required, in the following:

- Main Steam lines
- Reactor Water Cleanup/Shutdown Cooling System
- Fuel and Auxiliary Pools Cooling System
- Feedwater System
- Isolation Condenser System
- Other miscellaneous systems

Small leaks are generally detected by monitoring the air cooler condensate flow, radiation levels, equipment space temperature, and drain sump fill-up and pump-out rates. Large leaks are also detected by changes in reactor water level, drywell pressure and changes in flow rates in process lines.

Manual isolation control switches are provided to permit the operator to manually initiate (at the system level) isolation from the control room. In addition, each Main Steam Isolation Valve (MSIV) is provided with a separate manual control switch in the control room that is independent of the automatic and manual leak detection isolation logic.

Diverse Instrumentation and Control

Diversity Overview

To preclude common mode failures and to satisfy NRC requirements, the ESBWR DCIS is deliberately configured using different hardware and software platforms; the diversity schemes are within safety divisions, within systems and between Q-DCIS and N-DCIS. The NMS and RPS safety systems are diverse from the ECCS safety systems and both are diverse from the RTNSS/PIP and BOP power generation non-safety systems. Further, there is a Diverse Protection System (DPS) which, although nonsafety-related, duplicates many of the RPS scram functions and several ECCS functions, but does not use the same hardware/software of those safety systems.

The monitoring and control of the A and B RTNSS/PIP systems (like the PSWS and RWCU/SDC systems) are arranged in different network segments such that they can be operated independently (and separate from BOP power generation control and monitoring) should either of the A or B RTNSS/PIP DCIS systems fail.

Despite these different systems, the DCIS appears seamless to the operator in the main control

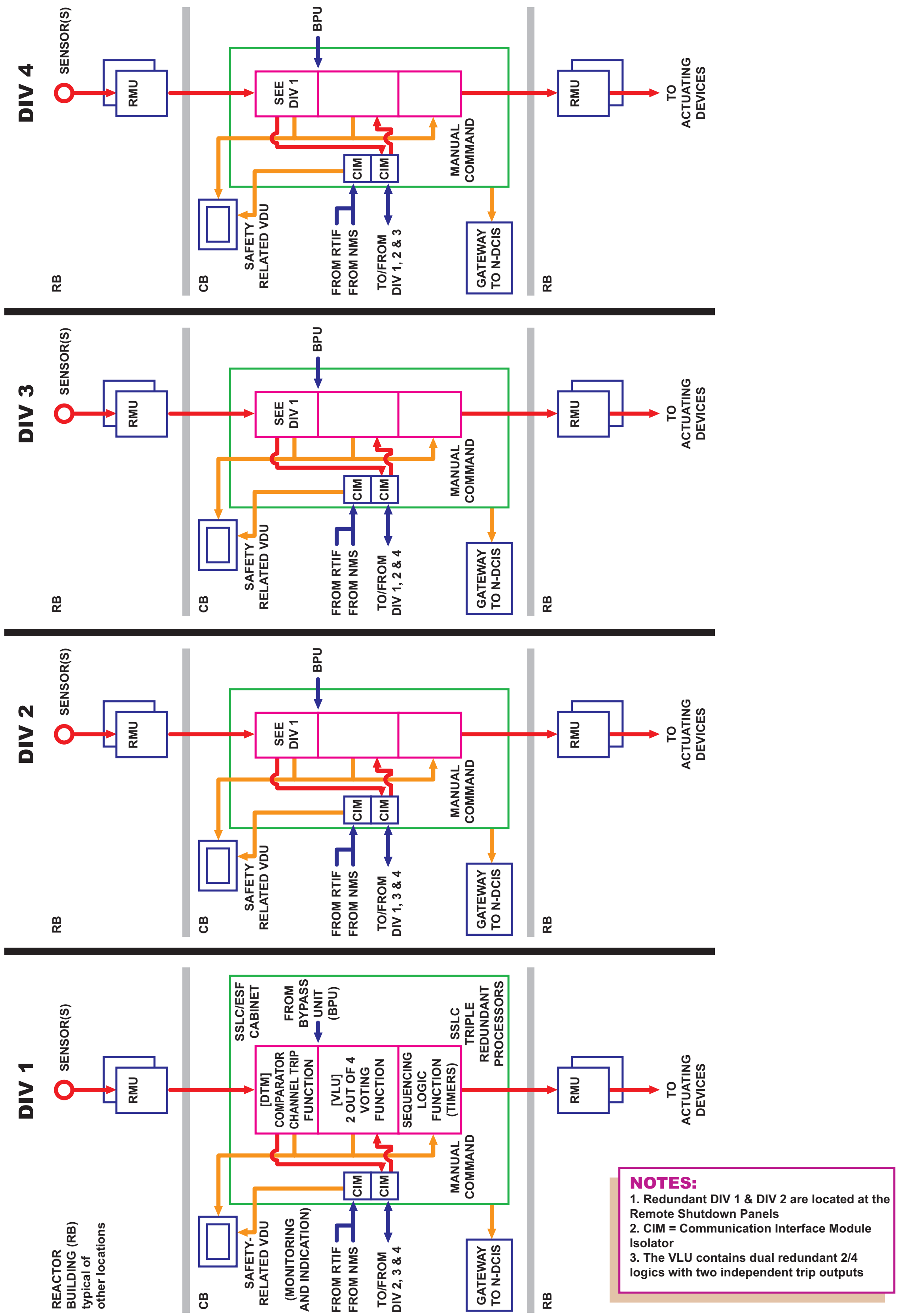


Figure 7-6. ESBWR SSLC/ESF Simplified Functional Block Diagram

room and remote shutdown panels because all of the interfaces through the VDU displays have the same operating format and menus. The nonsafety network is such that an A or B RTNSS/PIP system or BOP power generation system can be normally operated on any control room non-safety VDU However, should a DCIS network segment failure occur such that any of the DCIS is lost (more than a single failure is required), the remaining segments will remain operational. There are no common mode, single failures or control room evacuations that will prevent the operators from safely shutting down the plant using either safety or non-safety systems. There are no single failures that will result in the loss of power generation.

Diverse Scram/Shutdown System Descriptions

Backup/Manual Scram

The RTIF is a microprocessor-based system, but includes a backup scram system that is combined with the manual scram system. The latter system uses no software and uses hard-wired contactors to directly interrupt current to the scram solenoids when manual scram pushbuttons in the main control room or remote shutdown panels are activated. This action will cause a hydraulic scram whatever the status of the automatic scram logic or software.

Whenever there is an automatic scram demand in two of the four RTIF divisions or whenever the manual scram is used, software-free relay logic is used to blow down the air header that services the scram solenoids. This action will cause a scram whatever the status of the manual contactors or automatic scram load drivers.

The manual, backup and automatic scram systems will initiate a hydraulic scram by inputting high pressure water to the underside of each control rod's internal piston; this results in full rod insertion within 2 – 3 seconds.

Motor Scram

The RTIF system manual, backup and automatic scram actions will also override all RC&IS control actions to the FMCRDs and force their electric motors to drive the rods full in. Although slower than a hydraulic scram, the motors are independent of the hydraulic scram and can force a shutdown even if the various high pressure stored-energy systems and solenoids fail.

ATWS/SLC and DPS

The Anticipated Transient Without Scram/ Standby Liquid Control (ATWS/SLC) mitigation system and the Diverse Protection System (DPS) are additional diverse shutdown systems. These diverse I&C systems are part of the ESBWR defense-in-depth and diversity strategy and provide diverse backup to the RPS and the SSLC/ESF. The ATWS mitigating logic system is implemented within both the Q-DCIS and N-DCIS. The ATWS/Standby Liquid Control (SLC) mitigation logic provides a diverse means of emergency shutdown using the SLC for soluble boron injection. This logic is implemented in the safety-related ICP (Figure 7-3) and the injected boron solution is capable of shutting down the reactor whatever the status of the control rods and keeping the reactor sub-critical as it cools. The ICP safety-related logic does not use multiplexing; all necessary sensor inputs and actuator outputs are hard-wired. The ICP SLC initiation is implemented with two-out-of-four logic that reverts to two-out-of-three logic with operator bypass or self-detected failures.

On the N-DCIS side, the Alternate Rod Insertion (ARI) system hydraulically scrams the plant in a similar way as the safety-related backup scram by blowing down the air header serving the scram solenoids using the three sets of air header dump valves in the Control Rod Drive System (CRD) (the ARI dump valves are different than the backup scram dump valves). This logic is implemented in the DPS that is designed to mitigate the possibility of the safety-related digital protection system common mode failures. Figure 7-7 indicates the DPS functions.

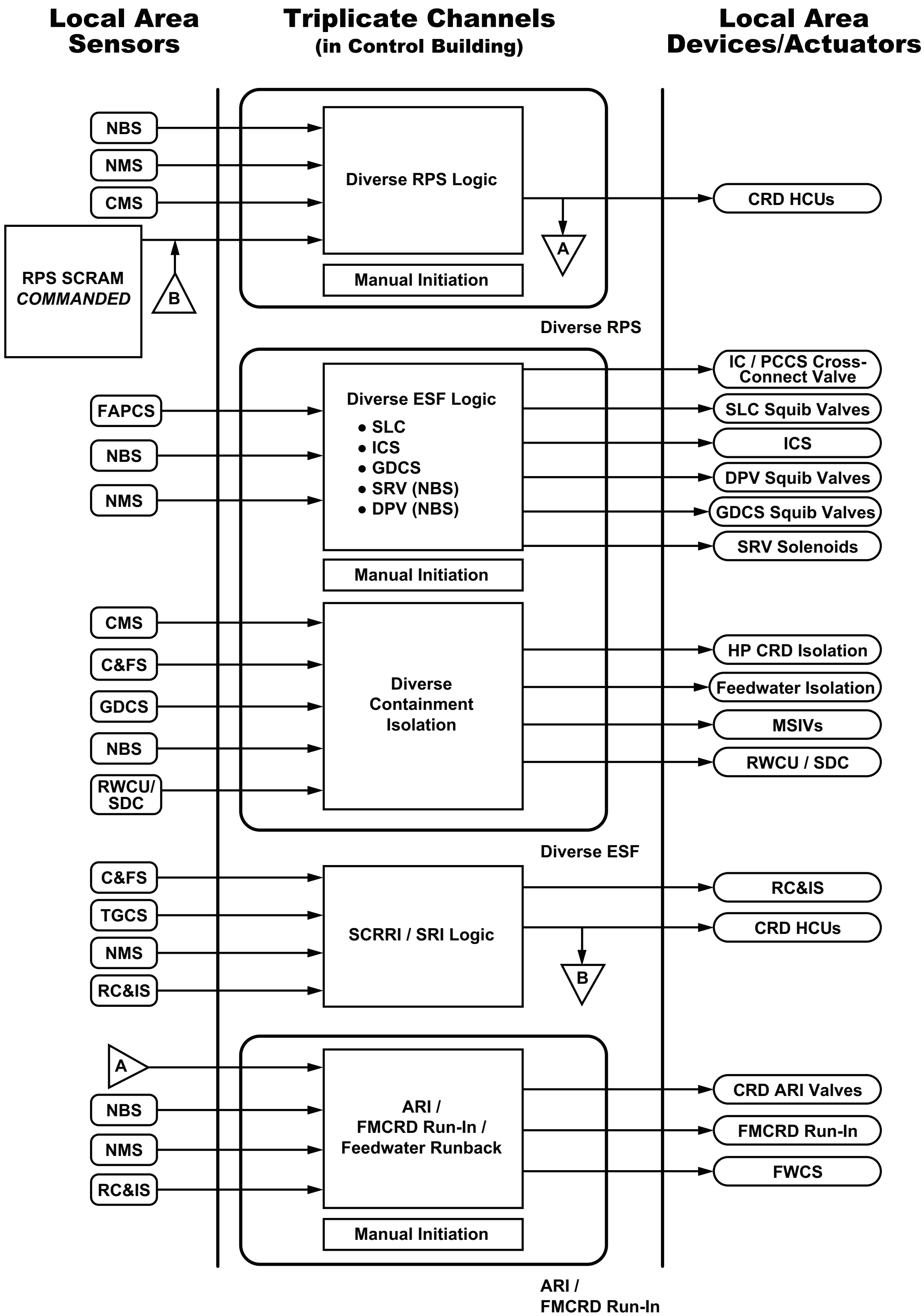


Figure 7-7. Diverse Protection System (DPS)

As part of the ATWS/SLC mitigation system, the DPS receives isolated SLC initiation signals from the four divisions of ICP and uses two-out-of-four logic to initiate the various ATWS mitigation functions. DPS has its own sensors for reactor pressure and level and can initiate the mitigation functions independently of the safety-related systems. These functions include:

- Automatic (non-safety) SLC system initiation
- Alternate Rod Insertion (ARI) as described above
- Fine Motion Control Rod Drive (FMCRD) run-in (motor scram) as described above
- Feedwater runback (FWRB) (reduces feedflow to increase core void content)
- Inhibition of the Automatic Depressurization System (ADS) and GDCS injection
- ARI and diverse scram plus delayed feedwater runback for events where the RPS scram command has been unsuccessful in shutting down the reactor, or Selected Control Rod Run-In (SCRRI)/Select Rod Insert (SRI) has been unsuccessful in reducing reactor power to an acceptable level

The ATWS/SLC control design also provides for MCR manual initiation of the SLC liquid boron injection with additional initiation of ARI and feedwater runback occurring from the same manual controls.

DPS Scram

The DPS is a nonsafety-related, triply-redundant system, powered by redundant nonsafety-related uninterruptible power sources. The highly reliable, physically isolated (it is not located in the same room as any of the safety-related logic cabinets) and independent DPS provides diverse reactor protection. Through an isolated gateway, the DPS will scram whenever two-out-of-four safety-related scram signals are generated, but the DPS includes its own sensors representing a subset of the RPS scram parameters and will independently scram if any two of its four like parameter sensors exceed their setpoints. The DPS sensors and scram actuators are independent of the safety sensors and actuators and the DPS logic operates on a triply-redundant hardware/software platform diverse from the safety-related logics. The DPS also provides the ability to initiate a manual scram from either hard-wired switches or the DPS operator displays. The DPS scram logic is two-out-of-four but reverts to two-out-of-three with operator bypass or self-detected failure.

DPS ECCS

In addition to the ATWS mitigation and backup scram functions, the DPS also provides diverse ECCS functionality core cooling by independently actuating the emergency core cooling systems. The DPS has a set of diverse ESF logics, which are implemented using separate and independent hardware and software from that of the safety-related SSLC/ESF hardware/software platforms.

The ESBWR has several ESF functions, including core cooling provided by the GDCS and SLC system, and the ADS function using SRV and DPVs. It also has the pressure relief and core cooling function provided by the Isolation Condenser System (ICS). The ESF functions of the GDCS (squib valves), SLC system (squib valves), ICS, and ADS (SRVs and DPVs) are included in the DPS to provide diverse initiation of emergency core cooling.

For the SRV opening function, three of the four SRV solenoids on each SRV are operated/powered by three of the four divisional logics/safety-related power sources. A fourth solenoid on each SRV is powered by redundant, nonsafety-related uninterruptible power with the trip logic controlled by the DPS. All ten SRVs in the ADS are controlled by the DPS through the fourth solenoid on each valve.

For the DPV opening function, one of the four squib initiators on each DPV is controlled by and connected to the nonsafety-related DPS logic. However, the other three squib initiators on each of all the DPVs are controlled simultaneously by the SSLC/ESF ADS logic. The reliability and availability of the SSLC/ESF ADS function is not affected by the existence or functionality of the DPS logic.

The triply-redundant and self-diagnostic capabilities of the of the DPS controllers precludes inadvertent actuation of the ADS because of logic hardware failures. The output load drivers of the DPS (which actuate the squib initiators and SRV solenoids) are arranged in series with the load drivers in separate cabinets and in separate fire areas to preclude inadvertent ADS initiation from single RMU/load driver failures.

The DPS will also diversely initiate the Isolation Condensers using its own sensors and two-out-of-four logic on low reactor water level and MSIV closure signals.

As with all DPS functions, provision is made for manual DPS initiation of the ESBWR ECCS systems that is separate and independent of the manual initiation from the safety-related systems.

DPS Support

The DPS provides additional support services as a diverse backup to other, important ESBWR functions, these include:

- Tripping the feedwater pumps at high reactor water level (this prevents steam line flooding in the very unlikely event that the triply-redundant FWCS did not first trip the pumps on high level).
- Opening the cross-connect lines from the equipment storage pool and the ICS/PCCS expansion pool when a low level signal is detected in the inner expansion pool. (This function assures that the IC/PCCS heat exchangers have 72 hours worth of cooling water in the unlikely event that there is a common cause failure of the safety-related hardware/software platforms.)
- Opening the ICS lower header vent valves with a predefined time delay following ICS initiation. (This function prevents the accumulation of non-condensibles in the ICS heat exchangers in the unlikely event that there is a common cause failure of the safety-related hardware/software platforms.)
- Performing selected containment isolation functions as part of the diverse ESF function. The DPS will close the MSIVs, isolate RWCU/SDC and isolate the feedwater lines based on two-out-of-four logic from its own steam line flow, RWCU/SDC differential flow, drywell pressure and feedwater line differential pressure measurements.

The DPS provides diverse monitoring and indication of critical safety functions and parameters from both its own sensors and (through isolated gateways) the safety-related sensors. The monitoring supports assessment of plant status and provides sufficient information to support manual operation of all DPS systems. The monitoring displays and display controllers use a diverse hardware and software technology from the safety-related displays and controllers.

All of the above DPS originated functions are independent of their corresponding functions implemented in the safety-related system's logic and will operate regardless of the status of the safety-related system hardware and software functionality. This includes the unlikely simultaneous common cause failure of the safety-related hardware or software. Similarly, the safety-related systems will operate regardless of the status of DPS hardware and software functionality. The DPS cabinets are physically located in different rooms and fire areas than the safety-related cabinets.

Independent Control Platform (ICP)

ICP VBIF

The vacuum breaker isolation function is provided to mitigate against the failure of the vacuum breakers to successfully reclose after design basis events. The VBIF ICP uses two-out-of-four logic and temperature sensors to detect excessive vacuum breaker (VB) leakage and prevents the loss of long-term containment integrity by closing an isolation valve on the vacuum breaker.

ICP HP CRD Isolation Bypast

The HP CRD isolation bypass function is provided to mitigate against the failure of GDCS to depressurize and inject. The Control Rod Drive Hydraulic Subsystem supplies high pressure makeup water to the reactor vessel in response to a low RPV water level condition, or in the event GDCS fails to inject following a LOCA. The normal response to a LOCA is reactor depressurization and GDCS actuation and a CRD injection system isolation. The ICP mitigates the beyond-design-basis failure of the GDCS to inject following a LOCA; for this unlikely event, CRD injection is desirable. Using two-out-of-four logic, upon detection of a LOCA and later detection of a subsequent failure of the GDCS to inject, the ICP allows additional CRD coolant inventory to be provided. The HP CRD isolation bypass logic ICP provides a "power permissive" to the redundant nonsafety-related motor-operated isolation bypass valves so that they can be opened by the same RTNSS/PIP controllers that operate the CRD pumps. Manual initiation capability of the HP CRD Isolation Bypass valves is provided in case of loss-of-instrument-air events.

ICP IDIF

The Isolation Condenser System (ICS) DPV Isolation Function (IDIF) is provided to mitigate against the unlikely common cause failure of SSLC/ESF to isolate the ICs after depressurization. This unlikely failure might allow hydrogen to accumulate in the ICs and a potential detonation. The ICP will use two-out-of-four logic to detect the vessel depressurization and isolate each IC.

Fault-Tolerant Process Control Systems

As previously described, the ESBWR control systems operating power generation systems and RTNSS/PIP systems are made up of a (redundant) network of triply-redundant and dually-redundant hardware/software platforms known collectively as Fault-Tolerant Digital Controllers (FTDCs). Single controllers may be used where the function is not important to power generation or require the reliability attendant to RTNSS/PIP systems.

In general, the key ESBWR boiler control systems such as the feedwater control system (level), feedwater control system (temperature), reactor pressure regulator and plant automation systems are based on the triplicated, microprocessor-based FTDC; the main turbine is also controlled with a triply-redundant control system to minimize DCIS failures causing either lost generation or reactor transients. Triply-redundant control systems are used to provide quantifiable reliability numbers that result in failure rates that make the resulting transients infrequent events as opposed to events expected to occur within the reactor's lifetime. The remaining important BOP control systems are based on dually-redundant FTDCs. Each FTDC includes two or three identical processing channels, each of which simultaneously receive all the redundant process sensor inputs and perform the system control calculations in parallel.

For triply-redundant process control, all three FTDC processors are active simultaneously and each receives data from and transmits signals to triply-redundant data acquisition in the RMUs. All inputs are made available to each of the three processors and voted before use; outputs are received from each of the three processors and voted before being sent to the final actuators. The voting is two-out-of-three for discrete signals and median select for analog signals. The triply-redundant controller cabinets and their RMUs receive three uninterruptible power feeds for their triply-redundant power supplies. The result is that the triply-redundant fault-tolerant controllers eliminate plant transients and trips with mean times to system failure of greater than 1,000 years.

For dually-redundant process control, one FTDC is active and the other is in hot standby; both processors receive all inputs, but only one processor at a time provides an output to the N-DCIS network and to the RMUs. The other FTDC is live and can automatically and seamlessly assume command if the normally controlling processor fails. This scheme results in single-failure-proof control.

To support the control system redundancy requirements, all important control signals are typically measured with three independent transducers; these input signals are delivered to their dually- or triply-redundant controllers by their RMUs and validated before control action is taken. Similarly, the mechanical system actuators are typically arranged redundantly to "continue" the redundancy of the controllers. The overall mechanical and control schemes are single-failure-proof.

The FTDC architecture includes:

- Two or three identical processing channels, each of which contains the hardware and application software necessary to control the system

- Dual interfaces to the redundant N-DCIS network segments
- Dual or triple multiplexing (RMU) interfaces per controller
- Interprocessor communication links between processing channels to exchange data in order to prevent divergence of outputs and to monitor processor failures
- Dually- or triply-redundant power supplies
- Signal processing techniques applied to validate the redundant input signals for use in control computations
- ·A Technician Interface Unit (TIU) for certain controllers providing a menu-driven system which allows the technician to monitor the control process, inject test signals, perform troubleshooting and calibrate process parameters (all other controllers can be accessed centrally for troubleshooting purposes

The fault-tolerant architecture of the FTDC design provides assurance that no single active component failure within the sensing, control, or communication equipment can result in loss of system function or plant power generation. The dually- and triply-redundant controller and data acquisition design also provides on-line repair capability to allow repair and/or replacement of a faulty component without disrupting any important plant process.

Feedwater Control System (FWCS) - Temperature and Level

The Feedwater Control System (FWCS) accomplishes both RPV water level control and FW temperature control. RPV water level control is accomplished by manipulating the speed of the FW pumps at high reactor powers and the low flow control valve position at low reactor powers. FW temperature control is accomplished by manipulating the heating steam flow to the two seventh-stage FW heaters or bypassing a portion of the FW flow around the high-pressure FW heaters. The two functions are performed by two sets of triply-redundant fault-tolerant digital controllers (FTDCs) located in separate cabinets. Each set of triply-redundant controllers is dedicated to perform one function.

The FWCS (level) regulates the flow of feedwater into the RPV to maintain the operator setpoint in normal operation and to maintain level between predetermined limits during transients. The desired range of water level during normal power operation is based on steam separator performance. The requirements include limiting carryover, which can affect turbine performance, radiation levels and limit carryunder; which can affect overall plant efficiency.

The FWCS (level) is a power generation (control) system that maintains proper RPV water level in the high (Level 8) to low (Level 3) operating range. During normal operation, FW flow is delivered to the RPV through three (of four) Reactor Feed Pumps (RFPs), which operate in parallel. Each RFP is driven by a variable-speed induction motor whose frequency and speed is controlled by an adjustable speed drive (ASD). In normal operation, a fourth RFP is in standby mode and starts automatically if any operating FW pump trips while at power. In abnormal operation, the fourth RFP can be set in manual mode or can be removed from service for maintenance. The reactor FW pumps receive suction from the FW booster pump discharge header. The FW booster pumps draw suction from the fourth open FW heater tank and increase FW pressure to the required suction pressure of the reactor FW pumps. There are four FW booster pumps with three in service during normal operation and the fourth in standby.

During normal power range operation the FWCS (level) operates in the three-element control mode using water level, total FW flow rate, total steam flow rate and individual feed pump suction flow rate along with pressure signals to determine the feed pump speed demand. At less than 25% of rated reactor power the FWCS uses single-element control based on RPV water level alone. A lower capacity Low Flow Control Valve (LFCV) is used to control level at low reactor powers and also operates in the single element control mode. In addition, the FWCS (level) can regulate the RWCU/SDC system Overboard Control Valve (OBCV) position to counter the effects of density changes and purge flows into the reactor during heatup when the steam flow rate is low.

The FWCS temperature controls FW temperature to allow reactor power control without moving control rods. FW temperature control allows independent control of temperature above or below the temperature normally provided by the FW heaters with turbine extraction steam. In normal operation FW temperature is not actively controlled, the temperature is a function of turbine steam flow and the turbine extraction steam pressures to the FW heaters one through six. If increased FW temperature is demanded, modulating valves admit steam from the main steam header to the seventh FW heater; if decreased FW temperature is demanded, a portion of the FW flow is directed around the high-pressure FW heaters. An increase in FW temperature decreases reactor power and a decrease in FW temperature increases reactor power. FW temperature can be set manually by the operator, or the setpoint can also be adjusted by the Plant Automation System (PAS). There are maximum and minimum allowed feedwater temperature limits that are enforced by the control system and separately by the reactor protection system. Furthermore, the FW temperature cannot be decreased when the reactor thermal power exceeds 100%. The system does not accept a temperature setpoint outside of an area allowed by the reactor power versus FW temperature map.

Steam Bypass and Pressure Control System (SB&PC)

The purpose of the SB&PC System is to control reactor pressure during startup, power generation and shutdown modes of plant operation. In normal power generation, reactor pressure is controlled by automatically positioning the turbine control valves; the SB&PC "passes through" the pressure control signal to the turbine controller that normally accepts the total demand. The turbine control system returns an actual turbine flow signal to the SB&PC such that during modes of operation where the turbine is off-line, flow limited (e.g., low stator cooling water), tripped or under control of its speed/acceleration control system during turbine roll or coastdown, the difference between the pressure control and actual turbine flow signals is used to position the bypass valves. When operating, the modulating bypass valves control reactor pressure by passing reactor steam directly to the main condenser.

This control scheme ensures that reactor steam is also automatically bypassed to the condenser whenever the reactor steaming rate exceeds the flow accepted by the main turbine, including transient events like turbine trips and generator load rejections. With the ESBWR full bypass design, the turbine bypass system has the ability to shed up to 100% of the turbine-generator-rated load without reactor trip or operation of the SRVs. For all these modes of operation, the pressure regulation system in combination with the turbine control and reactor bypass valves maintains a nearly constant reactor pressure. The Steam Bypass and Pressure Control System (SB&PC) is a triply-redundant process control system.

Turbine-Generator Control System (TGCS)

The Turbine Generator Control System (TGCS) is a triply-redundant process control system that provides both speed, acceleration and flow control and turbine protection, including turbine overspeed protection. The control system operates the turbine stop valves, control valves, and low pressure turbine stop and control valves. The turbine control system also provides automation functions like sequencing the appropriate turbine support systems and controlling turbine roll, synchronization of the main generator and initial loading.

In normal operation the turbine control valves are positioned by the SB&PC system and load "follow" the reactor steaming rate. In turbine startup and shutdown modes when the turbine control valves are positioned by the speed and acceleration control mode, whatever reactor steam flow that is not accepted by the turbine will be sent through the bypass valves to the main condenser. In all modes of operation the turbine is protected against overspeed by two triply-redundant control systems that use diverse hardware and software platforms.

The turbine generator control system, like SB&PC, can be manually switched to accept automation commands from the PAS; only the operator can switch the turbine controller to automatic, but either the operator, the turbine generator control system or PAS can switch the turbine controller to manual operation.

Non-redundant turbine-generator supervisory instrumentation is provided for operational analysis and malfunction diagnosis. Automatic control functions are programmed to protect the turbine-generator from overspeed and to trip it for adverse process or electrical conditions; the trip logic for all but bearing vibration is at least two-out-of-three logic.

Other Control Functions

The following control functions are dually-redundant. The software functions are deliberately spread through many controllers to facilitate verification and validation (V&V), quality assurance and initial construction setup.

Rod Control and Information System (RC&IS)

The RC&IS provides the means by which control rods arc positioned from the control room for power control. The RC&IS controls changes in the core reactivity, power and power shape via the FM-CRD mechanisms that move the neutron absorbing control rods within the core. For normal power generation, the control rods are moved by their electric motors in relatively fine steps; for reactor scrams, all RC&IS controls are overridden and the control rods are inserted both hydraulically and electrically. For operation in the normal gang movement mode, one gang of control rods can be manipulated at a time but the RC&IS will halt gang motion if the rods within the gang are not at the same position. The system includes the logic that restricts control rod movement (rod block) under certain conditions as a backup to procedural controls.

The RC&IS is monitored and, if necessary, blocked by the ATLM, RWM, MRBM and NMS systems. The safety-related NMS issues rod blocks (through appropriate isolation) whenever reactor power is too high or when the reactor is operating in disallowed areas of the feedwater temperature/reactor power map.

The RC&IS and the scram timing panels also support automatic measurement of control rod scram speeds for either planned or unplanned scrams for later comparison to Technical Specification Scram time requirements.

All of the rod blocks, sequence controls and thermal limit monitoring remain operative whether or not the plant is being controlled by PAS; they apply to all modes of plant operation.

The Rod Control and Information System (RC&IS) is a dual redundant process control system that has three modes of operation:

- In Manual, the operator can select and position the control rods manually, either one at a time or in a gang mode. If the selected rods deviate from the predefined rod sequence an alarm will sound
- In Semi-Automatic Mode, the operator needs to only give permission to start and stop control rod motion and the RC&IS will insert or withdraw the control rods following a predefined control rod sequence
- In the Automatic Mode, the RC&IS responds to rod insertion or withdrawal demands from the PAS; this mode will also follow a predefined control rod sequence

In all modes of operation, the safety- and non-safety-related NMS, ATLM, RWM and MRBM systems will supervise rod motion and are capable of issuing rod blocks. In all modes of operation the two RC&IS controllers must agree to allow rod motion; it is possible to bypass one RC&IS controller, but this will force the plant out of PAS control.

Automatic Thermal Limit Monitor (ATLM)

The RC&IS is monitored by the Automatic Thermal Limit Monitor (ATLM), which provides an on-line measurement of plant thermal limits from the LPRMs and periodic 3D Monicore updates. The ATLM will automatically block rod motion if it detects operation near Technical Specification thermal limits. The ATLM will also enforce the soft duty guidelines to mitigate the potential for fuel leakers. Although not a direct function of RC&IS monitoring, the ATLM will also monitor and mitigate loss of feedwater heating transients.

Rod Worth Minimizer (RWM)

The RC&IS is also monitored by the Rod Worth Minimizer (RWM), which forces compliance to the defined control rod sequencing rules by independently issuing rod blocks should a high worth rod pattern develop. This scheme and the control rod separation switches essentially eliminate the possibility of a control rod drop accident.

Process Radiation Monitoring System (PRMS)

The Process Radiation Monitoring System (PRMS) monitors and indirectly controls radioactivity in process and effluent streams and activates appropriate alarms, isolations and controls through its RTIF, SSLC/ESF and N-DCIS interfaces. The PRMS indicates and records radiation levels associated with selected plant liquid and gaseous process streams and effluent paths leading to the environment. All effluents from the plant, which are potentially radioactive, are monitored both locally and in the control room. Where the measurements are safety-related and a trip initiator, the PRMS is configured in four divisions. The measurements include the following:

- Main steam line tunnel area
- Reactor and Fuel Building ventilation and exhaust (including fuel handling area) and stack
- Control Building air intake supply
- Technical Support Center air intake supply
- Containment drywell sumps liquid discharge
- Radwaste liquid discharge
- Offgas discharge (pretreated and post-treated)
- Gland steam condenser offgas discharge
- Turbine Building vent exhaust and stack
- Radwaste Building ventilation exhaust and stack

Area Radiation Monitoring System (ARMS)

The Area Radiation Monitoring System (ARMS) provides operating personnel with a record and indication, in the main control room, of gamma radiation levels at selected locations within the various plant buildings and gives warning of excessive gamma radiation levels in areas.

The ARMS consists of gamma-sensitive detectors, digital radiation monitors, auxiliary units and local audible warning devices. System recording, like all process functions, is done by the N-DCIS historian plant computer function. The detector signals are digitized and multiplexed for transmission to the radiation monitors and to the main control room. Each local monitor has two adjustable trip circuits for alarm initiation. Auxiliary units are provided in local areas for radiation indication and for initiating the sonic alarms on abnormal levels. Radiation detectors are located in various areas of the plant to provide early detection and warning for personnel protection.

Containment Monitoring System (CMS)

The Containment Monitoring System (CMS) measures, alarms and records radiation levels and the hydrogen and oxygen concentrations in the containment in addition to various temperatures, pressures and level parameters under normal and post-accident conditions. It is automatically put in service upon detection of LOCA conditions.

The CMS provides normal, plant shutdown and post-accident monitoring for gross gamma radiation and hydrogen/oxygen concentration levels in both the drywell and suppression chamber. The CMS consists of redundant divisions designed so that failure of any single element will not result in the loss of all parameter information. Electrical separation is maintained between the redundant divisions. All components used for safety-related functions are qualified for the environment in which they are located. The system can be actuated manually by the operator, or automatically initiated by a LOCA signal.

The CMS also provides safety- and nonsafety-related suppression pool temperature monitoring (SPTM). The safety-related temperature measurement function is to prevent the suppression pool temperature from exceeding established limits by providing the four divisional signals for an automatic scram initiation that limits heat addition to the pool. The nonsafety-related function is to provide the pool temperature signals to DPS for its backup scram function.

Plant Computer Function (PCF)

On-line networked computers and workstations are provided to monitor and log process variables and make certain analytical computations. The process computer function is distributed in various workstations and cabinets on a different (redundant) network segment from the RTNSS/PIP and power generation BOP controllers. The important computer functions are redundant and there is no longer a centralized process computer.

The process computer functions include:

- Most nonsafety-related display support
- Reactor core three-dimensional power monitoring (3D Monicore)
- Balance-of-plant (BOP) performance calculations
- Sequence of Events (SOE) monitoring
- Manual and automatic logging
- SPDS (Safety Parameter Display System)
- Historian (normal slow speed recording)
- TRA (Transient Recording & Analysis – high speed recording of analog signals)
- Alarm Management (alarms and annunciation)

Firewalls to protect the DCIS, support the offsite simulator and emergency response functions

Remote Shutdown System (RSS)

In the event that the control room becomes inaccessible, the reactor can be brought from power range operation to cold shutdown conditions by use of controls and equipment that are available outside the control room. The reactor can be manually scrammed, and MSIVs can be closed from the Remote Shutdown System (RSS) using hard-wired switches and no software, but all of the remaining RSS control is accomplished with VDUs. There are two remote shutdown panels that each contain a safety-related division 1 and division 2 VDU and two nonsafety-related VDUs connected to the RTNSS/PIP A and B network segments.

These VDUs and their associated safety and nonsafety-related control systems remain operational with a complete loss of all control room equipment since neither the safety nor non-safety networks or controllers are physically in the main control room. The loss (including from fires) of the fiber and Ethernet cables connecting the control room VDUs to safety and non safety-related controllers in their DCIS rooms does not adversely affect the remaining DCIS equipment nor cause inadvertent operation of safety-related or nonsafety-related RTNSS/PIP equipment.

Control room fires or smoke will not adversely affect the safety-related or nonsafety-related controllers/network equipment or their power feeds.

As a result, all safety-related control and monitoring (division 1 and 2 only) and all nonsafety-related control and monitoring remain available to the remote shutdown panel operator using the same operator interface that was used to operate the same equipment from the main control room

Since the Q-DCIS and safety-related equipment is battery operated, the safety-related equipment is available for at least 72 hours. The nonsafety-related systems availability (i.e., control, not monitoring) is dependent on the status of offsite power and the standby and ancillary diesel generators.

The two remote shutdown system panels are located in enclosed rooms in separate division 1 and 2 quadrants of the reactor building and are normally locked; any access to these rooms is alarmed in the main control room.

Main Control Room (MCR)

The key elements of the ESBWR Main Control Room (MCR) design are:

- The compact Main Control Console (MCC) for primary operator control and monitoring functions
- The integrated wide display panel (WDP) which presents an overview of the plant status, clearly visible to the entire operating crew
- The Safety Surveillance Panel (SSP) which provides additional safety related displays and a location to perform surveillance tests away from the MCC
- The Non-Safety Surveillance Panel (NSSP)

which provides a location for the diesel generator, fire protection and main generator operator interfaces and the performance of non safety surveillance tests

- The Shift Supervisor's Console (SCC) which provides a location where plant and shift management can monitor both the plant and the plant operators at the MCC

Each of the operator interfaces incorporates advanced man-machine interface technologies to achieve enhanced operability and improved reliability. Human factors engineering principles are incorporated into the design of the panels and into the overall MCR arrangement.

The ESBWR MCR design is traceable to the first ABWRs and to the recently delivered Lungmen ABWR control room, all of which were subject to extensive human factors design and whose operating experiences were further used for additional optimization. The ESBWR MCR layout and panels are shown on Figure 7-8.

Main Control Console (MCC)

TThe Main Control Console (MCC) provides the displays and controls necessary to maintain and operate the plant during normal, abnormal, and emergency conditions. This console is used in conjunction with the information provided on the vertical surface of the WDP. The MCC comprises the work stations for the two control room plant operators (only one is necessary to operate the plant), and is configured such that the operators arc provided with controls and monitoring information necessary to perform assigned tasks and allows the operators to view all of the WDP from their seated positions at the MCC.

The console design incorporates flat panel display devices, and a limited number of hard switches as the primary operator interface devices. The safety-related flat panel displays are located on the left side of the MCC with nonsafety-related DPS and RC&IS displays located between the four divisions.

The four safety-related displays are connected to their corresponding division SSLC/ESF and

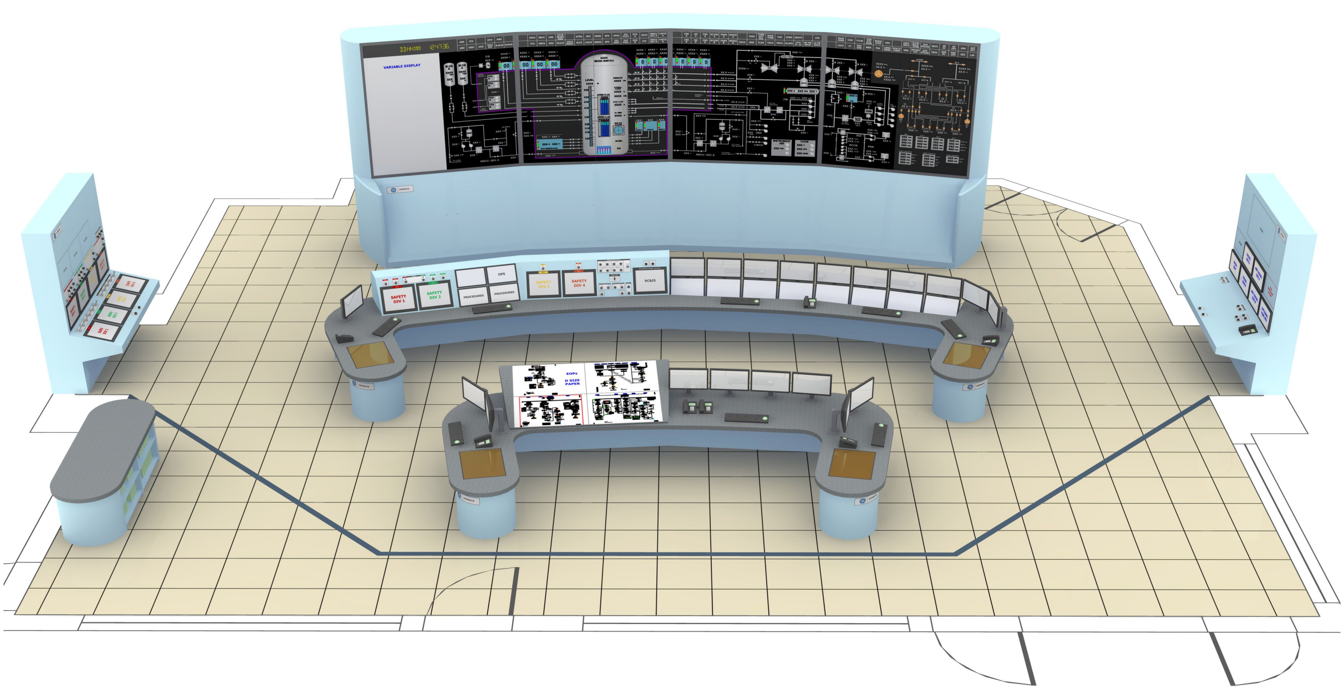


Figure 7-8. ESBWR Main Control Room (MCR) Layout

control and monitor only systems and processes within their division. The many nonsafety-related displays are driven by and connected to different N-DCIS network segments such that they can operate independently of one another if a segment fails, but normally the segments are “invisible” to the operator such that any nonsafety-related VDU can control and monitor any nonsafety-related system. Safety-related information sent through appropriate isolators can also be viewed on the nonsafety-related displays but no control of safety systems is possible. The main control console has a low profile so that the operators and shift supervisor can perform their duties from a seated position and still view the wide display panel.

The normal plant control and monitoring functions are performed in the central and right areas of the console; these are nonsafety-related systems, including the RTNSS/PIP systems, nonsafety-related nuclear systems and the power generation and balance-of-plant (BOP) functions.

The primary means for operator control and monitoring is provided by the color-graphic, flat-panel Visual Display Units (VDUs) mounted on the MCC. The displays are driven by the Q-DCIS and N-DCIS as appropriate. There are many types of system display formats that can be shown on the flat panels and all are accessible from hierarchical menu displays and directly linked to related displays. Displays include summary plant status displays, P&ID/one-line displays, trend plots, system and component status formats, alarm summaries, plant operating procedure guidance displays and plant automation guidance displays.

It is expected that plant operators will nominally assign VDUs to a specific system or systems, it should be understood that this is under the control of the operator; the VDUs are not dedicated by the DCIS (other than safety/non-safety) and any nonsafety-related VDU can show any nonsafety-related display format and each safety- related VDU can display anything within its associated division. Any system can be controlled by only one VDU at a time, but can be monitored by any VDU; this multi redundant display capability ensures continued normal plant operation in the event of a failure of one or more of the flat panels or their display generators.

The system status displays provide information on individual plant systems. The screens on the safety and nonsafety-related VDUs provide direct control and monitoring of safety-related and nonsafety-related systems respectively, at the system level (generally associated with the system’s “semiautomatic” mode of operation) and at the component level (generally associated with the system’s “manual” mode of operation). The application of this screen technology for control of nonsafety-related systems, along with the incorporation of automated plant operation features, was a major factor in reducing the size of the MCC to its present compact dimensions.

The alarm summary displays available on the MCC flat panel displays support the operators’ decision-making process. Once made aware of alarms via the WDP annunciator or mimic displays, alarms can be presented in various ways on the VDUs. For example, first in, last in, acknowledged alarms, unacknowledged alarms, sorted by priority, sorted by system, etc. Alarms are presented visually on P&ID/one-line displays associated with their corresponding components or systems. The presentation of alarms on the WDP or VDUs employs optimization techniques designed to prioritize alarms and filter or suppress alarms not appropriate to the plant condition. An example of this alarm processing would be the suppression of the audible alarms associated with each channel of the Reactor Protection System during the period of a reactor scram or bad vacuum with the mode switch in shutdown. Note that although an alarm may be suppressed, all alarms are recorded by the plant historian and all can be recalled by the operator.

The flat panel display devices are used to support both safety-related and nonsafety-related system monitoring and control functions. The flat panel displays, which are used as safety-system interfaces, are fully qualified to Class IE standards. The safety- related flat displays are located on the left side of the MCC and provide for control and monitoring of the redundant and independent divisions of the ECCS, safety-related post-accident monitoring signal, monitoring of the RTIF and NMS systems and the various ICPs. There are two flat panel displays per division, one on the MCC and the other on the safety surveillance panel.

Flat panel displays for monitoring and control of major non-safety systems are also located on the center and right sides of the MCC. In addition to the flat panel display devices described above, the MCC is equipped with dedicated, hard-switches located on the horizontal desk surfaces of the console. Although the design intent is to minimize the hard-switches, some provided for surveillance, bypass, fail-safe or not important to safety functions (i.e., reactor scram, turbine trip, fire pump start, etc.). A limited number of nominally dedicated operator interfaces are provided in the left side of the MCC for key systems such as the Rod Control and Information System. These dedicated interfaces may be co-located with hard switches and indicators to provide quick and convenient access to key system interfaces under all plant condition

Wide Display Panel (WDP)

The Wide Display Panel (WDP) is a large vertical board that provides information on overall plant status with real-time data during all phases of plant operation. The information presented on the WDP is clearly visible from the Main Control Console, the shift supervisor's console, and most other positions in the control room where support personnel may be stationed, The WDP provides system-level annunciator displays, a mimic display and a large variable display.

The WDP is located immediately in front of and is easily viewable by the operators when they are at their normal work positions at the main control console; WDP-displayed information is sized for the distances involved. The WDP includes a plant mimic display whose function is to present an overview of plant status with most of the important plant parameters displayed next to their mimic analog.

The mimic display is arranged over most of the WDP vertical area and includes the critical plant parameters required for a safety parameter display system and most post-accident monitoring parameters. Specific information displayed on this panel includes the status of the core cooling systems, reactor pressure vessel and core parameters, containment and radiation parameters and the status of safety-related equipment. The information displayed completely satisfies the requirement for safety parameter and post-accident monitoring without the need for any other display equipment (although any nonsafety-related VDU can also display SPDS and post-accident parameters and alarms). Generally the left side of the plant mimic displays the nuclear steam supply system and ECCS; the right side of the plant mimic display contains information on the BOP power generation cycle, such as the condensate and feedwater system, turbine generator, and power transmission systems. Also within the area of the mimic display dedicated alarm windows are provided for important, plant-level alarms that affect plant availability or safety. Examples of the plant-level alarms include high reactor pressure, low reactor water level and high suppression pool temperature.

Along the top of the WDP are located spatially dedicated detailed system-level annunciators generally located above their respective systems on the plant mimic display. Although a display, the annunciators are traditional in that they offer audible, color-coded, fixed and blinking displays as dictated by Human Factors Engineering (HFE). The annunciators and datalinks are redundant and are operated by the same redundant workstations and power supplies that implement the alarm management system and that also drive the alarm display formats on the VDUs. The annunciators use the same filtering/suppression algorithms used for the VDU alarms to minimize operator workload as a function of plant status. The intent of the annunciators is to bring the operator's attention to a system alarm (since the various VDUs might not be monitoring that system) so he can use the MCC VDUs to determine the alarm details.

The WDP also has a large variable display located on the right vertical. The basic purpose of the large variable display is to provide information on important plant process parameters that supplements the overview information on the mimic display. The information presented on the large variable display can be changed at the operator's discretion, depending on the plant operating conditions and the needs of the operating crew. Any display format available on the MCC VDUs can also be displayed on the large variable display. Examples of the full-color graphic displays that can be shown on the variable display are the various flat panel display formats (including SPDS and plant normal and emergency procedures) which would be selected under plant emergency conditions or alarm displays or trend plots if a specific component or system was being investigated.

The ESBWR is equipped with closed-circuit television cameras to provide remote observation of equipment and operations in areas that are not normally accessible and of other critical activities such as fuel handling and maintenance tasks. The operator can select a camera or cameras and display the resulting picture on the WDP. Communication between the control room crew and other areas of the plant is enhanced with this visual feedback capability. These closed-circuit TVs have high definition with color capability and are radiation hardened where appropriate and provide the capability for surveillance of systems and equipment during normal plant operation. In addition, these devices can be used for control and monitoring of plant systems during maintenance and refueling outages and during periods when a portion of the MCC may be taken out of service for maintenance.

Safety Surveillance Panel (SSP)

The SSP panel is not normally used but contains four (one per division) safety-related VDUs with the same capability as those on the MCC and can be used if the MCC VDU fails. The panel includes nonsafety-related VDUs that are dedicated to maintenance, diagnostics, calibration, monitoring and troubleshooting the NMS and RTIF systems; the VDUs cannot be used for anything but monitoring unless the associated division is made inoperative by a keylocked switch (which will cause that division to trip unless bypassed). The SSP also includes the various divisional bypass switches (only one division can be bypassed at a time) and specialized surveillance switches and mode switches. The SSP has a large VDU at the top of the vertical section that can be used to display any format available on the MCC but is usually used to display information associated with surveillance testing.

Non-Safety Surveillance Panel (NSSP)

The NSSP panel is not normally used but contains operator interfaces to:

- Support testing and synchronizing the plant diesel generators
- Provide the hard-wired and VDU interface to the fire protection system
- Provide for the synchronization of the main generator

The above interfaces can also support maintenance, diagnostics, calibration, monitoring and troubleshooting the generator and fire protection systems. The NSSP also includes the various redundant nonsafety-related system (e.g., ATLM, RWM) bypass switches (only one of a redundant pair can be bypassed at a time) and specialized surveillance switches and mode switches. The NSSP has a large VDU at the top of the vertical section that can be used to display any format available on the MCC but is usually used to display information associated with surveillance testing.

Shift Supervisor's Console (SSC)

The shift supervisor's console is set back directly behind the operators in a position which ensures that a clear view of all control room activities and the WDP. The SSC includes most of the MCR communication equipment and a supervisor's console that includes four VDUs for monitoring plant status or following the operators on the MCC. The SSC also includes two large horizontal VDUs that can be used to display plant drawings (P&IDs and one-lines) or emergency procedure guidelines at near full size. The SSC includes two normally unmanned positions and workstations for the reactor engineer and/or the shift technical advisor (who also has the equipment to communicate with the technical support center). The workstations are capable of showing any format that is available on the MCC (without control capability) as well as specialized formats (i.e., 3D Monicore detail displays or detail SPDS and TSC formats; these positions can be manned as needed without requiring access to or interfering with the MCC.

Plant Automation System

The ESBWR design incorporates extensive automation of the operator actions which are required during a normal plant startup, shutdown and power range maneuvers. The automation features adopted for the ESBWR provide for enhanced operability and improved capacity factor relative to conventional BWR designs. The extent of automation implemented in the ESBWR has been carefully selected to ensure that the primary control of plant operations remains with the operators. The operators remain fully cognizant of the plant status and can intervene in the operation at any time, if necessary.

The Plant Automation System (PAS) provides the capability for automatic, supervisory control of the entire plant by supplying setpoint commands to the various nonsafety-related RTNSS/PIP and power generation BOP controllers as operator demands, changing load demands and plant conditions dictate. The primary purposes of the PAS are reactivity control, heatup and pressurization control, turbine generator roll and synchronization control, reactor power control, generator power control (MWe control). Together these PAS functions can take the ESBWR from cold shutdown, through criticality, through heatup to rated temperature and pressure, and to rated power, and return from rated power to cold shutdown.

The ESBWR PAS and downstream controller automation design provides for three distinct automation modes: Automatic, Semi-Automatic, and Manual. In the Automatic mode, the operator initiates automated sequences of operation from the MCC. Periodic breakpoints are inserted in the automated sequence that require operator verification of plant status and manual actuation of a breakpoint control push-button to allow the automated sequence to continue. When a change in the status of a safety system is required, automatic prompts are provided to the operator and the automation is suspended until the operator manually completes the necessary safety-system status change (in the ESBWR design no nonsafety-related system, processor or component can control or adversely affect a safety-related system). In the Semi-Automatic mode of operation, the progression of normal plant operations is monitored and automated prompts and guidance are provided to the operator; however, all actual control actions must be performed manually by the operator. In the Manual mode of operation, no automated operator guidance or prompts are provided. The operator can completely stop an automatic operation at any time by selecting the Manual mode of operation; this will also happen automatically for any abnormal events, such as turbine trips or reactor scrams.

The PAS is implemented on a triply-redundant hardware/software platform. The functions of the PAS are accomplished by suitable algorithms for different modes of reactor operation which include approach to criticality, heatup, reactor power increase, automatic load following, reactor power decrease and shutdown. The PAS interfaces with the MCR main control console to perform its designed functions. From the MCR main control console the PAS provides for Human Factors Engineering (HFE) optimized breakpoints to allow the operator to review and monitor the automation process; PAS does not go to 100% power in one step but instead presents (for example) a breakpoint, like reactor criticality. When selected the PAS will check for suitable breakpoint initiation conditions, suitably command the RC&IS to pull rods to criticality while monitoring NMS, indicating to the operator that criticality has been achieved and then stopping until the next breakpoint is initiated. Approximately 25 – 30 breakpoints are used for automatic plant startup, power operation and shutdown functions.

The same monitoring systems (for example the RWM rod blocks) are in place for automation as for manual operation such that no plant safety-related or protective functions are bypassed; at no time can PAS control any reactor safety system, nor would the safety systems accept such commands. PAS can only issue supervisory commands to the "downstream" controllers; these controllers still remain responsible for the operation, interlocks, alarms and protective features of their controlled systems. The downstream systems must be individually, manually and consciously be put into the automation mode that allows them to be controlled by PAS; the operator and the downstream controller process and self-diagnostic alarms can drop the systems out of automation at any time. The major systems commanded by PAS include the RC&IS, Steam Bypass and Pressure Controller, the Feedwater Temperature Controller and the Turbine Generator Control systems. PAS will sequence the downstream controllers in the same way that they would be sequenced in manual plant operation.

As previously described, the PAS presents the operator with a series of breakpoint controls on the main control console nonsafety-related VDUs for a prescribed plant operation sequence. When all the prerequisites are satisfied for a prescribed breakpoint in a control sequence, a permissive is requested and upon operator acceptance, the prescribed control sequence is initiated or continued. The PAS then initiates demand signals to various system controllers to carry out the pre-defined control functions. For non-automated operations that are required during normal startup or shutdown (such as a change of Reactor Mode Switch status), automatic prompts are provided. Automated operations continue after the prompted actions are completed manually.

For reactor power control, the PAS contains algorithms that can change reactor power by control rod motions and feedwater temperature changes. A predefined control rod sequence is followed when manipulating control rods for reactor criticality, heatup, power changes and automatic load following. For reactor power control by FW temperature change, the PAS can provide the FW temperature control setpoint to allow reactor power maneuvering without moving control rods. Each of these functions has its own control algorithm to achieve its design objective. In combination, the two reactor power control methods are utilized to form a sequential step-by-step power maneuvering strategy for the control rod pattern/movement and FW temperature change. During automatic load following operation, the PAS interfaces with the turbine-generator control system to coordinate turbine and reactor power changes for stable operation and performance.

The normal mode of operation of the PAS is Automatic. If any system or component conditions are abnormal during execution of the prescribed sequences, the PAS automatically switches into the Manual mode. With the PAS in the Manual mode, any in-progress operation stops and alarms are activated in the MCR. Also with the PAS in Manual mode, the operator can manipulate control rods through the normal controls. A failure of the PAS does not prevent manual control of reactor power or any other plant controller and does not prevent safe shutdown of the reactor.

It should be noted that individual ESBWR system design, outside of PAS control, provides for Semi-Automatic operation within a system to improve operability and operator monitoring. For example the RWCU/SDC system has the ability to automate the cleanup system modes and shutdown cooling modes. If the RWCU/SDC is set (by the operator) to Automatic, it will accept the mode change command from PAS (which will issue the command in the proper sequence as the plant is brought to power). If RWCU/SDC is set to Semi-Automatic, the PAS is disconnected and the RWCU/SDC VDU format will allow the operator to command the mode change whenever he feels it is the correct time to do so. If RWCU/SDC is set to Manual, the operator must operate each component (valves, pumps, etc.) manually from the VDU in the correct order to accomplish the mode change

Operation

The ESBWR control room design provides the capability for a single operator to perform all required control and monitoring functions during normal plant operations as well as under emergency plant conditions. The ESBWR is designed such that no operator intervention is required for 72 hours for all design basis events.

One-man operation is possible due to implementation of several key design features:

- The Wide Display Panel for overall plant monitoring
- Plant-Level Automation
- System-Level Automation
- The compact MCC design
- Implementation of operator guidance functions, which display appropriate operating sequences on the main control panel flat panel displays or the WDP

The operator only has to click on any alarm to see the alarm response procedure text and, similarly, normal and emergency operating procedure text is available on any non-safety VDU (including the large variable display). The role of the operator will primarily be one of monitoring the status of individual systems and the overall plant and the progress of automation sequences, rather than the traditional role of monitoring and controlling individual system equipment. However, to foster a team approach in plant operation and to maintain operator vigilance, the operating staff organization for the reference ESBWR control room design can support having two operators normally stationed at the main control console.

During emergency plant operations, plant-level automation is automatically suspended, but system-level automation is available. One operator would be responsible for the NSS systems and the other for the BOP systems, with the supervisors providing both direction and guidance. Again, system-level automation allows for simplified execution of both the safety and non-safety system operations. In lieu of system-level automation, direct manual control of individual system equipment is available on the flat displays.

Surveillance

Unlike previous BWR designs, the ESBWR DCIS is designed to facilitate technical specification surveillance tests during normal operation for operator confirmation. For example, redundant parameters between the four safety divisions are automatically and continuously checked for consistency and alarmed when not consistent; the daily channel checks are either the absence of an alarm or an automatically generated printout of the appropriate parameters. Similarly, although digital setpoints do not drift, the DCIS automatically and continuously checks the various trip setpoints and alarms when they are not correct. The redundant ESBWR technical specification monitor system automatically performs the above tasks in addition to continuously receiving microprocessor diagnostics and application-specific information from safety-related and important nonsafety-related systems and will alarm on stalled, incorrect, or non-communicating systems.

Maintenance

As with all modern DCIS, the ESBWR incorporates extensive self-diagnostics and self-calibration; a self-detected fault will be alarmed (in fail safe equipment, tripped) and will direct the operator to the lowest level of failed module/card; these can be replaced online with no loss of power generation.

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Chapter 8
Plant Layout and Arrangement

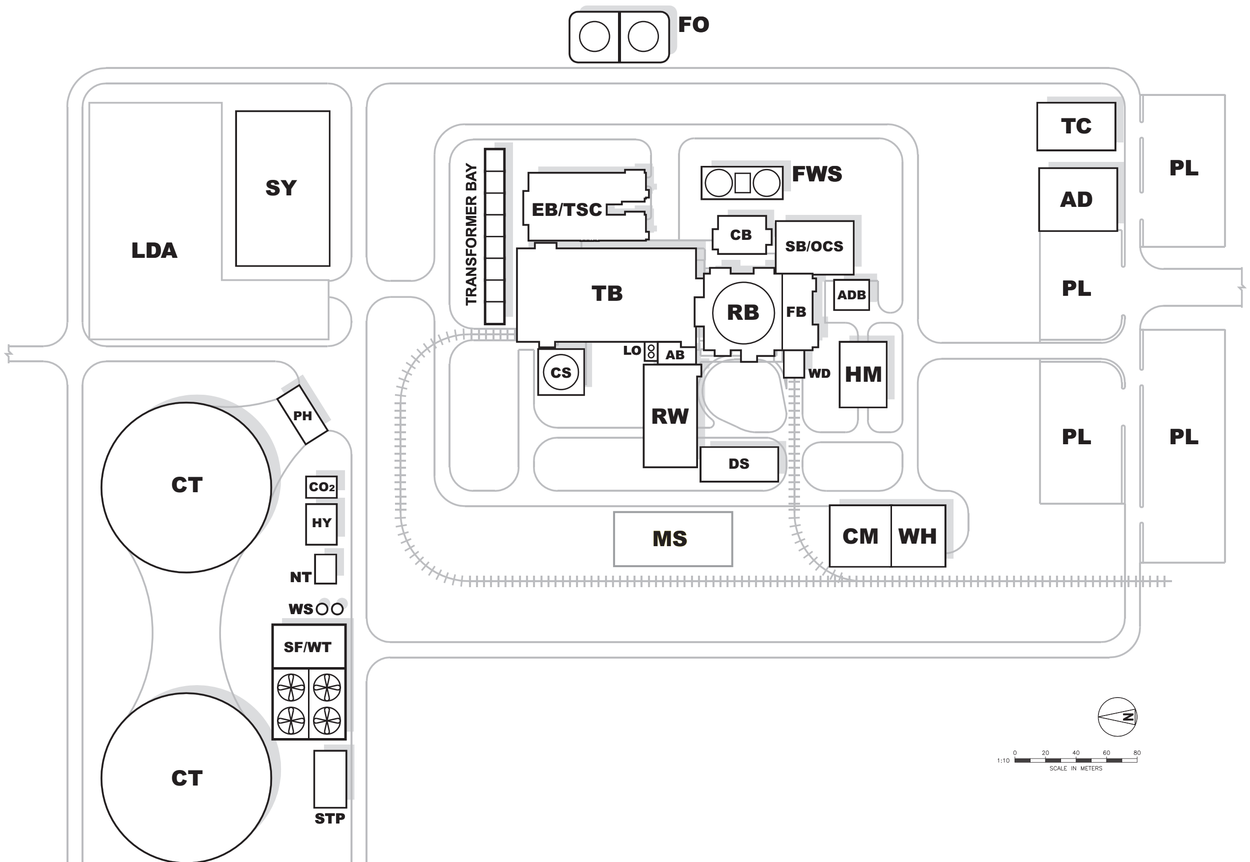
Plant Layout

The ESBWR Standard Plant includes buildings dedicated exclusively or primarily to housing systems and equipment related to the nuclear system or controlled access to these systems and equipment. Nine such main buildings are within the scope for the ESBWR.

- Reactor Building (RB) - houses safety-related structures, systems and components (SSC), except for the main control room, safety-related Distributed Control and Information System equipment rooms in the Control Building and spent fuel storage pool and associated auxiliary equipment in the Fuel Building. The Reactor Building includes the reactor, containment, refueling area and auxiliary equipment
- Fuel Building (FB) - houses the spent fuel storage pool and its associated auxiliary equipment
- Control Building (CB) - houses the main control room and safety-related controls outside the reactor building
- Turbine Building (TB) - houses equipment associated with the main turbine and generator, and their auxiliary systems and equipment, including the condensate purification system and the process offgas treatment system
- Electrical Building (EB) - houses the two non-safety-related standby diesel generators and their associated auxiliary equipment. It also houses the non-safety-grade batteries
- Radwaste Building (RW) - houses equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant
- Firewater service complex - consists of two water storage tanks and a fire pump enclosure that share a common basemat
- Ancillary Diesel Building - houses the ancillary diesel generators and their associated support systems
- Service Building - houses the equipment and control facilities associated with personnel entry into the reactor building and turbine building, eating areas, radiation protection, changing rooms, shops, and offices

Development of the ESBWR plant and building arrangements has been guided by the following criteria:

- Retain the passive and well-established BWR pressure suppression containment technology. Use of the horizontal vent configuration was confirmed for the Mark III containments
- Separate clean and controlled radiation areas to minimize personnel exposure during operation and maintenance
- Emphasize improved layout of systems to improve access and equipment maintenance activities
- Locate major equipment for early installation using opentop construction approach and large-scale modularization
- Arrange the Reactor Building around the primary containment to provide multiple barriers to post-accident fission product leakage, and high tolerance to external missiles
- Place the passive safety systems (GDCS, PCCS, ICS) within and adjacent to the primary containment



BUILDING LEGEND:

AB	=	AUXILIARY BOILER	FPE	=	FIRE PUMP ENCLOSURE	RW	=	RADWASTE BUILDING
AD	=	ADMINISTRATION BUILDING	FWS	=	FIRE WATER STORAGE TANK	SB	=	SERVICE BUILDING
ADB	=	ANCILLARY DIESEL BUILDING	HM	=	HOT MACHINE SHOP & STORAGE	SF	=	SERVICE WATER BUILDING
CB	=	CONTROL	HY	=	HYDROGEN BULK STORAGE	STP	=	SEWAGE TREATMENT PLANT
CM	=	COLD METAL SHOP	LDA	=	LAY DOWN AREA	SY	=	SWITCH YARD
CO^2	=	CARBON DIOXIDE STORAGE TANK	LO	=	DIRTY/CLEAN LUBE OIL STORAGE TANK	TB	=	TURBINE BUILDING
CS	=	CONDENSATE STORAGE TANK	MS	=	MISCELLANEOUS SERVICE AREA	TC	=	TRAINING CENTER
CT	=	MAIN COOLING TOWER	NT	=	NITROGEN STORAGE TANK	TSC	=	TECHNICAL SUPPORT CENTER
DS	=	INDEPENDENT SPENT FUEL STORAGE	OSC	=	OPERATION SUPPORT CENTER	WD	=	WASH DOWN BAYS (EQUIPMENT ENTRY)
EB	=	ELECTRICAL BUILDING	PH	=	PUMP HOUSE	WH	=	WAREHOUSE
FB	=	FUEL BUILDING	PL	=	PARKING LOT	WS	=	WATER STORAGE
FO	=	DIESEL FUEL OIL STORAGE TANK	RB	=	REACTOR BUILDING	WT	=	WATER TREATMENT

Figure 8-1. ESBWR Site Plan

- Separate temporary fuel storage from long-term fuel storage by adopting the Inclined Fuel Transfer System from BWR/6

The site plan of the ESBWR includes the Reactor, Control, Fuel, Turbine, Radwaste, Electrical, Firewater Service Complex, Ancillary Diesel and Service Buildings, as well as other supporting buildings. Provision is made within the Fuel Building for ten years plus a full-core offload of spent fuel storage. Separate buildings can be provided for additional onsite waste storage. Figure 8-1 illustrates a site plan of the ESBWR for a single unit arrangement. Although the heat sink is based on natural draft cooling towers in the figure, other heat sinks are possible. A 3-dimensional conceptual rendering of a single unit site is shown in Figure 8-2. In this picture, mechanical draft cooling towers are shown.

A 3-dimensional perspective of most of the power block (the Reactor, Fuel, Control, and Turbine Buildings) is shown in Figure 8-3.

The ESBWR design is an enhanced arrangement to minimize material quantities. This, when combined with the volume reduction compared to previous designs, contributes to the substantial reduction in both the estimated construction schedule and plant capital cost.

The layout of the Reactor and Turbine Buildings was based on the following considerations:

- Personnel access for all normal operating and maintenance activities is a primary concern. Access routes from the change room to contaminated Reactor and Turbine Building areas are as direct as possible and clearly separated from clean routes. At each floor, 360° access is provided, if practical, to enhance daily inspections and normal work activities. Access to equipment not reachable from floor level is via platform and stair access wherever possible.
- Equipment access is provided for all surveillance, maintenance, and replacement activities with local service areas and laydown space. Adequate hallways and other equipment removal paths, including vertical access hatches, are provided for moving equipment from its installed position to service areas or out of the building for repair. Lifting points, monorails and other installed devices are provided to facilitate equipment handling and minimize the need for re-rigging individual equipment movements. Equipment access also considers the need for temporary construction access.
- Radiation levels are controlled and minimized. The Reactor Building is divided into clean and controlled areas. Once personnel enter a clean or controlled area, it is not possible to crossover to the other area without returning to the change area. Redundant equipment is located in shielded cells to permit servicing one piece of equipment while the plant continues to operate. Valve galleries are provided to minimize personnel exposure during system operation or preparation for maintenance.
- The turbine-generator is aligned with its axis in-line with the Reactor Building. This is done to minimize the possibility of turbine missile impact on the containment vessel.

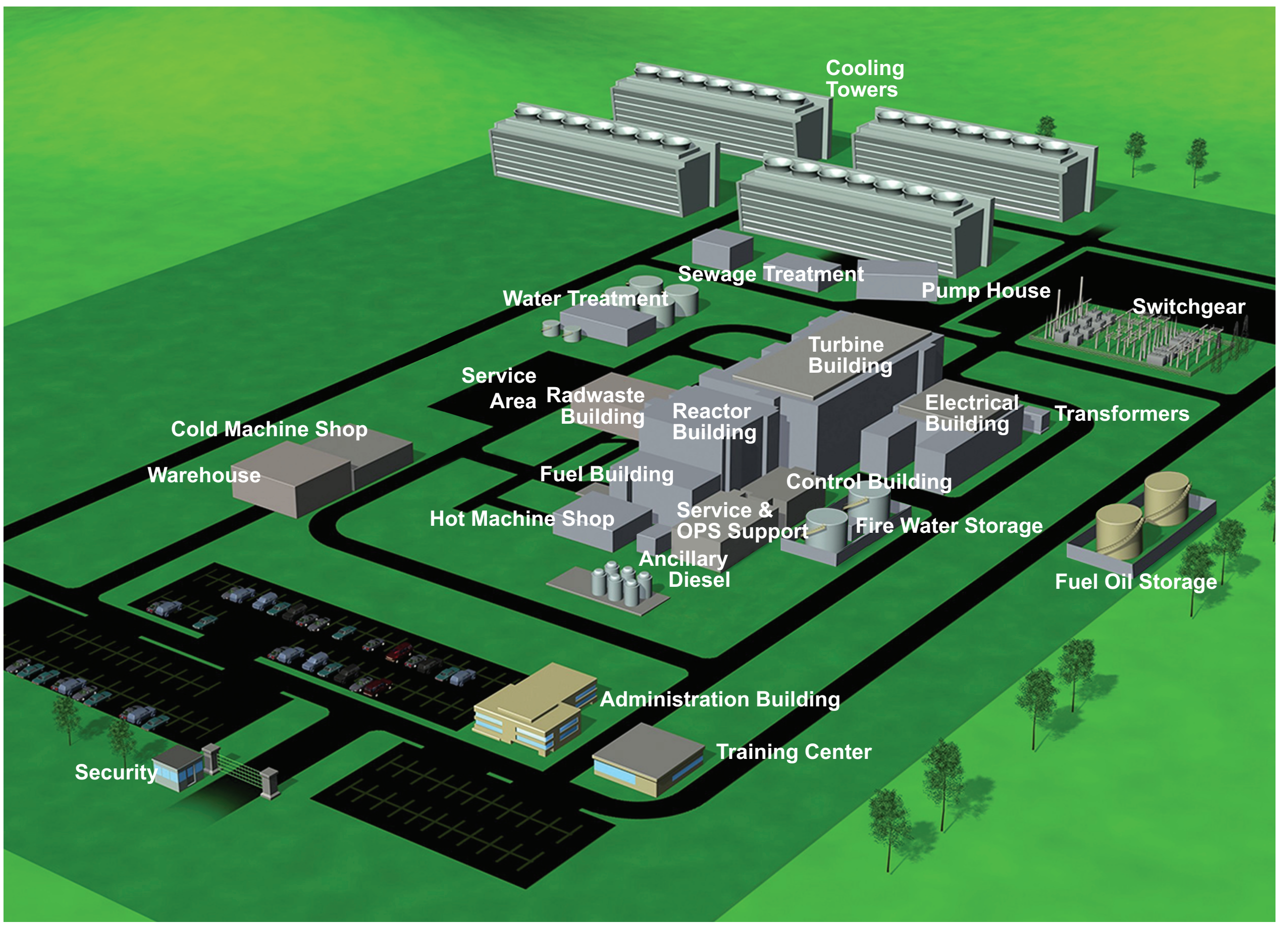


Figure 8-2. ESBWR Conceptual Site Plan

- The main and auxiliary transformers are located adjacent to the main generator at the end of the Turbine Building. This location minimizes the length of the isophase bus duct between the generator and transformers, as well as the power supply cables back to the main electrical area of the power block

The site plan includes consideration for construction space and site access. The arrangement provides a clear access space around the Reactor and Turbine Buildings for heavy lift mobile construction cranes without interference with other cranes, access ways and miscellaneous equipment.

Safety Buildings

The ESBWR safety buildings are the Reactor Building, Fuel Building and Control Building.

The Reactor Building houses the containment, drywell, and major portions of the Nuclear Steam Supply System, steam tunnel, refueling area, Isolation Condensers, Emergency Core Cooling Systems, HVAC System and other supporting systems.

The ESBWR Reactor/Fuel Building is a reinforced concrete structure. The integrated Reactor-Fuel Building and containment structure has been analyzed for a safe shutdown earthquake (SSE) of a minimum of 0.3g for an all-soils site envelope.

The Reactor Building surrounds the primary containment and provides a secondary barrier to fission-product release.

Figure 8-4 shows an elevation view of the Reactor Building and Fuel Building. These two buildings have a common basemat. Visible in this view is the Inclined Fuel Transfer System (IFTS), which manages the transfer of fuel between the Reactor Building and the Fuel Building.

Figure 8-5 shows an elevation view of the Reactor Building and adjacent Control Building.

The refueling floor is shown in Figure 8-6. On this figure the Reactor Building crane and refueling machine are displayed.

The four Isolation Condenser (IC) heat exchangers, six Passive Containment Cooling System (PCCS) heat exchangers and their interconnected pools are shown in Figure 8-7.

Main Steam and Feedwater piping and part of the steam tunnel connecting the Reactor Building with the Turbine Building can be seen in Figure 8-8. At this elevation the SRVs and MSIVs can be seen on the Main Steam piping. Also showing at this elevation are the Standby Liquid Control System (SLC) accumulators in the Reactor Building and HVAC equipment in the Fuel Building.

The drywell-wetwell vent piping and SRV quenchers can be seen in Figure 8-9. Also shown at this elevation is the sliding block support system for the RPV.

The safety-grade batteries are the main feature of Figure 8-10.

The basemat level in the Reactor Building contains the Hydraulic Control Units (HCU) for the FMCRDs as well as the pumps and heat exchangers for the RWCU/SDC system, shown in Figure 8-11. Also seen at this level are the FAPCS pumps and heat exchangers in the Fuel Building.

Figure 8-12 shows the Main Control Room (MCR) general layout.

Careful attention has been given to ease of construction with this building arrangement. The construction scheme embodied assumes that the major cooling equipment has been placed on the lowest floors of the building to allow early installation during construction.

Modularization techniques are implemented to reduce costs and improve construction schedules. These techniques are applied to such Reactor-Fuel Building items as: (1) building reinforcing bar assemblies, (2) structural steel assemblies, (3) steel liners for the containment and associated water pools, (4) selected equipment assemblies, and (5) drywell platform and piping supports.

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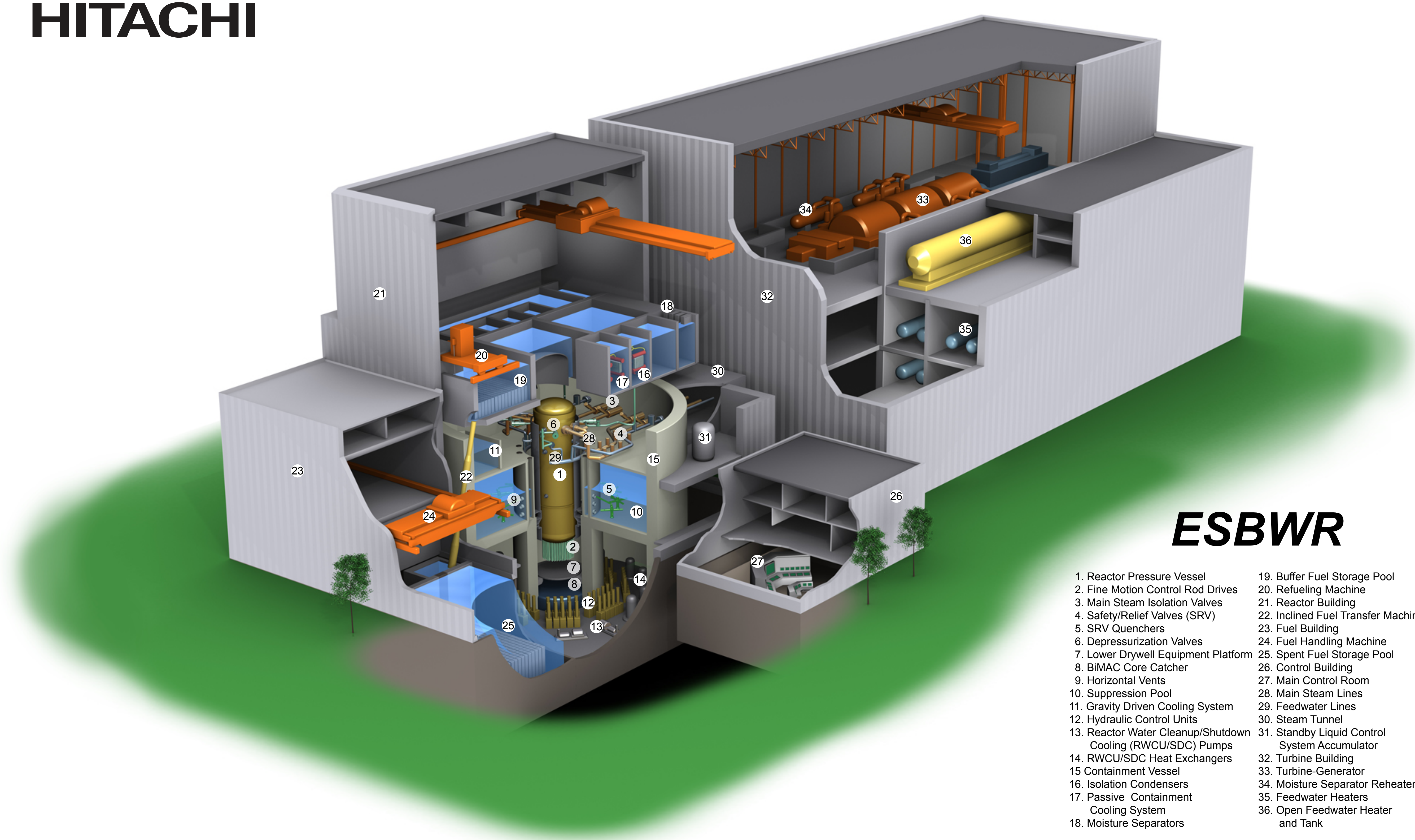

ESBWR

1. Reactor Pressure Vessel
2. Fine Motion Control Rod Drives
3. Main Steam Isolation Valves
4. Safety/Relief Valves (SRV)
5. SRV Quenchers
6. Depressurization Valves
7. Lower Drywell Equipment Platform
8. BiMAC Core Catcher
9. Horizontal Vents
10. Suppression Pool
11. Gravity Driven Cooling System
12. Hydraulic Control Units
13. Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) Pumps
14. RWCU/SDC Heat Exchangers
15 Containment Vessel
16. Isolation Condensers
17. Passive Containment Cooling System
18. Moisture Separators
19. Buffer Fuel Storage Pool
20. Refueling Machine
21. Reactor Building
22. Inclined Fuel Transfer Machine
23. Fuel Building
24. Fuel Handling Machine
25. Spent Fuel Storage Pool
26. Control Building
27. Main Control Room
28. Main Steam Lines
29. Feedwater Lines
30. Steam Tunnel
31. Standby Liquid Control System Accumulator
32. Turbine Building
33. Turbine-Generator
34. Moisture Separator Reheater
35. Feedwater Heaters
36. Open Feedwater Heater and Tank

Figure 8-3. ESBWR Cutaway View of the Reactor, Fuel, Control and Turbine Buildings

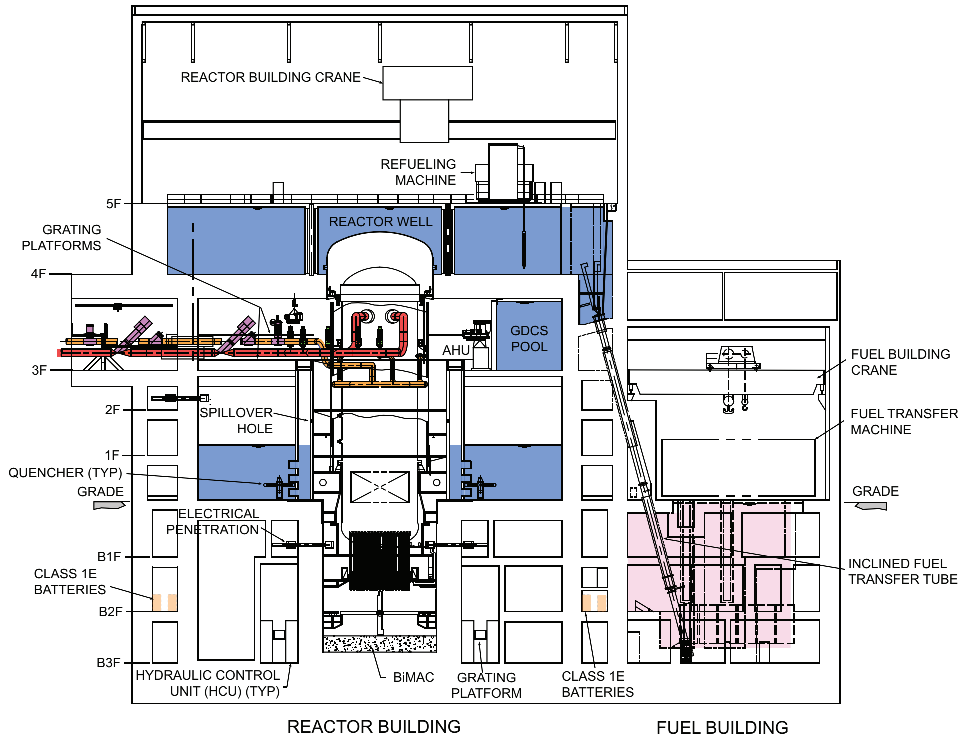

Figure 8-4. ESBWR Reactor and Fuel Building Section AA

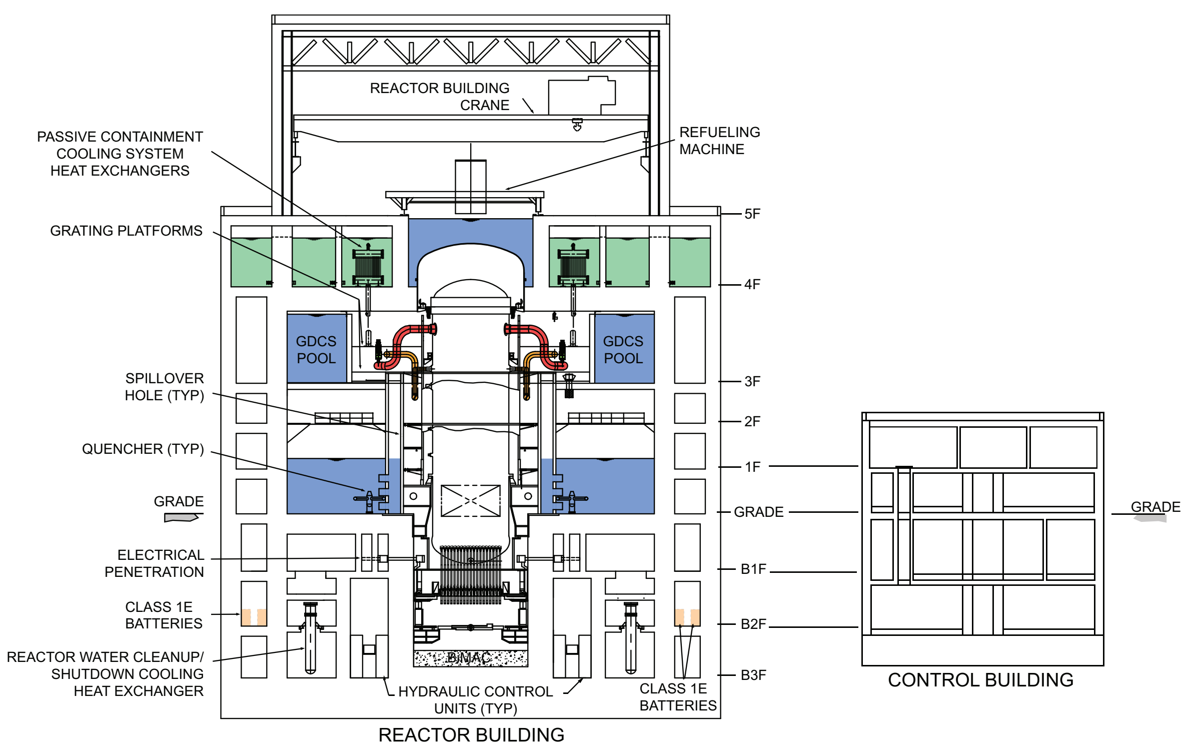

Figure 8-5. ESBWR Reactor and Control Building Section BB

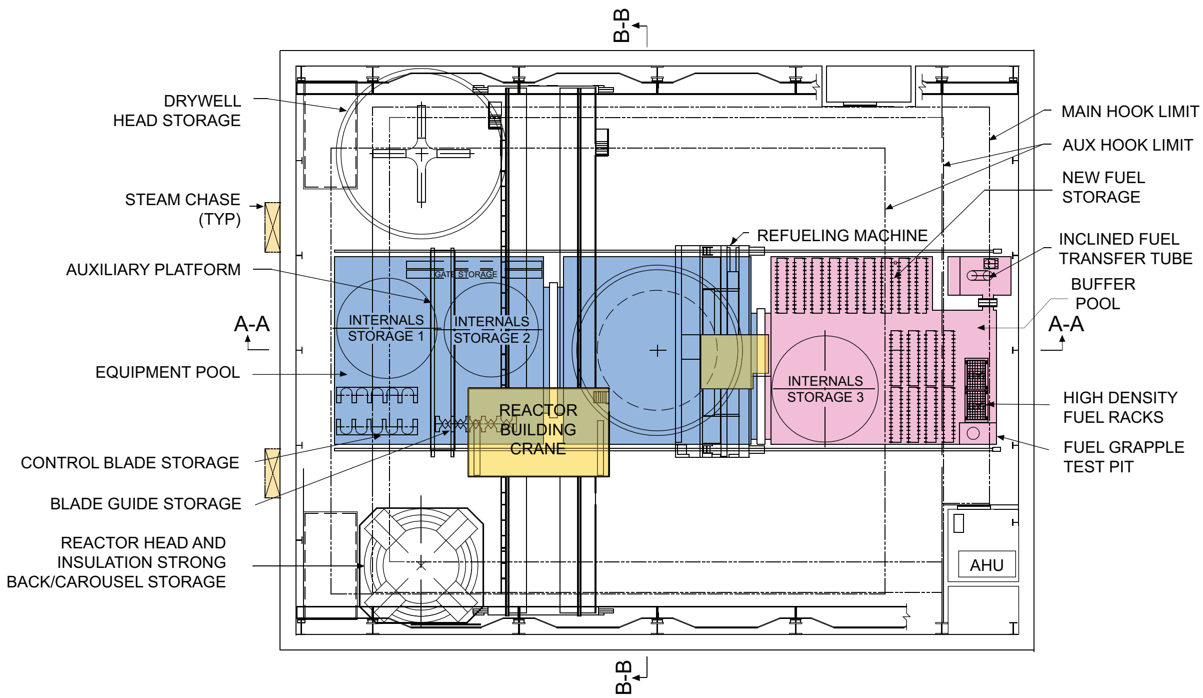

Figure 8-6. ESBWR Reactor Building Refueling Floor (5F)

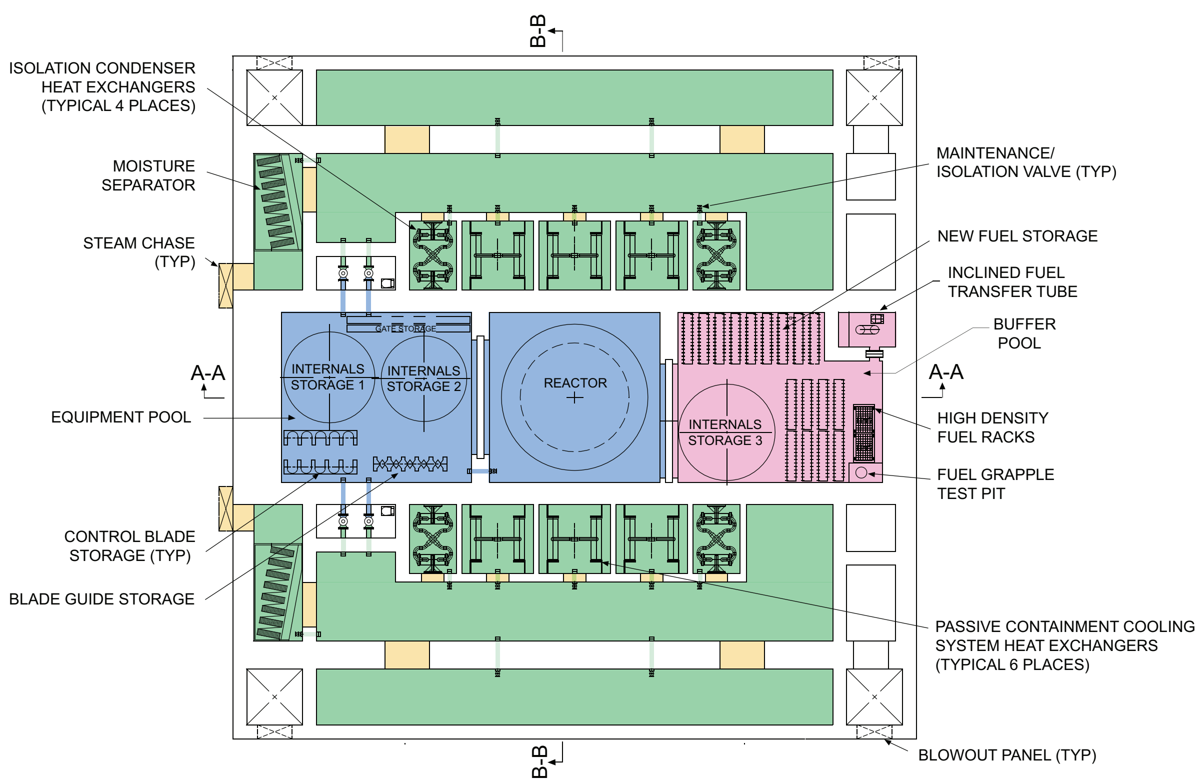

Figure 8-7. ESBWR Reactor Building IC-PCCS Level (4F)

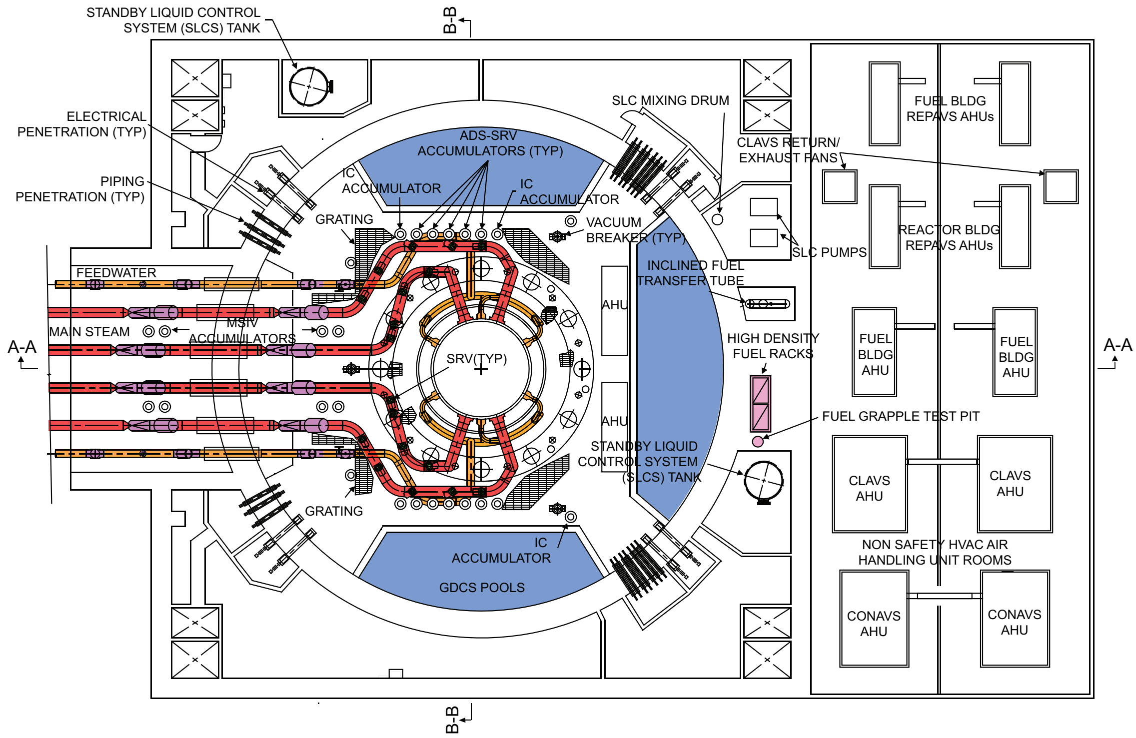

Figure 8-8. ESBWR Reactor and Fuel Building Steam Line Level (3F)

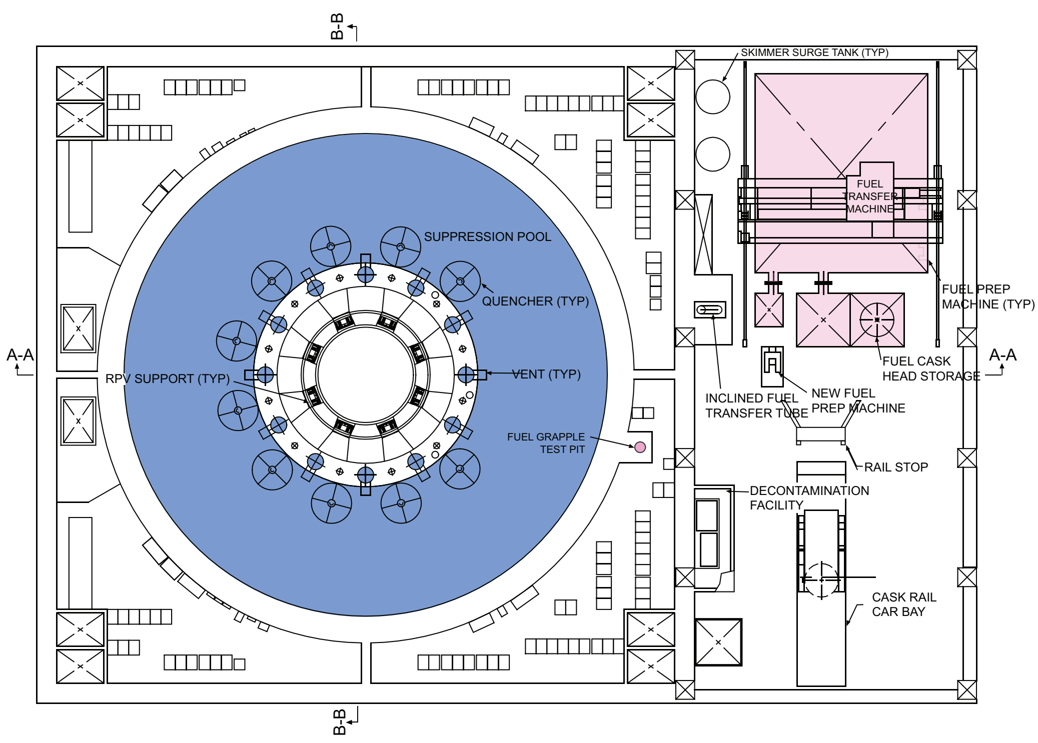

Figure 8-9. ESBWR Reactor and Fuel Building Suppression Pool Level (Grade)

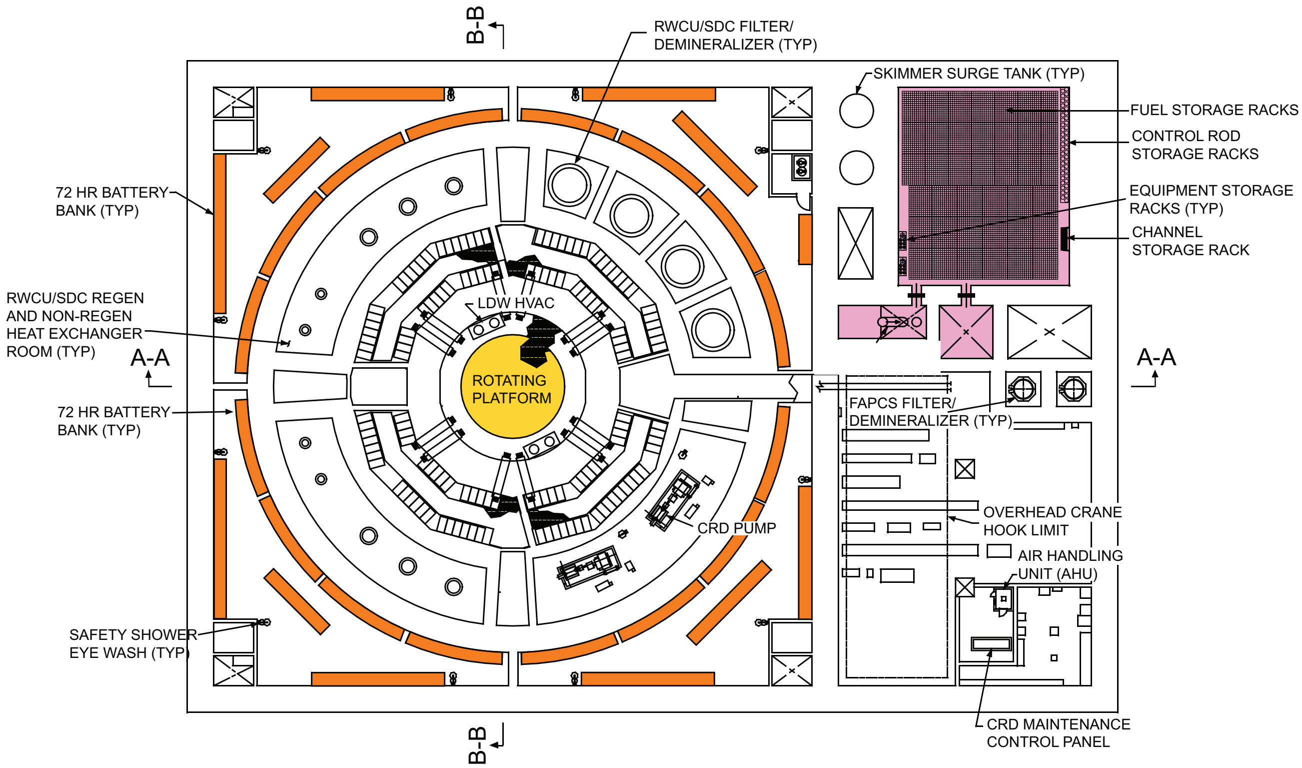

Figure 8-10. ESBWR Reactor and Fuel Building Battery Level (B2F)

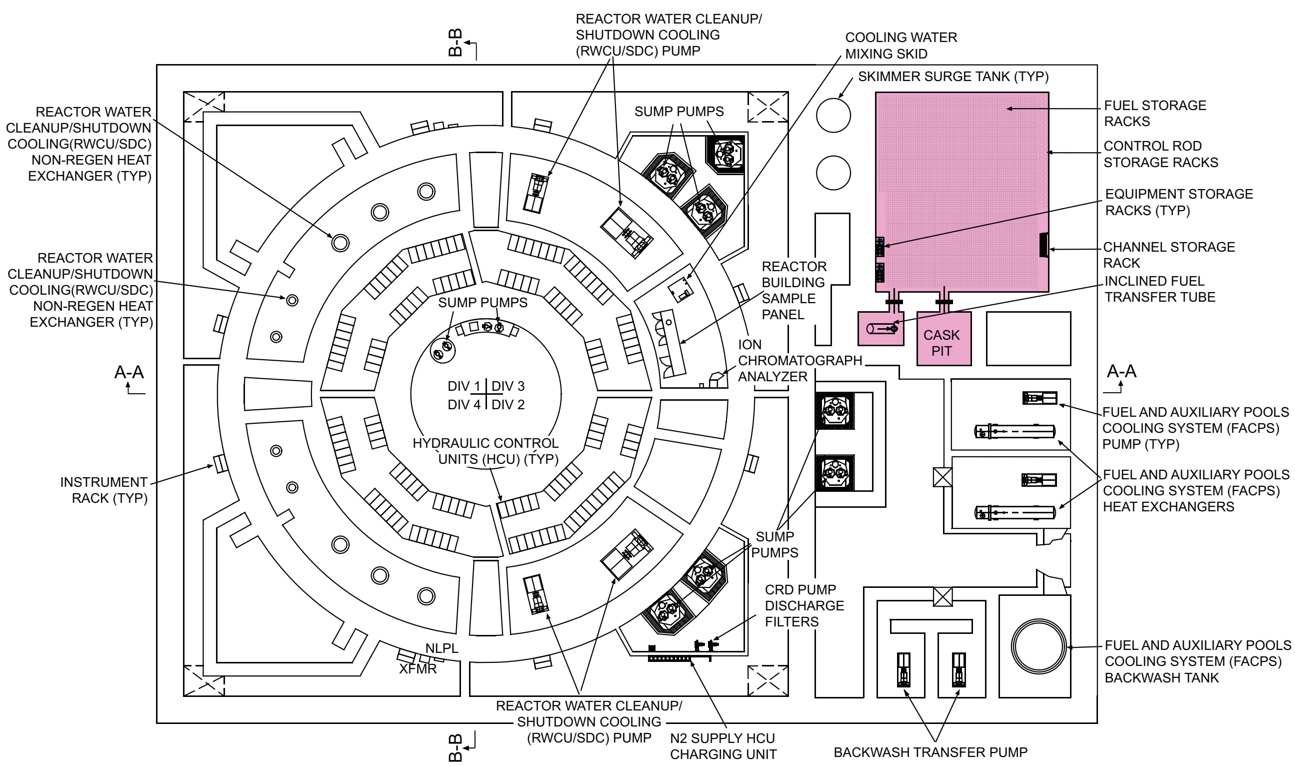

Figure 8-11. ESBWR Reactor and Fuel Building Basemat Level (B3F)

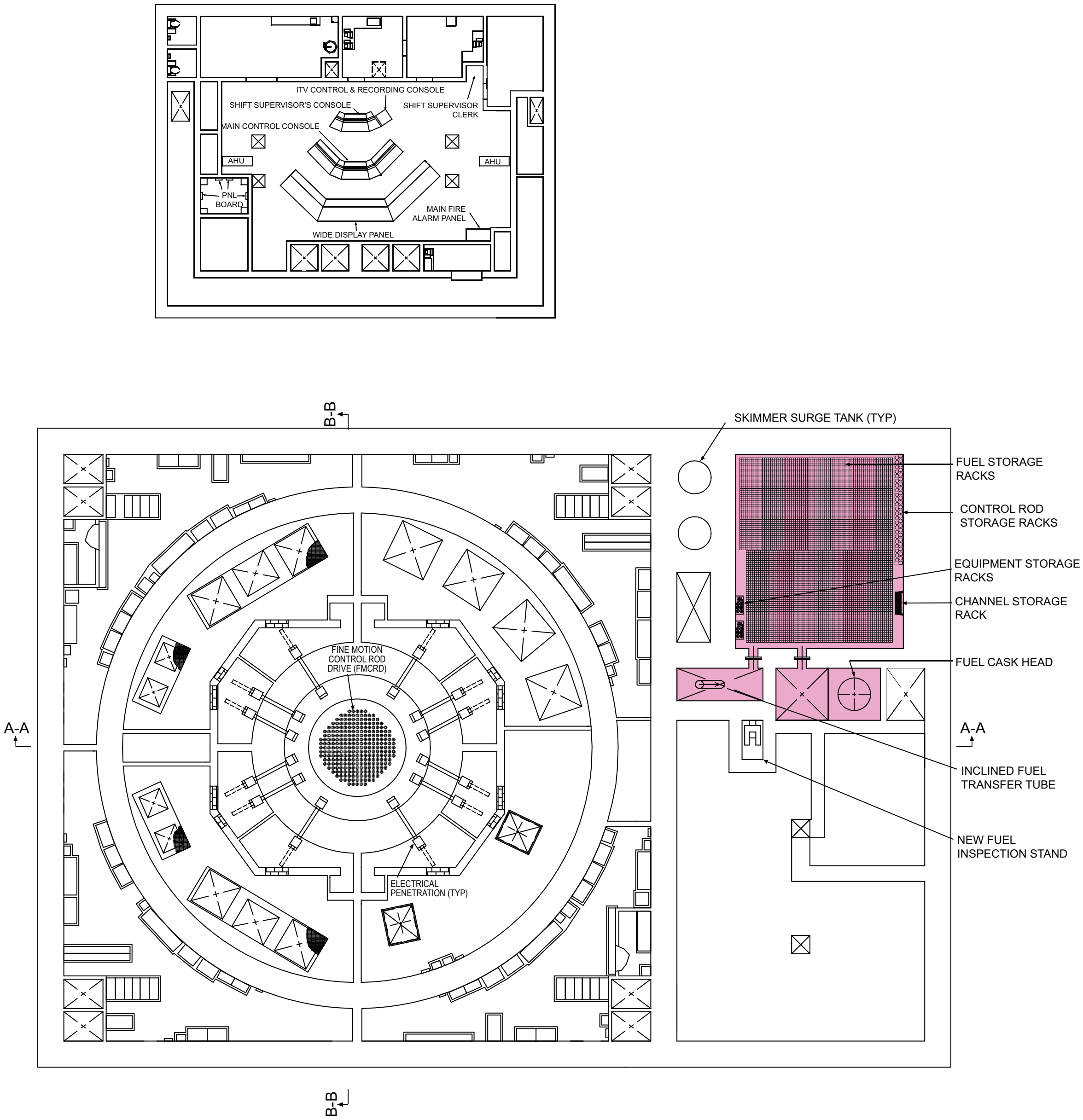

Figure 8-12. ESBWR Reactor, Fuel and Control Buildings MCR Level (B1F)

Removal of decay heat when the reactor is isolated from the main turbine is achieved by the Isolation Condenser System (see Chapter 3). Removal of the post-LOCA decay heat is achieved by the Passive Containment Cooling System (see Chapter 4). The large volume of water in the suppression pool serves as a fission-product scrubbing and retention mechanism. The Reactor Building serves as an additional barrier between the primary containment and the environment. Any fission-product leakage from the primary containment is expected to be contained within the Reactor Building

Analyses of the radiological dose consequences for design basis accidents, based on an assumed containment leak rate of 0.5% per day, show that the offsite doses after an accident are about 15 Rem TEDE for the standard U.S. site (see Chapter 11).

Key distinguishing features of the ESBWR Reactor-Fuel Building design include:

- Elimination of the recirculation system, which reduces the containment volume associated with high construction costs

- Reduced building volume which reduces material costs and construction schedule.
- Design with conventional structural shapes to improve constructability, which reduces capital costs and construction schedule.
- Improved personnel and equipment access for enhanced operability and maintainability.

The Reactor Building layout utilizes the grade-level entry area for major servicing of the cooling equipment. All of the major pieces of equipment can be moved into the reactor building area through hatches (not shown on the figures).

The volume of the ESBWR Safety Buildings is reduced to approximately 210,000 m3, an 18% reduction compared to ABWR. Since this reduced volume was obtained by simplification of the reactor supporting systems and optimization of their arrangement with improved access (rather than simply by compaction), it provides attractive material cost savings over previous BWRs and helps reduce the construction schedule without adversely impacting maintenance.

Inclined Fuel Transfer System

The Reactor and Fuel Buildings share the fuel storage requirements. There is a buffer pool in the Reactor Building with sufficient storage for 60% of a full-core load of fresh fuel plus 154 spent fuel assemblies (in the deep pit section of the pool). The spent fuel storage in the Fuel Building is sufficient for the spent fuel from 10 calendar years of operation plus a full-core offload. All fuel pools are lined with stainless steel.

The ESBWR is equipped with a non-safety-grade, but Seismic Category I, Inclined Fuel Transfer System (IFTS). In general, the arrangement of the IFTS (refer to Figure 8-13) consists of a terminus at the upper end in the Reactor Building buffer pool that allows the fuel to be placed in a fuel transport carriage and tilted from a vertical position to an inclined position prior to transport to the Spent Fuel Pool. There is a means to lower the fuel transport carriage, means to seal off the top end of the transfer tube, and a control system to effect transfer. It has a lower terminus in the fuel building storage pool and a means to tilt the fuel transport carriage into a vertical position allowing it to be removed from the transport device. There are controls contained in local control panels to monitor the transfer, opening and closing valves, and raising or lowering of the fuel transport carriage. There is a means to seal off the upper and lower end of the tube while allowing filling and venting of the tube.

There is sufficient redundancy and diversity in equipment and controls to prevent loss of load (carriage with fuel released in an uncontrolled manner) and there are no modes of operation that allow simultaneous opening of any set of valves that could cause draining of water from the upper pool in an uncontrolled manner. The carriage and valves may be manually operated in the event of a power failure to allow completion of the fuel transfer process.

The IFTS terminates in a separate pit in the fuel storage pool. The lower terminus of the IFTS allows for thermal expansion (axial movement relative to the anchor point in the Reactor Building). The lower terminus allows for differential movement

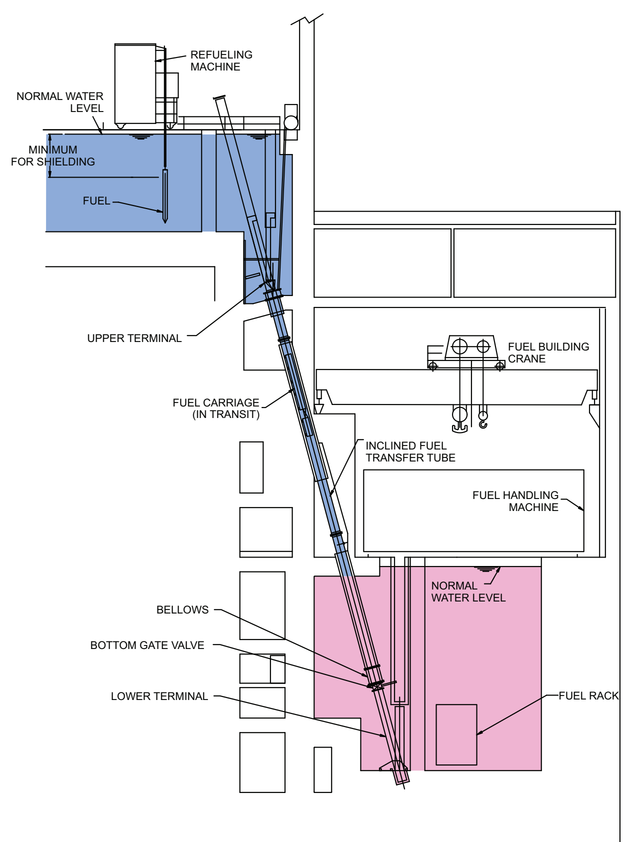

Figure 8-13. ESBWR Inclined Fuel Transfer System

between the anchor point in the Reactor Building and the fuel pool terminus, and allows it to have rotational movement at the end of the tube relative to the anchor point in the Reactor Building. The lower end interfaces with the fuel storage pool with a bellows to seal between the transfer tube and the Spent Fuel Pool wall.

The IFTS carriage primarily handles nuclear fuel using a removable insert and control blades in a separate insert in the transfer cart. Other contaminated items may be moved in the carriage utilizing a suitable insert.

For radiation protection, personnel access into areas of high radiation or areas immediately adjacent to the IFTS is controlled. Access to any area adjacent to the transfer tube is controlled through a system of physical controls, interlocks and an annunciator.

The IFTS has sufficient cooling such that a freshly removed fuel assembly can remain in the IFTS until it is removed without damage to the fuel or excessive overheating.

Primary Containment System

The ESBWR containment is a low-leakage reinforced concrete structure with an internal steel liner in the drywell and suppression chamber to serve as a leak-tight membrane. The containment is a cylindrical shell structure, which consists of the reactor pressure vessel (RPV) pedestal, the containment cylindrical wall, the top slab, the suppression pool slab, and the foundation mat. The containment is divided by the diaphragm floor and the vent wall into a drywell chamber (DW) and a suppression chamber, or wetwell (WW). The top slab of the containment is an integral part of the Isolation Condenser/Passive Containment Cooling (IC/PCC) pools and the services pools for storage of Dryer/Separator and other uses. The pool girders, which serve as barriers for the pools, rigidly connect the top slab and the Reactor Building (RB) walls. The RB floors that surround the containment walls and walls that are under the suppression pool floor slab are also integrated structurally with the concrete containment. The containment foundation mat is continuous with the RB foundation mat and the Fuel Building (FB) as well. The containment and the structures integrated with the containment are constructed of cast-in-place, reinforced concrete. The containment system is designed to have the following functional capabilities (Figure 8-14):

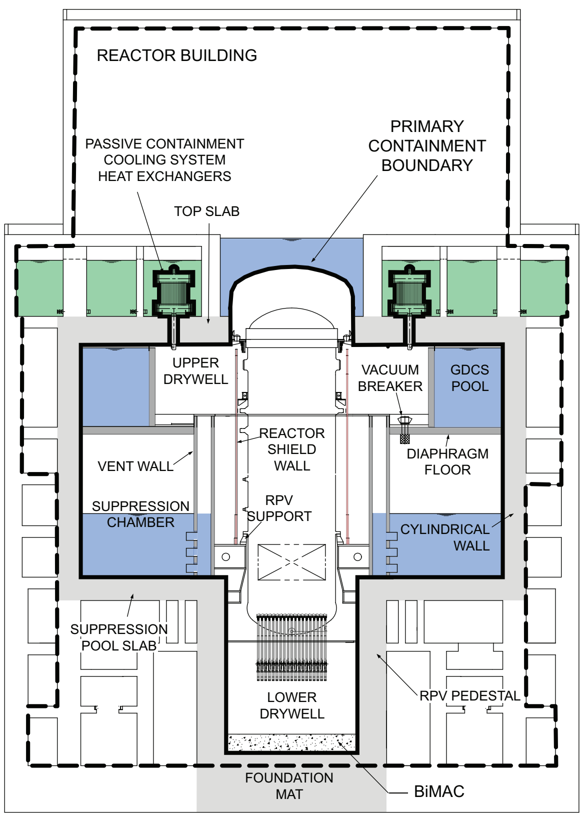

Figure 8-14. ESBWR Reactor Building and Containment

- The containment structure has the capability to maintain its functional integrity during and following the peak transient pressures and temperatures which would occur following any postulated loss-of-coolant accident (LOCA). The containment structure is designed for the full range of loading conditions consistent with normal plant operating and accident conditions, including LOCA-related loads in and above the suppression pool (SP) together with a concurrent safe shutdown earthquake (SSE)
- The containment structure is designed to accommodate the maximum internal negative pressure difference between DW and WW, and the maximum external negative pressure difference relative to the reactor building surrounding the containment
- The containment has capability for rapid closure or isolation of all pipes and ducts that penetrate the containment boundary in order

to maintain leak tightness within acceptable limits

- The containment structure and isolation, with concurrent operation of other accident mitigation systems, is designed to limit fission-product leakage during and following the postulated Design Basis Accident (DBA) to values less than leakage rates that could result in offsite radiation doses greater than those set forth in 10CFR52.47
- The containment structure is designed to accommodate flooding to a sufficient depth above the active fuel to permit safe removal of the fuel assemblies from the reactor core after the postulated DBA
- The containment structure design provides means to channel the flow from postulated pipe ruptures in the DW to the suppression pool

Drywell Structure

The drywell (DW) is comprised of two volumes:

- An upper drywell (UDW) volume surrounding the upper portion of the RPV and housing the main steam and feedwater piping, Gravity Driven Cooling System (GDCS) pools and piping, PCCS piping, Isolation Condenser System (ICS) piping, SRVs and piping, depressurization valves (DPVs) and piping, DW coolers and piping, the reactor shield wall, RPV support brackets, and other miscellaneous systems
- A lower drywell (LDW) volume below the RPV support system housing the lower portion of the RPV, fine motion control rod drives, other miscellaneous systems, and equipment below the RPV, and vessel bottom drain piping

The UDW is a cylindrical, reinforced-concrete structure with a removable steel head and a diaphragm floor constructed of steel girders with concrete fill. The RPV sliding block support system supports the RPV while accommodating relative thermal expansion and has openings for communication between the UDW and LDW. There are eight sliding block supports. One end of each sliding support is fastened to a circumferential RPV flange segment that is forged integral to the vessel shell ring at that RPV elevation. The other end of each sliding block is restrained by sets of steel guide blocks that are attached to the reactor pedestal support brackets. Under this configuration, each sliding support is relatively free to expand in the radial direction but is restrained in the vertical and vessel tangential directions.

Penetrations through the liner for the DW head, equipment hatches, personnel locks, piping, electrical and instrumentation lines are provided with seals and leak-tight connections.

Wetwell Structure

The wetwell (WW) is comprised of a gas volume and suppression pool filled with water to rapidly condense steam from a reactor vessel blowdown via the SRVs or from a break in a major pipe inside the drywell through the vent system. The WW is connected to the DW by a vent system comprised of twelve vertical/horizontal vent modules. Each module consists of a vertical flow steel pipe, with three horizontal vent pipes extending into the suppression pool water. Each vent module is built into the vent wall, which separates the DW from the WW. The cylindrical vent wall is supported off the RPV pedestal. The WW boundary is the annular region between the vent wall and the cylindrical containment wall and is bounded above by the DW diaphragm floor. All normally-wetted surfaces of the liner in the WW are stainless steel and the rest are carbon steel.

Containment Structure

The containment structure includes a steel liner to reduce fission-product leakage. All normally wetted surfaces of the liner in the suppression pool are made of stainless steel. Penetrations through the liner for the drywell head, equipment hatches, personnel locks, piping, and electrical and instrumentation lines are provided with seals and leak-tight connections. The allowable leakage is 0.35% per day from all sources, excluding main steam isolation valve (MSIV) leakage.

Containment System

The DW is designed to withstand the pressure and temperature transients associated with the rupture of any primary system pipe inside the DW, and also the negative differential pressures associated with containment depressurization events, when the steam in the DW is condensed by the PCCS, the

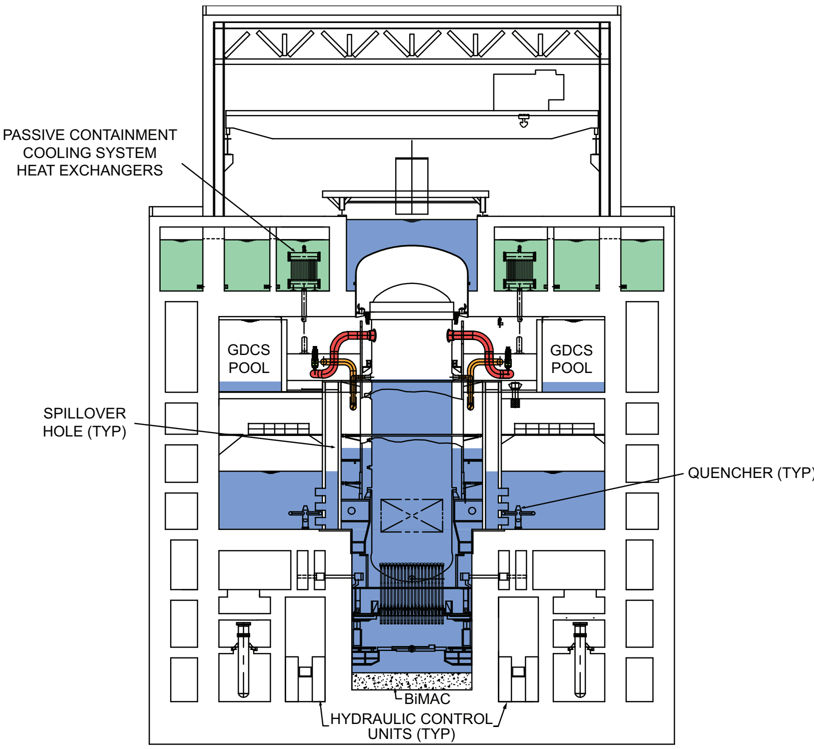

Figure 8-15. Water Levels After a FW Line Break

GDCS, the Fuel and Auxiliary Pools Cooling System (FAPCS), and cold water cascading from the break following post-LOCA flooding of the RPV.

In the event of a pipe break within the DW, the increased pressure inside the DW forces a mixture of noncondensible gases, steam, and water through either the PCCS or the vertical/horizontal vent pipes and into the suppression pool where the steam is rapidly condensed. The noncondensible gases transported with the steam and water are contained in the free gas space volume of the WW. The design pressure of the containment is 310 kPa(g) (44.1 psig).

There is sufficient water volume in the suppression pool to provide adequate submergence over the top of the upper row of horizontal vents, as well as the PCCS return vent, when the water level in RPV reaches one meter above the top of active fuel and water is removed from the pool during post-LOCA equalization of pressure between RPV and the WW. Water inventory, including the GDCS, is sufficient to flood the RPV to at least one meter above the top of active fuel.

To help manage the water within containment there are 12 spillover vents connecting the lower part of the UDW with the SP. The actual water levels in the RPV and containment compartments will depend on the LOCA break location. For example, Figure 8-16 shows the water levels in the GDCS, RPV, DW and SP in the long term after a FW pipe break LOCA, and Figure 8-16 shows the water levels after an RPV bottom drain line break LOCA.

In both of these examples, the equalizing lines in the GDCS did not open. If they did, the water level in the SP and RPV would be almost the same, and in the UDW the level would be at the spillover pipe level.

Control of potential hydrogen generation in an accident is handled by main-

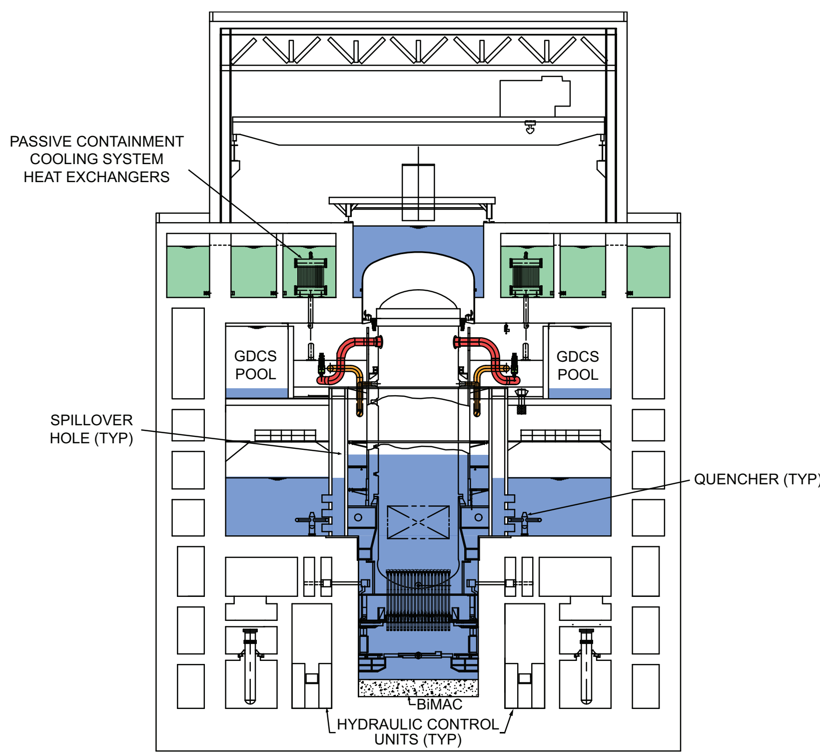

Figure 8-16. Water Levels After a Bottom Drain Line Break

taining an inerted containment atmosphere via the Containment Inerting System (CIS), which inerts the containment atmosphere with nitrogen to maintain < 3% oxygen- see Chapter 5. In addition to inerting containment, the Passive Autocatalytic Recombiner System, a nonsafety-related system, is provided as defense-in-depth against the potential buildup of combustible gases generated by the radiolytic decomposition of water post-LOCA.

Containment Heat Removal

The containment design includes a Drywell Cooling System (DCS) to maintain DW temperatures during normal operation within acceptable limits for equipment operation (see Chapter 5).

Isolation transients do not present a heat removal challenge to the ESBWR containment, due to the use of the Isolation Condenser System (ICS). See Chapter 3 for more details.

A safety-related PCCS is incorporated into the design of the containment to remove decay heat from the DW following a LOCA. The PCCS uses six elevated heat exchangers (condensers) located outside the containment in large pools of water at atmospheric pressure to condense steam that has been released to the DW following a LOCA. This steam is channeled to each of the condenser tube-side heat transfer surfaces where it condenses and the condensate returns by gravity flow to the GDCS pools. Noncondensible gases are purged to the suppression pool via vent lines. The PCCS condensers are an extension of the containment boundary, do not have isolation valves and start operating immediately following a LOCA. These low pressure PCCS condensers provide a thermally-efficient heat removal mechanism. No forced circulation equipment is required for operation of the PCCS. Steam produced, due to boil-off in the pools surrounding the PCCS condensers, is vented to the atmosphere. There is sufficient inventory in these pools to handle at least 72 hours of decay heat removal. The PCCS is described in more detail in Chapter 4.

After an accident, the non-safety related Fuel and Auxiliary Pools Cooling System (FAPCS) may be available in the suppression pool cooling mode and/or containment spray mode to control the containment pressure and temperature conditions. Heat is removed via the FAPCS heat exchanger(s) to the Reactor Component Cooling Water System (RCCWS) and finally to the Plant Service Water System (PSWS). While this is the preferred method to provide long-term containment cooling and cold shutdown after a LOCA, Train A of the RWCU/SDC system can be cross-connected to the FAPCS suppression pool suction and the FAPCS containment cooling line to provide containment cooling in the unlikely event there has been a fuel failure. This will provide containment cooling while maintaining contaminated water inside the reactor building. These systems are described in Chapter 5.

Vacuum Breakers

A vacuum breaker system, consisting of 3 vacuum breakers, has been provided between the DW and WW. The purpose of the DW-to-WW vacuum breaker system is to protect the integrity of the diaphragm floor slab and vent wall between the DW and the WW, and the DW structure and liner, and to prevent backflooding of the suppression pool water into the DW. Refer to Figure 8-17.

Each vacuum breaker is designed for high reliability, leak-tightness, stability (i.e., elimination of chatter), and resistance to debris. It operates passively in response to a negative WW-to-DW pressure gradient and is otherwise held closed by a combination of gravity and the normally positive WW-to-DW pressure gradient. A vertical-lift poppet disk with two bearings to maintain alignment constitutes the only moving part. The valve assembly is equipped with inlet and outlet screens to prevent debris entry. A leak-tight design is achieved by use of a non-metallic main seat and a backup hard seat. The seats are designed such that the lodging of a particle of the maximum size which can pass through the inlet/outlet screens on either seat will not prevent sealing of the valve. An anti-chatter ring around the periphery of the disk reduces seat-to-disk impact force and provides damping by energy absorption.

Each vacuum breaker is provided with redundant proximity sensors to detect its closed position. On the upstream side of the vacuum breaker, pneumatically-operated, fail-as-is safety-related isolation valves are provided to isolate a leaking or stuck-open vacuum breaker. During a LOCA, when the vacuum breaker opens to equalize the DW and WW pressure

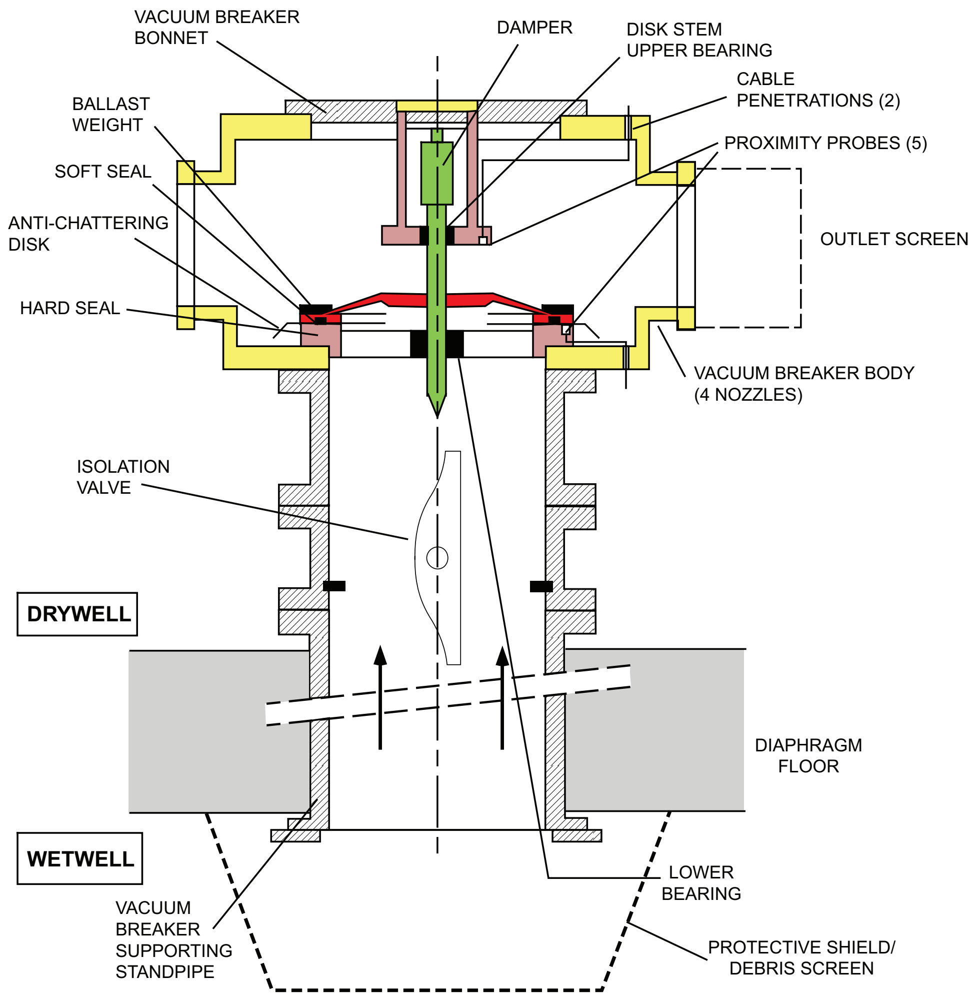

Figure 8-17 ESBWR Wetwell-to-Drywell Vacuum Breaker with Backup Closure Valve

and subsequently does not completely close as detected by the proximity sensors, a control signal will close the upstream isolation valve to prevent extra bypass leakage due to the opening created by the vacuum breaker and therefore maintain the pressure suppression capability of the containment.

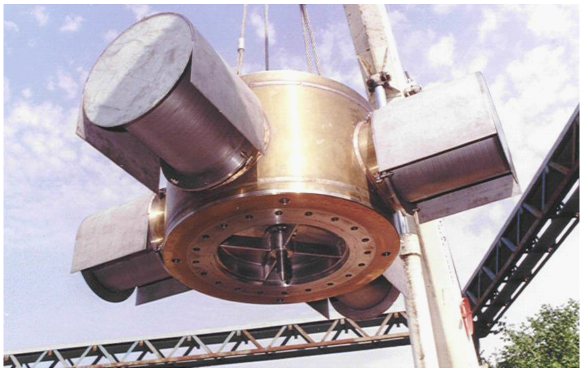

Figure 8-18. Prototype Vacuum Breaker

In addition to the proximity sensors, there are temperature sensors located on and in the vacuum breaker/vacuum breaker isolation valve assembly. These temperature sensors will also detect bypass leakage above a prescribed limit and close a vacuum breaker isolation value. Plant operators can also manually close the backup valve. Redundant vacuum breaker systems are provided to protect against a single failure.

The ESBWR vacuum breaker has undergone engineering development testing using a full-scale prototype to demonstrate the proper operability, reliability, and leak-tightness of the design. Figure 8-19 shows one of the modules tested.

Severe Accident Mitigation

The ESBWR design uses a passively-cooled boundary that is impenetrable by the core debris in whatever configuration it could possibly exist on the

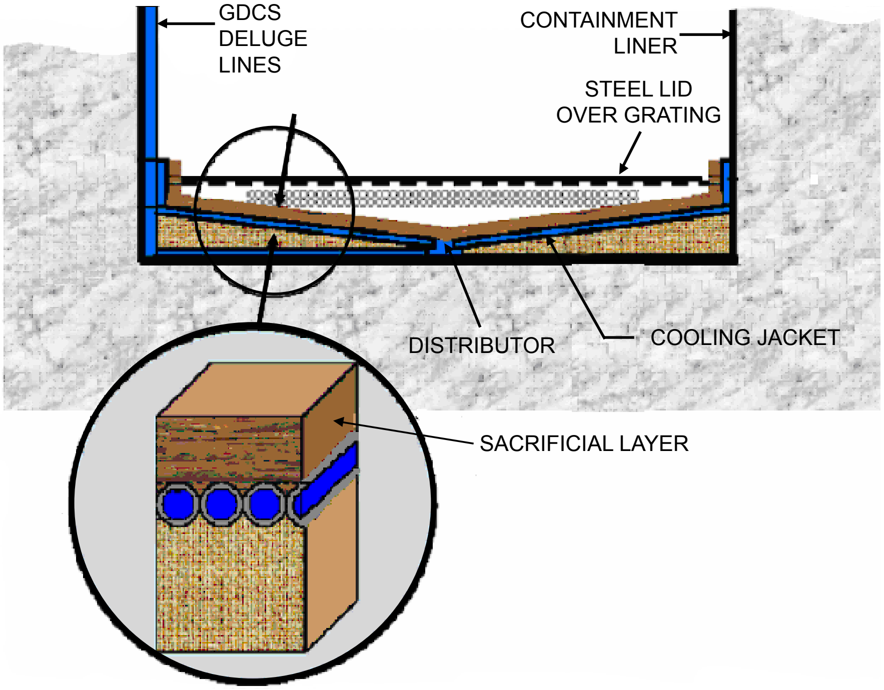

Figure 8-19. BiMAC Concept

LDW floor in severe accident scenarios. For ex-vessel implementation, this boundary is conveniently and advantageously made by a series of side-by-side placed inclined pipes, forming a jacket which can be effectively and passively cooled by natural circulation when subjected to thermal loading on any portion(s) of it. Water is supplied to this device from the GDCS pools via a set of temperature actuated squib-valve-activated deluge lines. The timing and flows are such that (a) cooling becomes available immediately upon actuation, and (b) the chance of flooding the LDW prematurely, to the extent that opens up a vulnerability to steam explosions, is very remote. The jacket is buried inside the concrete basemat and would be called into action only in the event that some or all of the core debris on top is non-coolable.

The device, called Basemat Internal Melt Arrest and Coolability device (BiMAC), is illustrated in Figure 8-19. Important considerations in implementation of this concept are as follows:

- Pipe inclination angle. The inclined pipes are designed with consideration of critical heat fluxes generated by the molten corium, to permit natural circulation flow
- Protective layer. A refractory material is laid on top of the BiMAC pipes so as to protect against melt impingement during the initial corium relocation event, and to allow some adequately short time for diagnosing that conditions are appropriate for flooding. This is to minimize the chance of inadvertent, early flooding. The material is selected to have high structural integrity and high resistance to melting
- Cover plate. As shown in Figure 8-19, a supported steel plate covers the BiMAC. This allows that the top is a normal floor as needed for operations and that the BiMAC is basically out of the way until its function is ever needed. The plate is made to sit on top of normal floor grating, which itself is supported by steel columns. The cover plate is designed so that debris will penetrate it in a short period of time while providing protection for the BiMAC from CRD housing falling from the vessel
- Lower Drywell cavity. The space available below the BiMAC plate is sufficient to accommodate the full-core debris, and the entire coolable volume, up to the height of the vertical segments of the BiMAC pipes is ~400% of the full-core debris. Thus, there is no possibility for the melt to remain in contact with the LDW liner. Similarly, the two sumps needed for detecting leakage flow during normal operation are positioned and protected, as is the rest of the LDW liner, from being subject to melt attack
- The LDW deluge system interface. This system consists of lines that feed off the independent GDCS pools, respectively, each separating into a pair of lines that connect to the BiMAC main header

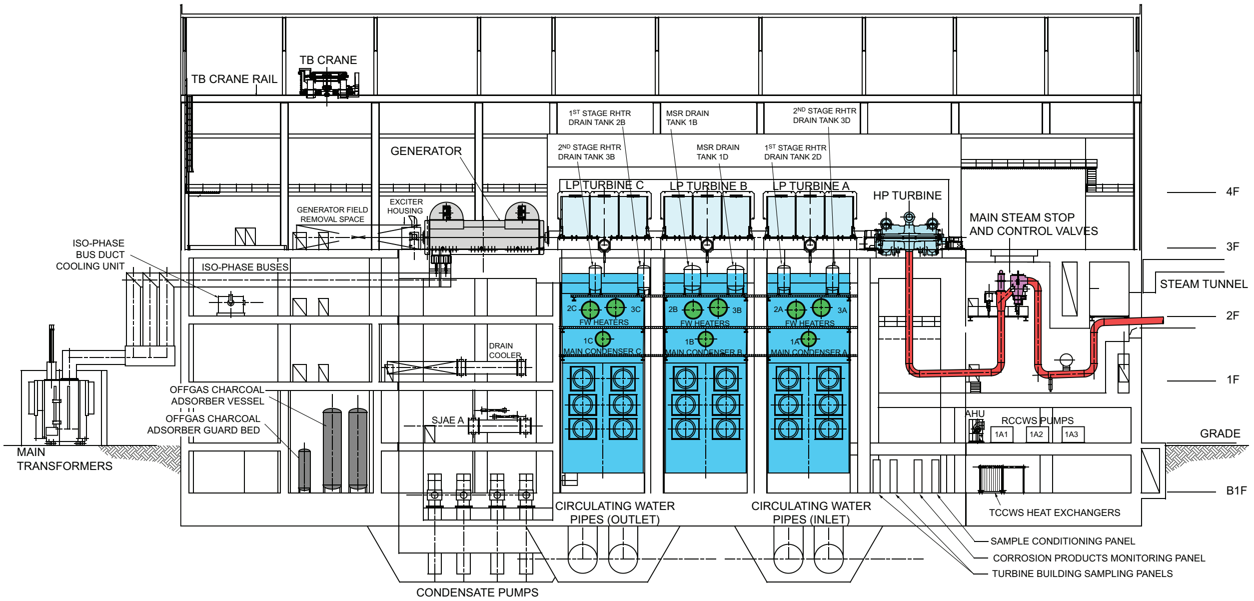

Figure 8-20. ESBWR Turbine Building Section BB

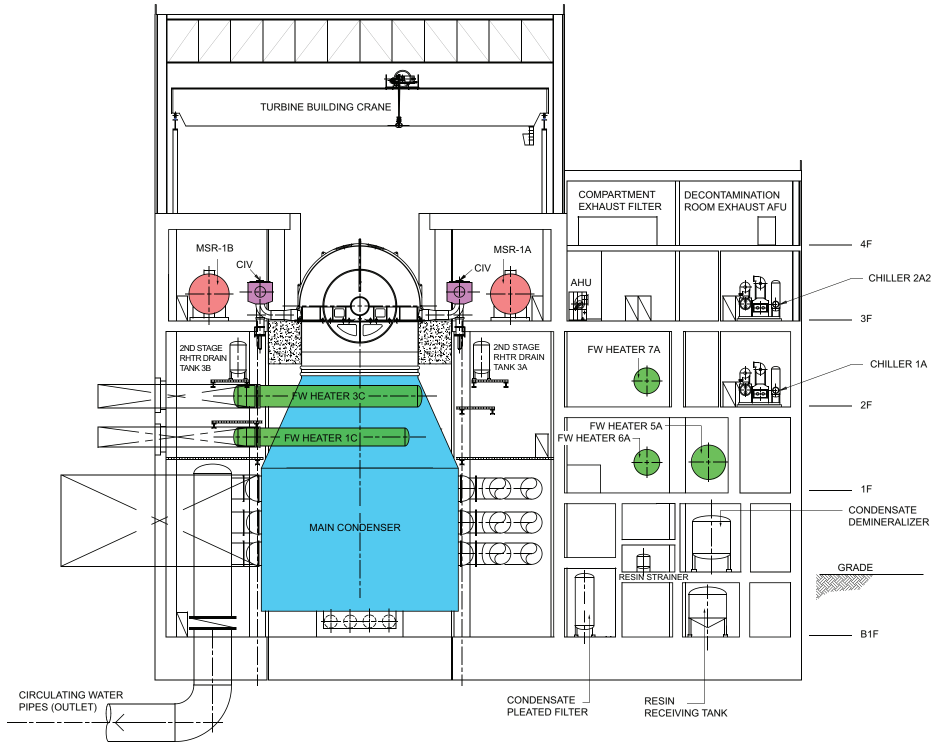

Figure 8-21. ESBWR Turbine Building Section AA

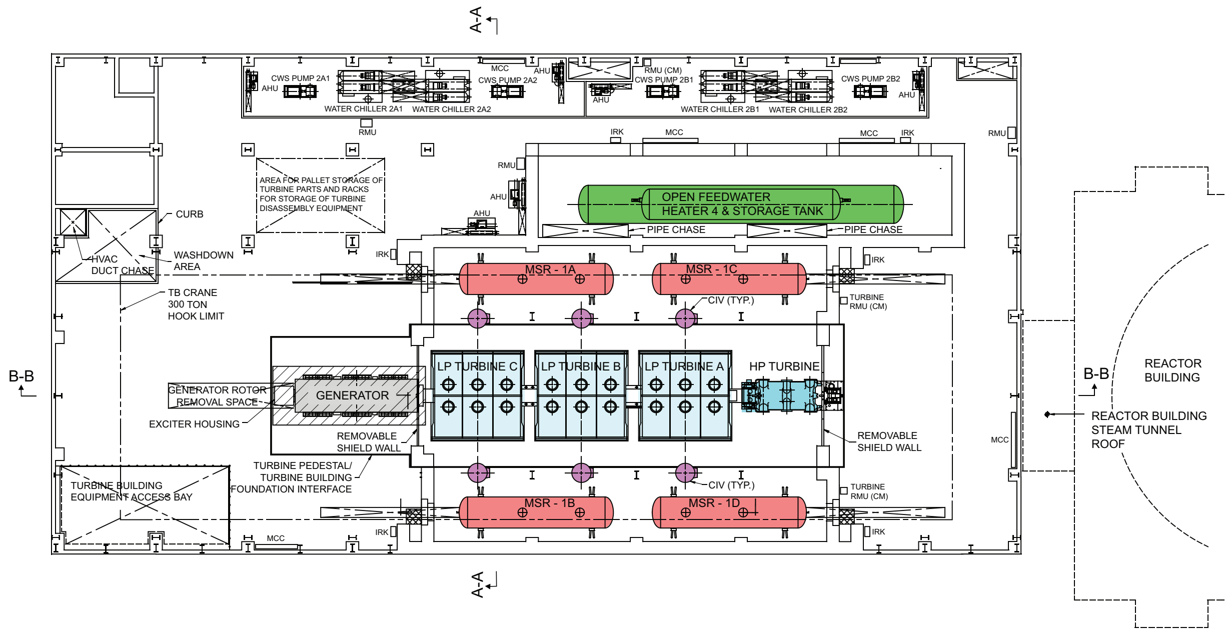

Figure 8-22. ESBWR Turbine Building Operating Floor

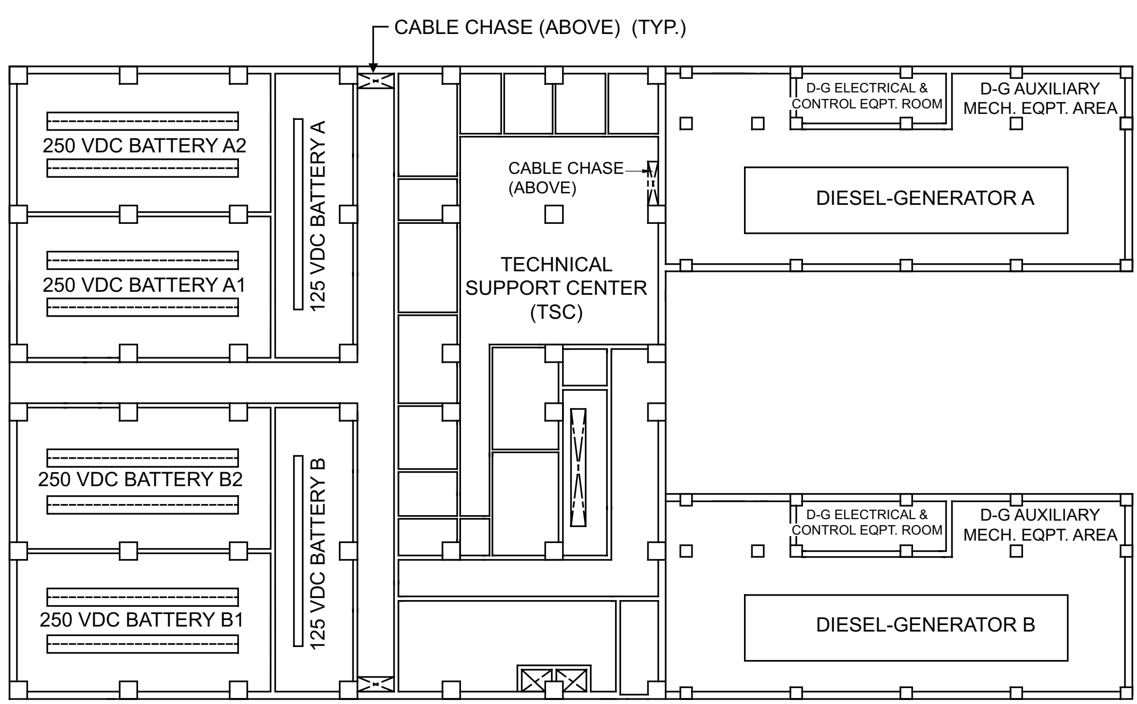

Figure 8-23. ESBWR Electrical Building Grade Level

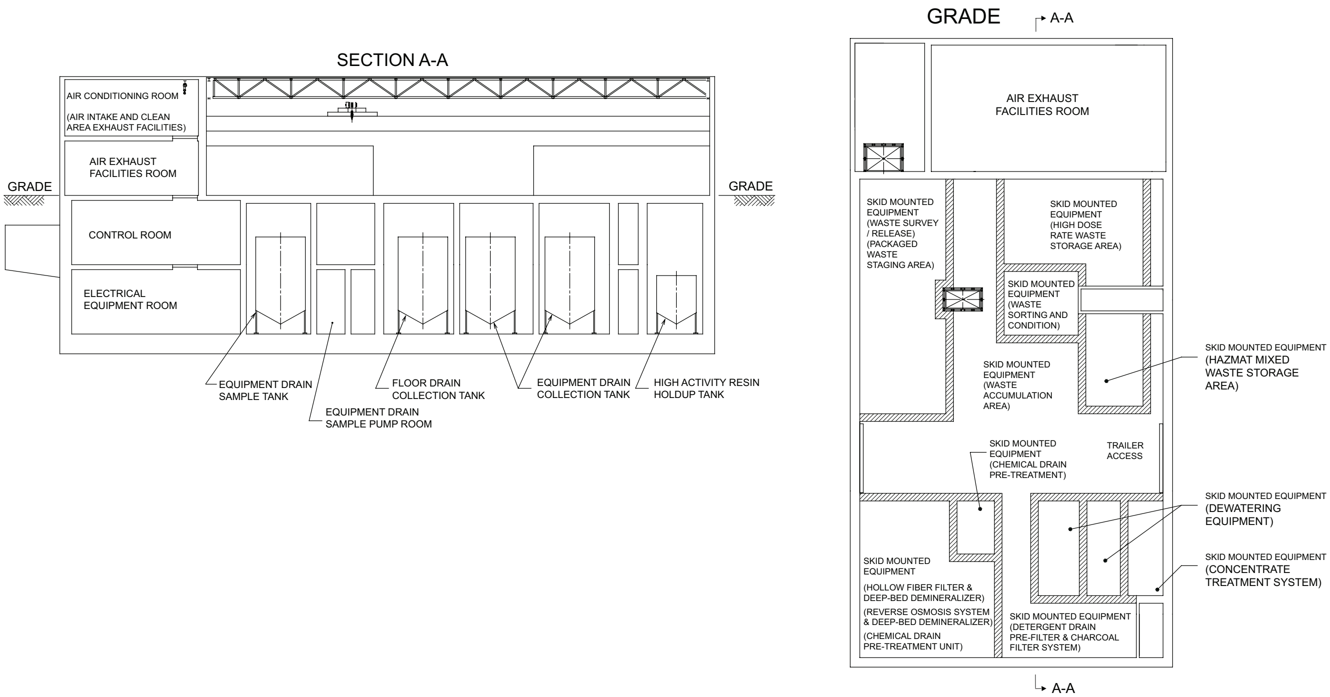

Figure 8-24 ESBWR Radwaste Building Section and Grade Elevation

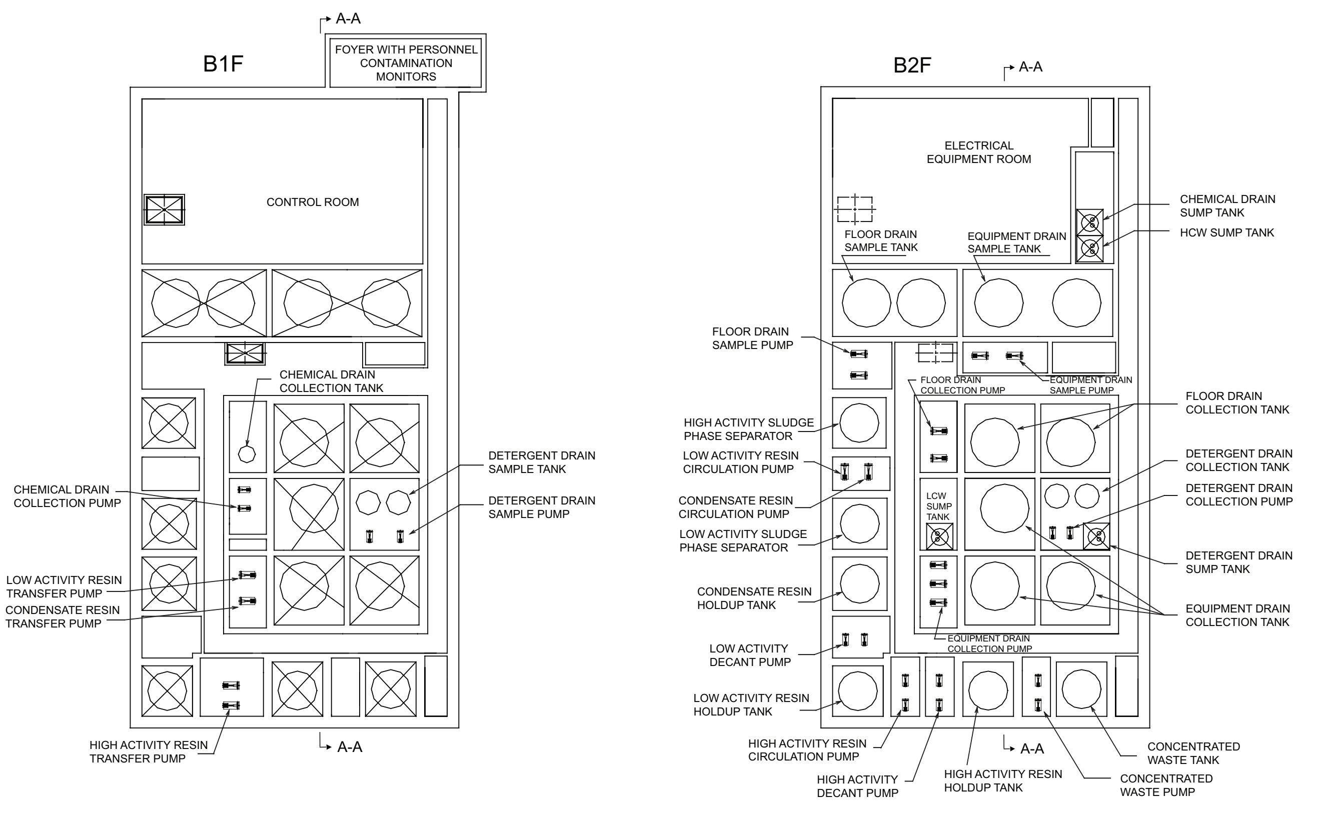

Figure 8-25. ESBWR Radwaste Building Basement Levels

Turbine Building

The Turbine Building (TB) houses all the components of the power conversion system. This includes the turbine-generator, main condenser, air ejector, steam packing exhauster, offgas condenser, main steam system, turbine bypass system, condensate demineralizers, and the condensate and feedwater pumping and heating equipment. It also includes the Chilled Water Systems, the Reactor Component Cooling Water System (RCCWS), and the Turbine Component Cooling Water System (TCCWS). The small size of the ESBWR Turbine Building makes a significant contribution to capital cost savings and a shorter construction schedule.

The TB is a Seismic Category II, non-safety building. Figures 8-20 and 8-21 show sectional views. Figure 8-22 shows a plan view of the operating floor.

Electrical Building

The Electrical Building houses the two nonsafety-related standby diesel generators and their associated auxiliary equipment, as well as the nonsafety-grade batteries. It also houses the Technical Support Center. Figure 8-23 shows the grade level floor layout. The building is nonsafety-related and Seismic Category NS.

Radwaste Building

The Radwaste Building houses all equipment associated with the collection and processing of the liquid and solid radioactive waste generated by the plant. The Offgas System components are located in the Turbine Building (see Chapter 10 for system descriptions). Figures 8-24 and 8-25 show the general arrangements and access for radwaste processing technologies.

Other Principal Buildings

Other buildings on the site include the Service Water Building, Service Building, Water Treatment Building, Administration Building, Training Center, Sewage Treatment Plant, warehouse, hot and cold machine shops, the intake structure, heat sink (cooling towers) and yard facilities for electrical equipment.

Fire Protection

The basic layout of the plant and the choice of systems to mitigate the effects of fire enhance the resistance of the ESBWR plant to fire. The safety-related systems are designed such that there are four independent divisions. In addition, there are nonsafety-related systems, such as the RWCU/SDC, which can be used to achieve safe shutdown. The plant arrangement is such that points of possible common cause failure between nonsafety-related systems and safety-related systems have been eliminated.

Plant Arrangement

The plant is laid out in such a way that power and control signals from the Reactor and Turbine Buildings are routed directly to the Control and Electrical Buildings. This arrangement ensures that a potentially damaging fire in the Turbine Building will not disable nonsafety-related systems capable of providing safe shutdown in both the Turbine and Reactor Buildings.

Divisional Separation

There are four complete divisions of passive cooling systems. In general, systems are grouped by safety division so that in case of fire only one division is affected. Complete burnout of any fire area does not prevent safe shutdown of the plant, since there will always be three other divisions available.

Redundant remote shutdown panels, located in two separate, independent divisions, provide redundant control of the safe shutdown function from outside the control room in case the control room becomes uninhabitable.

Fire Containment

Fire containment is achieved through the use of concrete fire barrier floors, ceilings and walls designed to contain a fire for a duration of three hours without structural failure. Fire dampers are required for any HVAC duct penetrating a fire barrier and they have a rating of three hours. Electrical and piping penetrations through a fire barrier have seals with a three-hour rating.

There are four firewater pumps in the plant, two motor-driven and two diesel-driven. Each of these meets requirements for flow and pressure demand at the most hydraulically remote hose connection in the plant. Fire water supply piping and systems in the Reactor, Control, and Fuel Buildings are designed to remain functional following an SSE.

Flood Protection

The ESBWR design incorporates measures for flooding protection of safety-related structures, systems, and components from both external flooding and flooding from plant component failures.

Flood Protection from External Sources

Seismic Category I structures remain protected for safe shutdown of the reactor during all external flood conditions. The safety-related systems and components are flood-protected either because they are located above the design flood level or are enclosed in reinforced concrete Seismic Category I structures. These structures have features for flood protection, including minimum thickness for walls below flood level, water stops in construction joints, waterproof coating on external surfaces, roof design to prevent pooling of large quantities of water, and penetrations and access openings below grade are watertight.

Flood Protection from Internal Component Failures

All piping, vessels, and heat exchangers with flooding potential in the Reactor Building are seismically qualified.

Water spray, foaming, and flooding effects in a room with a pipe crack or break are conservatively assumed in the safety analysis to take any safe-shutdown equipment in the room out of service. The following provisions have been made to limit the flooding effects to one safety division:

- Watertight doors and sealed penetrations to prevent water seepage or flow
- Fire doors designed to hold back water pressure, which also prevent spray from crossing divisional boundaries
- Floors, floor penetrations and equipment hatches designed to prevent water seepage to lower elevations through the use of seals and curbs and routing of drain lines
- Water-sensitive, safety-related equipment raised on pads above the floor elevation for protection against expected seepage under non-watertight doors

HITACHI

Chapter 9 Major Balance of Plant Features

It is difficult to completely standardize the plant design beyond the Nuclear Island. In addition to utility preferences in the steam and power conversion system, there are also site-unique issues, such as the Normal Power Heat Sink (NPHS) location and temperature, and the offsite power distribution system, which can play a significant role in the selected configuration. What follows, therefore, is an example configuration, showing one possible implementation. Changes in this part of the plant will not have any significant impact on the Nuclear Island design or operation.

Steam and Power Conversion System

The Turbine Building houses all equipment associated with the main turbine generator and other auxiliary equipment. The turbine employs a conventional regenerative cycle with condenser deaeration and condensate demineralization. The turbine-generator is equipped with an electrohydraulic control system and supervisory instruments to monitor performance. The gross electrical output of the turbine-generator is approximately 1,600 MWe.

The components of the Steam and Power Conversion (S&PC) System are designed to produce electrical power utilizing the steam generated by the reactor, condense the steam into water, and return the water to the reactor as heated feedwater, with a major portion of its gaseous, dissolved, and particulate impurities removed in order to satisfy the reactor water quality requirements.

The S&PC System includes the turbine portion of the main steam system, the main turbine-generator, main condenser, condenser evacuation system, turbine gland seal system, turbine bypass system, extraction steam system, condensate purification system, and the condensate and feedwater pumping and heating system. The heat rejected to the main condenser is removed by a circulating water system and discharged to the power cycle heat sink.

Steam generated in the reactor is supplied to the high-pressure turbine and the steam reheaters. Steam leaving the high-pressure turbine passes through a combined moisture separator/reheater prior to entering the low-pressure turbines. The moisture separator drains, steam reheater drains, and the drains from the two high-pressure feedwater heaters are returned to a direct contact feedwater heater (deaerator). The low-pressure feedwater heater drains are cascaded to the condenser.

Steam exhausted from the low-pressure turbines is condensed and deaerated in the condenser. The condensate pumps take suction from the condenser hotwell and deliver the condensate through the filters and demineralizers, gland steam condenser, Steam Jet Air Ejector (SJAE) condensers, and offgas recombiner condensers to the direct contact feedwater heater where it is mixed with turbine extract steam and high-pressure feedwater heater and MSR drains. The feedwater booster pumps and feedwater pumps take suction from the direct contact feedwater heater and discharge through the high-pressure feedwater heaters to the reactor.

The S&PC System main conceptual features are illustrated on Figure 9-1, assuming a triple pressure condenser. This type of condenser and other site dependent ESBWR plant features and parameters

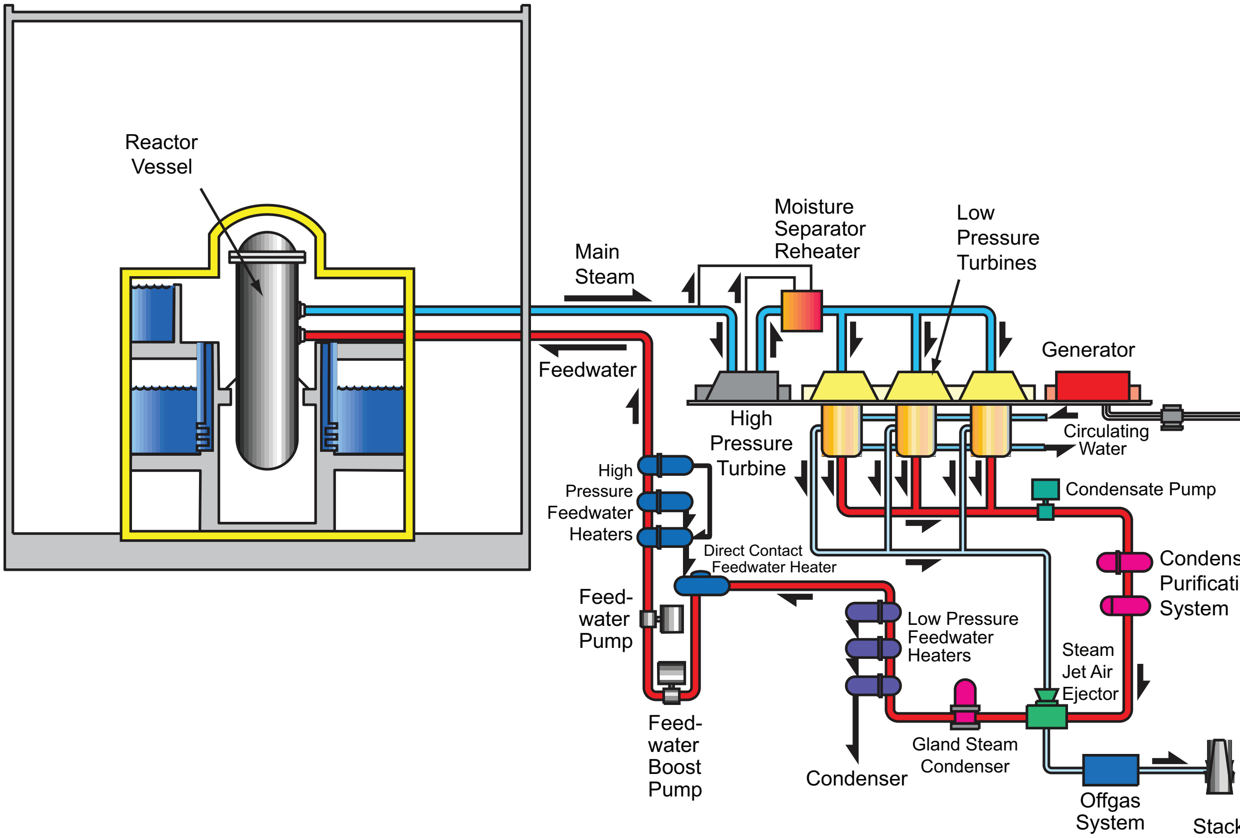

Figure 9-1. ESBWR Steam and Power Conversion System

are reported herein based on typical central U.S. site conditions.

Normally, the turbine power heat cycle utilizes all the steam being generated by the reactor; however, an automatic pressure-controlled turbine bypass system designed for 110% of the rated steam flow is provided to discharge excess steam directly to the condenser. This allows a loss of full load with the ability to drop to house load without a turbine overspeed trip or a reactor scram.

Turbine Main Steam Systems

The Turbine Main Steam System (TMSS) delivers steam from the reactor to the turbine generator, the reheaters, the turbine bypass system, and the SJAEs from warmup to full-load operation. The Main Steam System also supplies the steam seal system and the auxiliary steam system when other sources are not available.

The main steam piping consists of four lines from the seismic interface restrain to the main turbine stop valves. The four main steam lines are connected to a header upstream of the turbine stop valves to permit testing of the MSIVs during plant operation with a minimum load reduction. This header arrangement is also provided to ensure that the turbine bypass and other main steam supplies are connected to operating steam lines and not to idle lines.

A drain line is connected to the low points of each main steam line, both inside and outside the containment. Both sets of drains are headered and connected with isolation valves to allow drainage to the main condenser. To permit intermittent draining of the steam line low points at low loads, orificed lines are provided around the final valve to the main condenser. The steam line drains, maintain a continuous downward slope from the steam system low points to the orifice located near the condenser. The drain line, from the orifice to the condenser, also slopes downward. To permit emptying the drain lines for maintenance, drains are provided from the line low points going to the radwaste system sumps.

The drains from the steam lines inside containment are connected to the steam lines outside the containment to permit equalizing the pressure across the MSIVs during startup and following steam line isolation.

Main Turbine-Generator and Moisture Separator/Reheaters

The turbine-generator (TG) consists of an 1,800-rpm turbine, generator, exciter, controls and associated subsystems.

The turbine for the ESBWR reference plant consists of a double-flow, high-pressure unit, and three double-flow low-pressure units in tandem. The high-pressure turbine has two stages of steam extraction, which is directed to high-pressure feedwater heaters and Moisture Separator/Reheater (MSR)

Moisture separation and reheating of the high-pressure turbine exhaust steam is performed by four MSRs installed in the steam path between the high- and low-pressure turbines. The MSRs are located on each side of the TG centerline. The MSRs serve to dry and reheat the high-pressure turbine steam exhaust before it enters the low-pressure turbines. This improves cycle efficiency and reduces moisture-related erosion and corrosion in the low pressure turbines. Moisture is removed in chevron-type moisture separators and is drained to the moisture separator drain tank and from there to the direct contact feedwater heater. The dry steam passes upward across the heater, which is supplied with both main, and extraction steam. Finally, the reheated steam is routed to the combined intermediate valves which are located upstream of the low-pressure turbine inlet nozzles.

The steam passes through the low-pressure turbines, each with an extraction point for the direct contact feedwater heater, as well as three extraction points for the three low-pressure stages of feedwater heating, and exhausts into the main condenser. In addition to the external MSRs, the turbine blades are designed to separate water from the steam and drain it to the next lowest extraction point feedwater heater.

The generator is a direct driven, three-phase, 60-Hz, 1,800 rpm synchronous generator with a water-cooled stator and hydrogen-cooled rotor.

The turbine-generator uses a digital monitoring and control system, which, in coordination with the turbine steam bypass and Pressure Control System, controls the turbine speed, load, and flow for startup and normal operations. The control system operates the turbine stop valves, control valves, and combined intermediate valves (CIVs). TG supervisory instrumentation is provided for operational analysis and malfunction diagnosis.

TG accessories include the bearing lubrication oil system, turbine control system (TCS), turning gear, hydrogen and CO2 system, seal oil system, stator cooling water system, exhaust hood spray system, turbine gland sealing system and turbine supervisory instrument system.

The TG unit and associated piping, valves, and controls are located completely within the Turbine Building. Any local failure associated with the TG unit will not affect any safety-related equipment. Failure of TG equipment cannot preclude safe shutdown of the reactor system.

The gross electrical output of the turbine generator is approximately 1,600 MWe. For utilities generating 50-Hz power, the turbine shaft speed is 1,500 rpm.

Main Condenser

The main condenser for the ESBWR reference plant design is a multi-pressure, three-shell, reheating/deaerating unit. Each shell is located beneath its respective low-pressure turbine.

The three condenser shells are designated as the low-pressure shell, the intermediate-pressure shell, and the high-pressure shell. Each shell has at least two tube bundles. Circulating water flows in series through the three single-pass shells.

Each condenser shell hotwell is divided longitudinally by a vertical partition plate. The hotwells of the three shells are interconnected by condensate channels. The condensate pumps take suction from the high-pressure condenser hotwell.

The condenser shells are located below the Turbine Building operating floor and are supported on the Turbine Building basemat (see Chapter 8).

Failure of or leakage from a condenser hotwell during plant shutdown only results in a minimum water level in the Turbine Building condenser area. Expansion joints are provided between each turbine exhaust opening and the steam inlet connections of the condenser shell. Water seals and their level indication, if required, are provided around the entire outside periphery to prevent leakage through the expansion joints. Level indication provides detection of leakage through the expansion joint. Three low-pressure feedwater heaters are located in the steam dome of each shell. Piping is installed for hotwell level control and condensate sampling.

During plant operation, steam expanding through the low-pressure turbines is directed downward into the Main Condenser and is condensed. The Main Condenser also serves as a heat sink for the turbine bypass system, emergency and high-level feedwater heater and drain tank dumps, and various other startup drains and relief valve discharges.

Main Condenser Evacuation System

The Main Condenser Evacuation System (MCES) removes the noncondensible gases from the power cycle. The MCES removes the hydrogen and oxygen produced by radiolysis of water in the reactor, and other power cycle noncondensible gases, and exhausts them to the Offgas System during plant power operation, and to the Turbine Building compartment exhaust system at the beginning of each startup and during shutdown.

The MCES consists of two 100% capacity, double-stage Steam Jet Air Ejector (SJAE) units (complete with intercondenser) for power plant operation where one SJAE unit is normally in operation and the other is on standby, or both in half load operation, as well as two 50% capacity mechanical vacuum pumps for use during startup.

During the initial phase of startup, when the desired rate of air and gas removal exceeds the capacity of the SJAEs, and nuclear steam pressure is not adequate to operate the SJAE units, the mechanical vacuum pumps establish a vacuum in the Main Condenser. The discharge from the vacuum pumps is then routed to the Turbine Building Compartment Exhaust (TBCE) system, since there is then little or no effluent radioactivity or hydrogen present. Radiation detectors in the TBCE system and plant vent stack alarm in the Main Control Room (MCR) if abnormal radioactivity is detected. Radiation monitors are provided on the main steam lines which trip the vacuum pump if abnormal radioactivity is detected in the steam being supplied to the condenser.

The SJAEs are placed in service to remove the gases from the Main Condenser after a vacuum is established in the Main Condenser by the mechanical vacuum pump and when sufficient nuclear steam pressure is available.

During normal power operation, the SJAEs are normally driven by main steam, with the auxiliary steam supply system on automatic standby. The main steam supply, however, is normally used during startup and low load operation, and auxiliary steam is available for normal use of the SJAEs during early startup, as an alternative to the main steam or should the mechanical vacuum pumps prove to be unavailable.

Turbine Gland Seal System

The Turbine Gland Seal System (TGSS) provides steam to the turbine glands and the turbine valve stems. The TGSS prevents leakage of air into or radioactive steam out of the turbine shaft and turbine valves. The gland steam condenser collects air and steam mixture, condenses the steam, and discharges the air leakage to the atmosphere via the main vent by one of two redundant motor-driven blowers.

Turbine Bypass System

The Turbine Bypass System (TBS) provides the capability to discharge main steam from the reactor directly to the condenser to minimize step-load reduction transient effects on the Reactor Coolant System. The TBS is also used to discharge main steam during reactor hot standby and cooldown operations.

The TBS consists of Turbine Bypass Valves (TBV) connected to the TMSS Main Steam Lines via TMSS System Piping. The outlets of TBVs are connected to the Main Condenser via pressure reducers.

The system is designed to bypass at least 110% of the rated main steam flow directly to the condenser. The TBS, in combination with the reactor systems, provides the capability to shed 100% of the TG rated load without reactor trip and without the operation of SRVs.

The turbine bypass valves are opened by triply-redundant signals received from the Steam Bypass and Pressure Control System whenever the actual steam pressure exceeds the preset steam pressure by a small margin. This occurs when the amount of steam generated by the reactor cannot be entirely used by the turbine. This bypass demand signal causes fluid pressure to be applied to the operating cylinder, which opens the first of the individual valves. As the bypass demand increases, additional bypass valves are opened, dumping the steam to the condenser. The bypass valves are equipped with fast-acting solenoid valves to allow rapid opening of bypass valves upon turbine trip or generator load rejection.

The bypass valves automatically trip closed whenever the condenser pressure increases to a preset value. Individual bypass valves are also closed on loss of electrical power or hydraulic system pressure. The bypass valve hydraulic accumulators have the capability to open the associated valve for at least six seconds with hydraulic supply unit failure or loss of preffered power.

When the plant is at zero power, hot standby or initial cooldown, the system is operated manually by the control room operator or by the plant automation system. The measured reactor pressure is then compared against, and regulated to, the pressure set by the operator or automation system.

Steam Extraction System

Extraction steam from the high-pressure turbine supplies the 5th and 6th stages of feedwater heating, and MSRs, and extraction steam from the low-pressure turbines supplies the first three stages and the 4th stage direct contact feedwater heater. An additional low-pressure extraction drained directly to the condenser protects the last-stage buckets from erosion induced by water droplets.

Condensate Purification System

The Condensate Purification System (CPS) consists of high-efficiency filters arranged in parallel and operated in conjunction with a normally-closed filter bypass. The CPS also includes bead resin, ion-exchange demineralizer vessels arranged in parallel. A resin trap is installed downstream of each demineralizer vessel to preclude gross resin leakage into the power cycle in case of vessel resin retention screen failure. The CPS system achieves the water quality effluent conditions required for reactor power operation defined in the plant water quality specification. The CPS components are located in the Turbine Building.

Provisions are included to permit cleaning and replacement of the ion-exchange resin. Each of the demineralizer vessels has inlet and outlet isolation valves which are remotely controlled from the local CPS control panel and the main control room.

A demineralizer system bypass valve is also provided which is manually or automatically controlled from the local control panel or main control room. Pressure downstream of the demineralizer or high demineralizer differential pressure is indicated and is alarmed in the main control room to alert the operator. The bypass can be used during startup and in an emergency for short periods of time until the CPS flow is returned to normal or the plant is brought to an orderly shutdown.

During power operation, the condensate is well deaerated in the condenser and continuous oxygen injection is used to maintain the level of oxygen content in the final FW.

To minimize corrosion product input to the reactor during startup, recirculation lines to the condenser are provided from the high-pressure FW heater outlet header.

Prior to plant startup, cleanup is accomplished by allowing the system to recirculate through the condensate polishers for treatment prior to feeding any water to the reactor during startu.

Condensate and Feedwater System

The Condensate and Feedwater System (C&FS) consists of the piping, valves, pumps, heat exchangers, controls and instrumentation, and the associated equipment and subsystems that supply the reactor with heated FW in a closed steam cycle utilizing regenerative FW heating. The system described in this subsection extends from the main condenser outlet to (but not including) the seismic interface restraint outside of containment. The remainder of the system, extending from the restraint to the reactor, is described in Chapter 3. Turbine-cycle steam is utilized for a total of seven stages of FW heating, six stages of closed FW heaters, and one direct-contact FW heaters (feedwater tank). The drains from each stage of the closed low-pressure FW heaters are cascaded through successively lower pressure FW heaters to the main condenser. The high-pressure heater drains are routed to the feedwater tank.

The highest pressure FW heater is fed steam directly from the main steam line and is used only during power maneuver operations. Feedwater temperature is raised from its normal 215.6°C (420 °F) up to 252.2°C (486°F) in order to lower reactor power by ~15%. This will retain the same flexibility in power maneuvers for minimizing fuel duty that forced circulation BWRs have.

The C&FS consists of four 33-37% capacity condensate pumps (three normally operating and one on automatic standby), four 33-45% capacity reactor FW pumps (three normally in operation and one on automatic standby), four 33% nominal capacity feedwater booster pumps (three normally in operation and one on automatic standby), three stages of low-pressure closed FW heaters, a direct-contact FW heater (feedwater tank) and three stages of high-pressure FW heaters, piping, valves and instrumentation. The condensate pumps take suction from the condenser hotwell and discharge the deaerated condensate into one common header, which feeds the Condensate Purification System (CPS). Downstream of the CPS, the condensate is taken by a single header, through the auxiliary condenser/coolers (one gland steam exhauster condenser and two sets of SJAE condensers and offgas recombiner condenser/coolers). The condensate then branches into three parallel strings of low-pressure FW heaters. Each string contains three stages of low-pressure FW heaters. The strings join together at a common header, which is routed to the #4 feedwater heater tank, which supplies heated feedwater to the suction of the reactor FW booster pumps, then FW pumps. Each reactor FW pump is driven by an adjustable-speed electrical motor.

Another input to the feedwater tank consists of the drains, which originate from the crossaround steam moisture separators and reheaters and from the two sets of high-pressure FW heaters.

The reactor FW pumps discharge the FW into two parallel high-pressure FW heater strings, each with three stages of high-pressure FW heaters. Downstream of the high-pressure FW heaters, the two strings are then joined into a common header, which divides into two FW lines that connect to the reactor with six penetrations (each FW line branches to three).

A bypass is provided around the FW tank and reactor FW pumps to permit supplying FW to the reactor during early startup without operating the FW pumps, using only the condensate pumps. During startups, a low-flow control valves, with flow supplied by either the condensate pumps or via preselected (two out of four) FW pumps operating at their minimum fixed speed, control the RPV level.

One more bypass, equipped with a flow control valve, is provided around the high-pressure heaters for isolating them during power operation for heater maintenance or for reducing final FW temperature to extend the end of fuel cycle.

Circulating Water System (CIRC)

The Circulating Water System (CIRC), which operates continuously during power generation, including startup and shutdown, provides cooling water for removal of the power cycle waste heat from the main condensers and transfers this heat to the normal power heat sink.

The CIRC consists of the following components:

- Screen house and intake screens
- Pumps and pump discharge valves
- Condenser water boxes and piping and valves
- Condenser tube cleaning equipment
- Water box drain subsystem
- Related support facilities for inventory makeup and blowdown

The cooling water is circulated by motor-driven pumps. The pumps typically are arranged in parallel and discharge lines combine into two parallel circulating water supply lines to the main condenser. For the ESBWR conceptual design, each circulating water supply line connects to a low-pressure condenser shell inlet water box. An interconnecting line fitted with a butterfly valve is provided to connect both circulating water supply lines. The discharge of each pump is fitted with a remotely-operated valve. This arrangement permits isolation and maintenance of any one pump while the others remain in operation and minimizes the backward flow through a tripped pump.

The CIRC and condenser are designed to permit isolation of each set of the three series connected tube bundles to permit repair of leaks and cleaning of water boxes while operating at reduced power.

The CIRC includes water box vents and or a vacuum priming pumps to help fill the condenser water boxes during startup and removes accumulated air and other gases from the water boxes during normal operation.

A chemical additive subsystem and ball tube-cleaning system is also provided to prevent the accumulation of biological growth and chemical deposits within the wetted surfaces of the system.

Other Turbine Auxiliary Systems

Turbine Component Cooling Water System

The Turbine Component Cooling Water System (TCCWS) is a closed-loop cooling water system that supplies cooling water through the TCCW heat exchangers to Turbine Island equipment coolers and rejects heat to the Plant Service Water System (PSWS, see Chapter 5). It operates at a higher pressure than the PSWS, so that any intersystem leakage will not affect Turbine Building equipment. The system consists of a single loop with three 50% capacity pumps and four 50% capacity heat exchangers.

Station Electrical Power

Offsite Power System

The offsite power system consists of the set of electrical circuits and associated equipment that are used to interconnect the offsite transmission system with the plant main generator and the onsite electrical power distribution system, as indicated on the one-line diagram, Figure 9-2.

The system includes the plant switchyard and the high-voltage tie lines to the main generator circuit breaker, the high-side motor-operated disconnects (MODs) of the unit auxiliary transformers (UATs), and the high-side MODs of the reserve auxiliary transformers (RATs).

Power is supplied to the plant from the switchyard connected to two electrically independent and physically separate offsite power sources as follows:

- **Normal Preferred** source through the unit auxiliary transformers (UAT)
- **Alternate Preferred** source through the reserve auxiliary transformers (RAT)

During plant startup, normal or emergency shutdown, or during plant outages, the offsite power system serves to supply power from the offsite transmission system to the plant auxiliary and service loads.

During normal operation, the offsite power system is used to transmit generated power to the offsite transmission system and to the plant auxiliary and service loads.

The onsite power distribution system is powered continuously by the offsite power source throughout plant startup, normal operation, and normal or emergency shutdown. When the generator breaker is tripped, power to the plant continues to be fed from the offsite power source through either the UATs or the RATs.

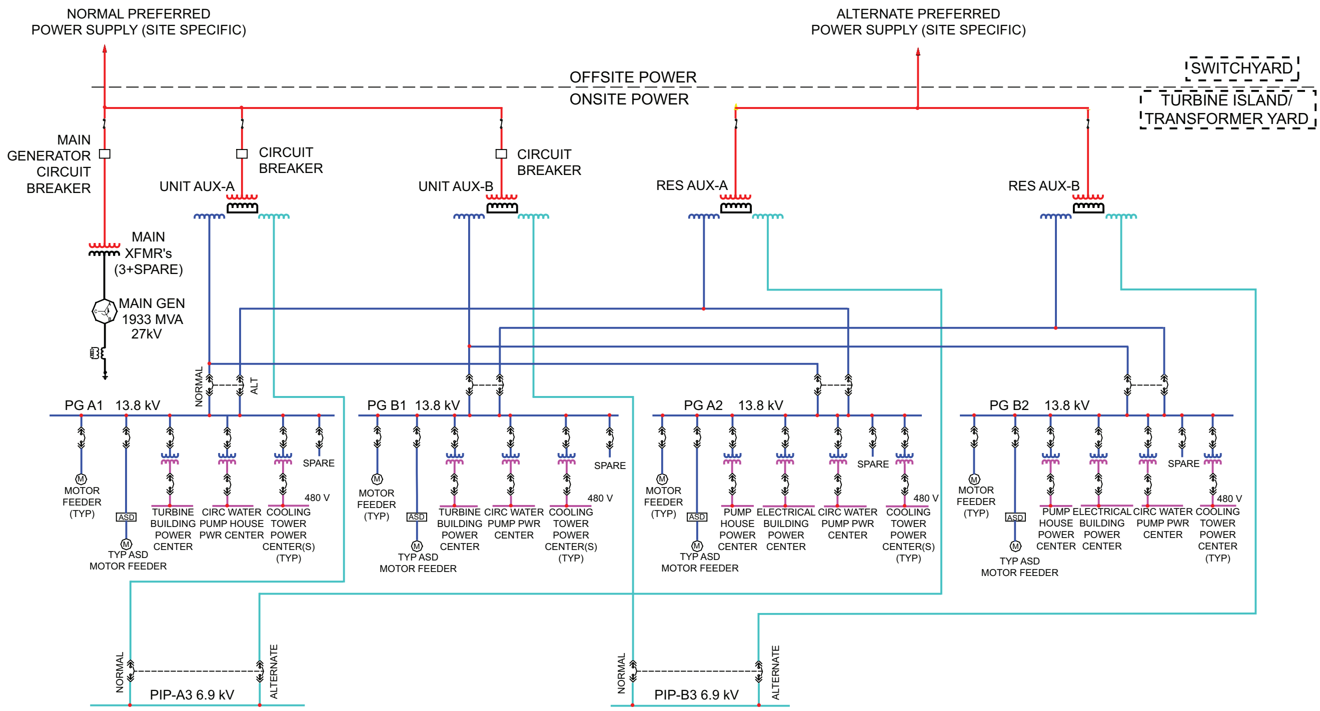

Figure 9-2. ESBWR Electrical One-Line

Onsite AC Power Distribution

General

The onsite AC power system is configured into two separate power load groups (see Figure 9-2). Each power load group is fed by a separate unit auxiliary transformer (UAT), each with a redundant reserve auxiliary transformer (RAT) for backup, and consists of two types of buses:

- Power Generation (PG) nonsafety-related buses - are those buses that are not directly backed by standby onsite AC power sources and have connections to the normal or alternate offsite source through the UATs or RATs, respectively. The PG nonsafety-related buses are the 13.8-kV unit auxiliary switchgear and associated lower voltage load buses
- Plant Investment Protection (PIP) nonsafety-related buses - are those buses that are backed by the standby onsite AC power supply system and have connections to the normal preferred and alternate preferred offsite sources through the UATs and RATs, respectively. Backfeed to the standby onsite AC power source is prevented by reverse power relaying. The PIP nonsafety-related buses are the 6.9-kV PIP buses and associated lower voltage load buses exclusive of the safety-related Isolation Power Center buses

The PG nonsafety-related buses feed nonsafety-related loads that are required exclusively for unit operation and are normally powered from the normal preferred power source through the UATs. These buses are also capable of being powered from the alternate preferred power source (RATs) in the event that the normal preferred power source is unavailable. On restoration of UAT power, a manually-selected bus transfer may be performed.

The PIP nonsafety-related buses feed nonsafety-related loads generally required to remain operational at all times or when the unit is shut down. In addition, the PIP nonsafety-related buses supply AC power to the safety-related buses. The PIP nonsafety-related buses are backed up by a separate standby onsite AC power supply system connected to each PIP bus. These buses are also capable of being powered from the alternate preferred power source (RATs), through an auto bus transfer, in the event that the normal preferred power source is unavailable. On restoration of UAT power, a manually-selected bus transfer may be performed. Refer to Figure 9-3 and 9-4.

Medium Voltage AC Power Distribution System

Power is supplied from the UATs and RATs at 13.8 kV and 6.9 kV to the PG and PIP buses. There are four PG buses, each being powered from one of the two UATs, or if the UATs are unavailable, from one of the two RATs. The source breakers for each PG bus are electrically interlocked to prevent simultaneous connection of the UATs and RATs to the PG buses.

Two 6.9-kV PIP buses (PIP-A and PIP-B) provide power for the nonsafety-related PIP loads. PIP-A and PIP-B buses are each backed by a separate standby onsite AC power supply source. Each PIP bus is normally powered from the normal preferred

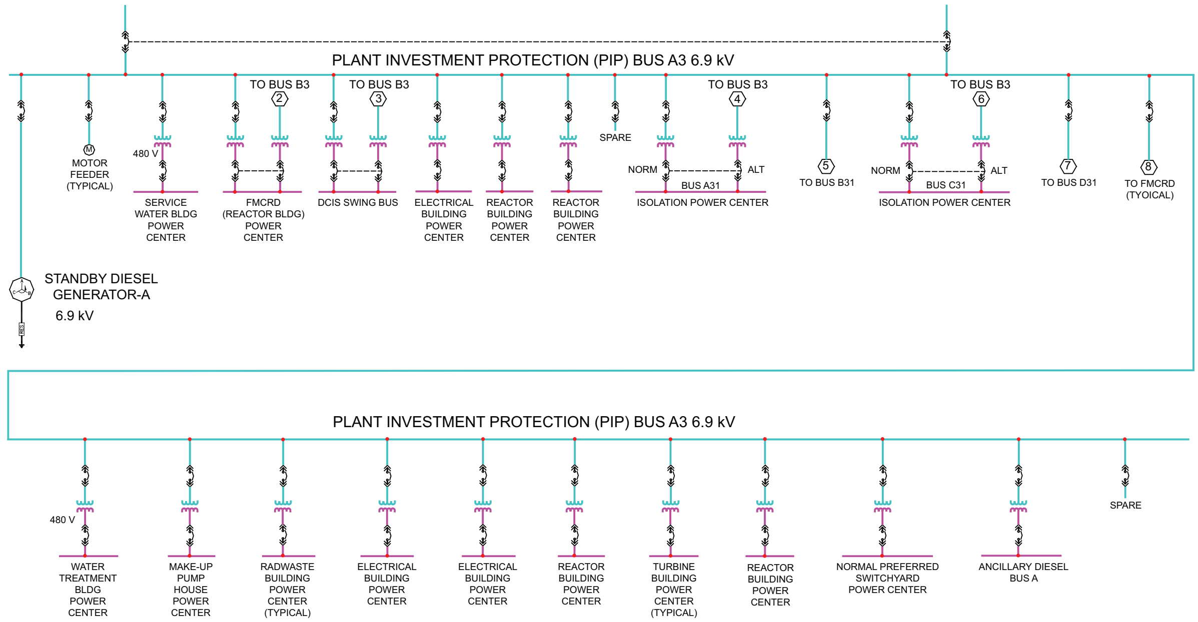

Figure 9-3. ESBWR PIP Bus - A

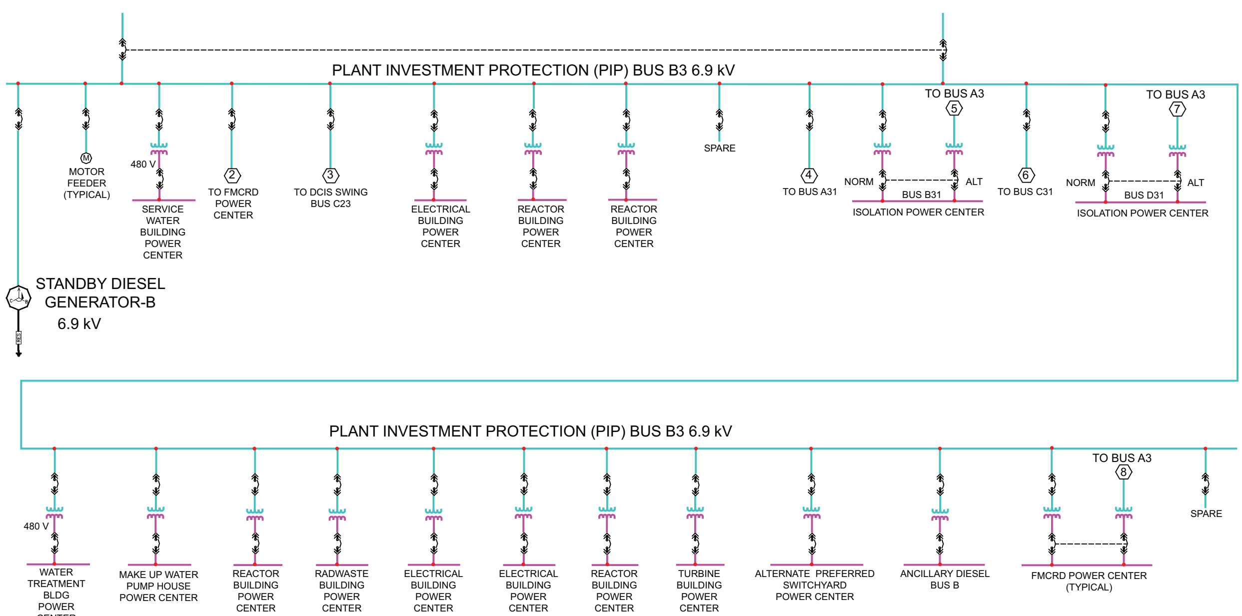

Figure 9-4. ESBWR PIP Bus - B

power source through the UAT of the same load group. Additionally, in the event of unavailability of the normal preferred power source, each PIP bus has connections to and can be powered from the alternate preferred power source through the RAT of the same load group. The source breakers of the normal and alternate preferred power sources are electrically interlocked to prevent simultaneous connection of UATs and RATs to the PIP buses.

Standby AC power for the PIP nonsafety-related buses is supplied by standby diesel generators at 6.9 kV and distributed by the nonsafety-related power distribution system. The 6.9-kV PIP buses are automatically transferred to the standby diesel generators when the normal and alternate preferred power supplies to these buses are lost. The startup time for the standby diesel generators is much less critical than in previous BWRs, due to the passive ECCS - approximately two minutes to start and fully load.

Low Voltage AC Power Distribution System

The low-voltage AC power distribution system includes power centers, motor control centers (MCCs), distribution transformers, and distribution panels as well as the associated overcurrent protective devices, protective relaying, and local instrumentation and controls. It also includes all cables interconnecting the buses to their sources and loads.

Power centers supply circuits that operate at 480-VAC through 120-VAC, and include MCCs, motor loads and the ancillary diesel buses. The power centers are of the single-fed or double-ended type depending on the redundancy requirements of the loads powered by a given power center. The power supplies to the double-ended power center transformers of the PIP nonsafety-related buses are supplied from different buses. Each double-ended power center is normally powered by its normal power source through its normal source main

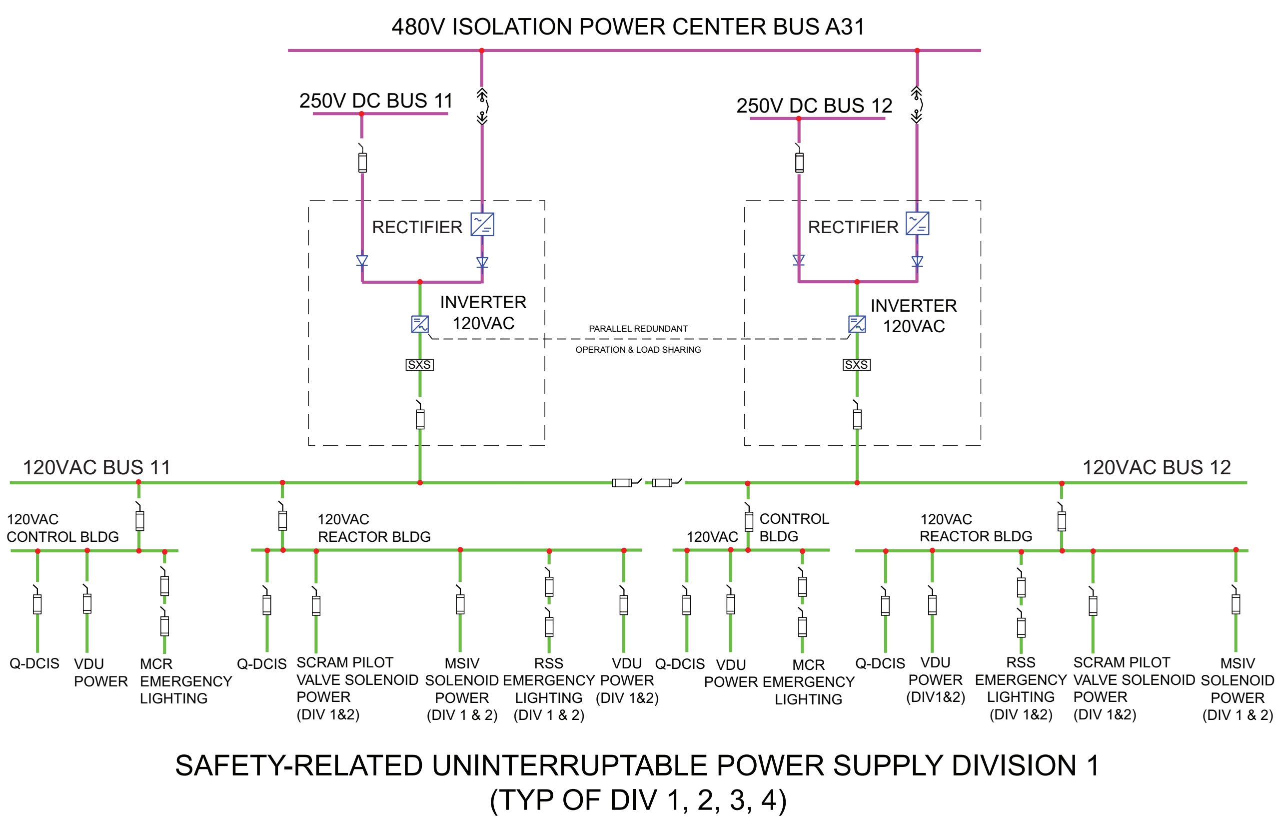

Figure 9-5. ESBWR Safety-Related Uninterruptable AC Power

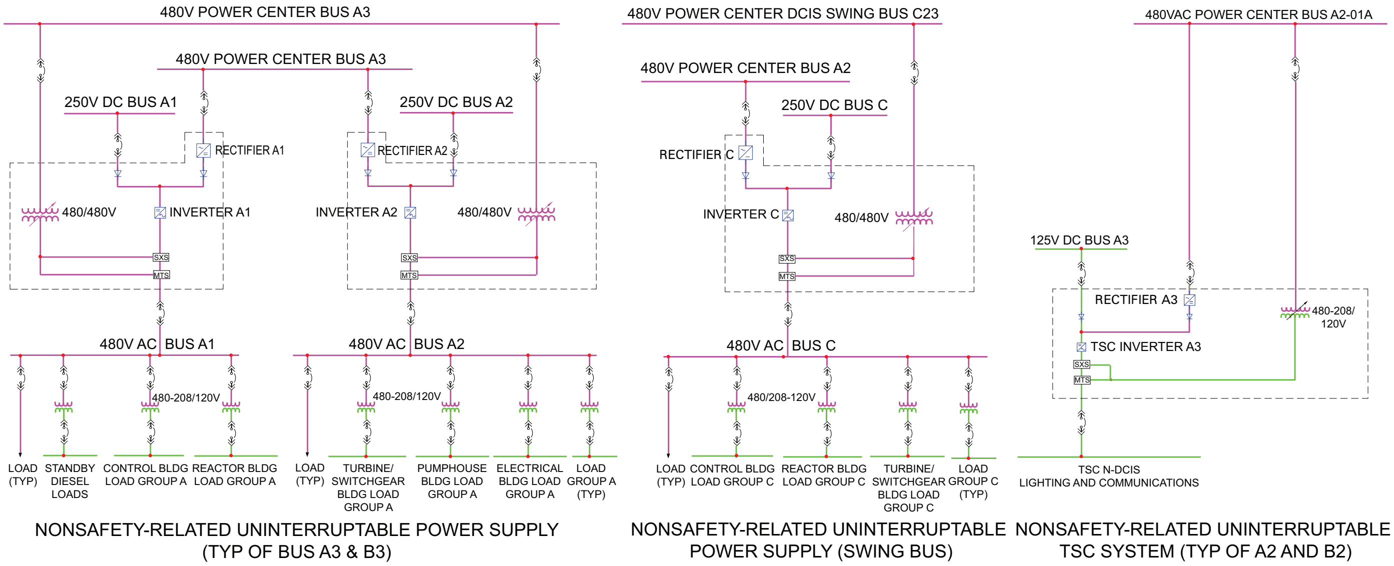

Figure 9-6. ESBWR Nonsafety-Related Uninterruptable AC Power

breaker, with the alternate source main breaker open. The power center normal and alternate source main breakers are electrically interlocked to prevent simultaneous powering of the power center by normal and alternate sources.

Isolation Power Centers

The isolation power centers are powered from the PIP nonsafety-related buses, which are backed up by the standby diesel generators. There are four isolation power centers, one each for Divisions 1, 2, 3 and 4. Each isolation power center is double-ended and can be powered from either of the PIP load group buses. The normal and alternate source main breakers of each isolation power center are electrically interlocked to prevent powering the isolation power center from the normal and alternate sources simultaneously. The isolation power centers are shown in Figures 9-3 and 9-4.

The isolation power centers supply power to safety-related loads of their respective division. These loads consist of the safety-related battery chargers or rectifiers. In addition, there is no safety-related lighting that operates directly from the 480-VAC in the ESBWR design. There are no safety-related actuators (pumps, valves, etc.) that operate directly from 480-VAC (or higher) in the ESBWR design

Motor Control Centers

MCCs supply 99-kW and smaller motors, control power transformers, process heaters, motor operated valves and other small electrically operated auxiliaries, including 480-VAC to 208/120-VAC and 480-VAC to 240/120-VAC transformers. MCCs are assigned to the same load group as the power center that supplies their power.

Safety-Related Uninterruptible AC Power Supply System

Figures 9-5 and 9-7 show the overall safety-related Uninterruptible AC Power Supply (UPS) system and Direct Current Power Supply (250-VAC). The safety-related UPS for each of the four divisions is supplied from a 480-VAC isolation power center in the same division. The isolation power centers are connected to PIP non safety-related buses, which are backed by standby diesel generators. Divisions 1, 2, 3, 4 each have two rectifiers, two batteries, and two inverters. Each rectifier receives 480-VAC normal power from the isolation power center of that division converts it to 250-VDC. The 480-VAC/250-VDC rectifier and a safety-related 72-hour battery of that division supply 250-VDC power through diodes to a common inverter with an output of 120-VAC single phase.

Power is distributed to the individual safety-

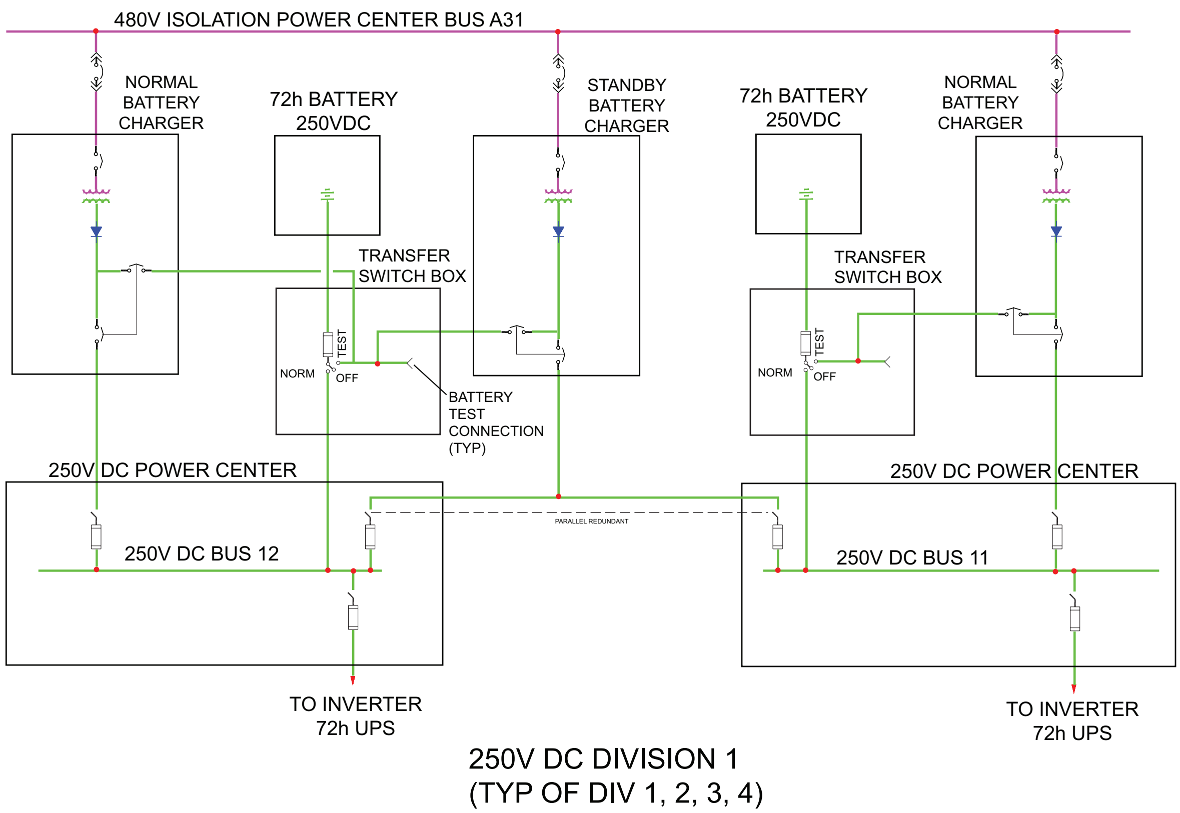

Figure 9-7. ESBWR Safety-Related DC Power

related loads from associated 120-VAC distribution panels, which supply power to the Reactor and Control Buildings.

Nonsafety-Related Uninterruptible Power Supply System

Figure 9-6 shows the overall nonsafety-related UPS. The nonsafety-related UPS for each of the two plant power distribution load groups is supplied from a 480-VAC power center in the same load group, with standby onsite AC power of the same load group providing backup power should a failure of the normal supply occur. Emergency power of the same load group from 250-VDC batteries is provided should loss of normal and standby onsite AC power sources occur.

A third nonsafety-related UPS is provided to supply the nonsafety-related DCIS loads. This load group's nonsafety-related UPS is normally powered from a 480-VAC double-ended power center, which can receive power from either of the two power load groups. The power center normal and alternate source main breakers are electrically interlocked to prevent the normal and alternate sources from simultaneously providing power to the power center. Additionally, standby onsite AC power from either of the two load groups provides backup power should a failure of the normal and alternate supplies occur. Emergency power of the same load group from 250-VDC batteries is provided should loss of normal, alternate, and standby onsite AC power sources occur.

Two dedicated, nonsafety-related UPSs are provided for the Technical Support Center (TSC), also in a two-load group configuration. Power for each TSC nonsafety-related UPS is normally supplied from a 480-VAC power center in the same load group, with standby onsite AC power of the same load group providing backup power should a failure of the normal supply occur. Backup power of the same load group from 125-VDC batteries is provided should loss of normal and standby onsite AC power sources occur.

The nonsafety-related UPS provides reliable, uninterruptible AC power for nonsafety-related equipment needed for continuity of power plant operation. UPS loads are divided into three load groups. Each UPS load group includes a solid-state inverter, solid-state rectifier, solid-state transfer switch, manual transfer switch, and distribution transformers with associated distribution panels.

DC Power Distribution

General

Completely independent safety-related and nonsafety-related DC power systems are provided. The safety-related DC system is shown in Figure 9-7. The nonsafety-related DC system is shown in Figure 9-8.

Eight independent safety-related 250-VDC systems are provided, two each for Divisions 1, 2, 3 and 4. They provide four divisions of independent and redundant onsite sources of power for operation of safety-related loads, monitoring, and MCR emergency lighting.

Five independent nonsafety-related DC systems are provided consisting of five 250-VDC batteries and two 125-VDC batteries. The nonsafety-related DC systems supply power for control and switching, switchgear control, TSC, nonsafety-related instrumentation and turbine auxiliaries.

Safety-Related Station Batteries and Battery Chargers

250-VDC Safety-Related DC Systems Configuration

Figure 9-7 shows the overall 250-VDC system provided for Safety-Related Divisions 1, 2, 3 and 4. Divisions 1, 2, 3, and 4 consist of two separate battery sets for each division. Each set supplies power to the safety-related inverters for at least 72 hours following a design basis event. The DC systems are operated ungrounded for increased reliability. Each of the safety-related battery systems has a 250-VDC battery, a battery charger, a standby battery charger, a main distribution bus, and a ground detection panel. One divisional battery charger is used to supply each group's DC distribution panel bus and its associated battery. The divisional battery charger is fed from its divisional 480-VAC Isolation Power Center. The main DC distribution bus feeds the UPS inverter. Each division has a standby charger to act as backup to either of the batteries of that division.

The four safety-related divisions are supplied power from four independent Isolation Power Centers. The 250-VDC systems supply DC power to Divisions 1, 2, 3 and 4, respectively. The safety-related DC system is designed so that no single active failure in any division of the 250-VDC system results in conditions that prevent safe shutdown of the plant while a separate division has been taken out of service for maintenance.

The plant design and circuit layout of the DC systems provide physical separation of the equipment, cabling, and instrumentation essential to plant safety. Each 250-VDC battery is separately housed in a ventilated room apart from its charger, distribution, and ground detection panels. Equipment of each division of the DC distribution system is located in an area separated physically from the other divisions. All the components of safety-related 250-VDC systems are housed in Seismic Category I structures.

Safety-Related Batteries

In divisions 1, 2, 3 and 4 the two separate 250 volt In Divisions 1, 2, 3 and 4 the two separate 250-VDC safety-related batteries per division are sized together so that their total rated capacity will exceed the required battery capacity per division for 72-hour station blackout conditions. The DC system minimum battery terminal voltage at the end of the discharge period is 210-VDC.

The safety-related batteries have sufficient stored capacity without their chargers to independently supply the safety-related loads continuously for the time period stated above. The battery banks are designed to permit the replacement of individual cells.

Safety-Related Battery Chargers

The safety-related battery chargers are full-wave, silicon-controlled rectifiers. The chargers are suitable for continuously float charging the batteries. The chargers operate from a 480-VAC, three phase, 60-Hz supply. The power for each divisional battery charger is supplied by that divisions dedicated Isolation Power Center. The standby battery charger is

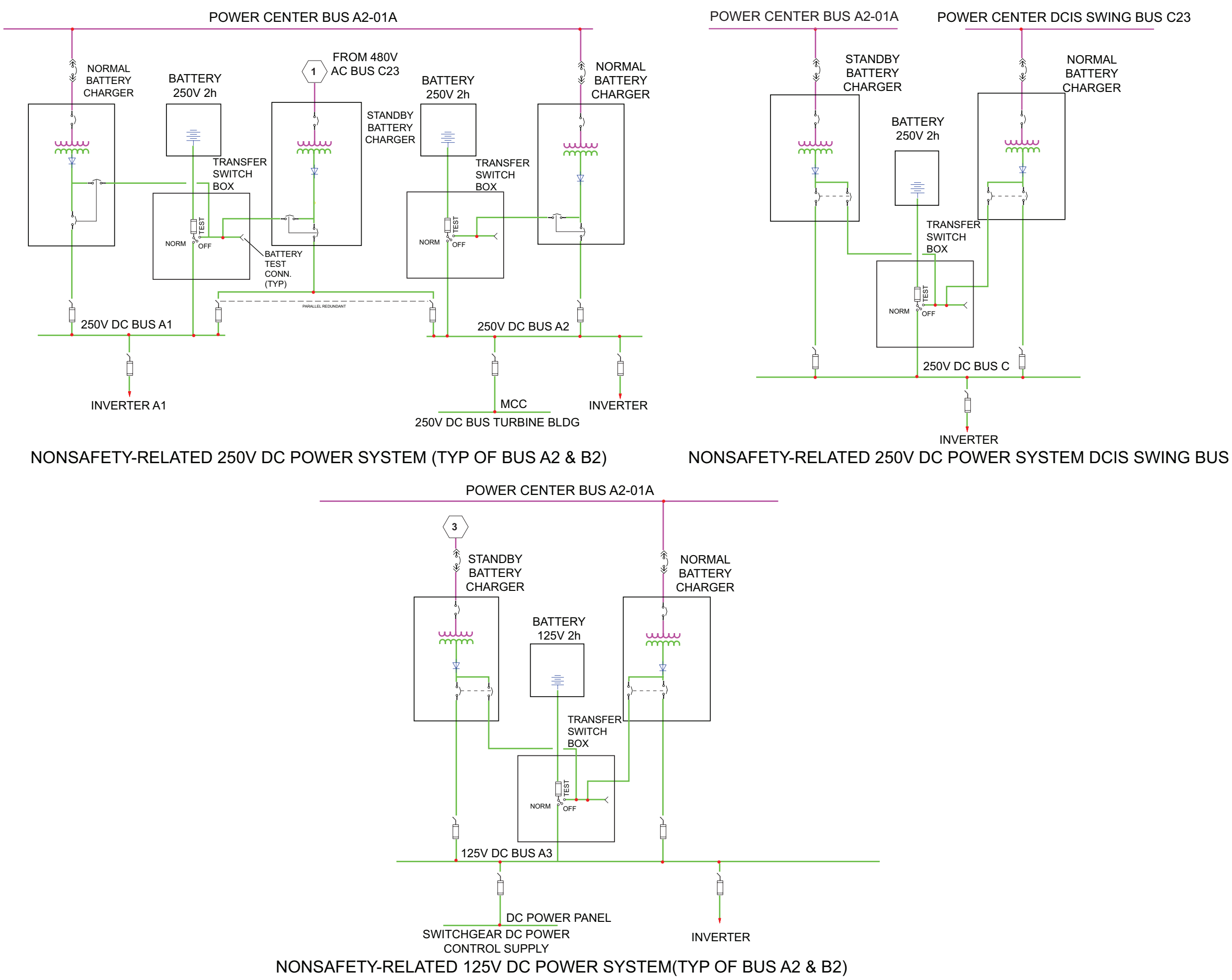

Figure 9-8. ESBWR Nonsafety-Related DC Power

used to equalize either of its associated divisional batteries, or as a replacement to the normal charger associated with that battery.

Standby chargers are supplied from the same Isolation Power Center as the normal charger. Each battery charger is capable of restoring its battery after a bounding design basis event discharge within 24 hours to a state that the battery can perform its design basis function for subsequent postulated operational and design basis functions, while at the same time supplying the largest combined demands associated with the individual battery.

The battery chargers are the constant voltage type, adjustable between 240 and 290 volts, with the capability of operating as battery eliminators. The battery eliminator feature is incorporated as a precautionary measure to protect against the effects of inadvertent disconnection of the battery.

The battery chargers output is of a current limiting design. The battery chargers are designed to prevent their AC source from becoming a load on the batteries because of power feedback from loss of AC power.

Nonsafety-Related Station Batteries and Battery Chargers

125-VDC and 250-VDC Nonsafety-Related DC Systems Configuration

Figure 9-8 shows the overall 125-VDC and 250-VDC nonsafety-related DC systems. The DC systems are operated ungrounded for increased reliability. Each of the DC systems has a battery, a battery charger, a standby battery charger, main DC distribution bus, and ground detection panel. The main DC distribution buses feed the local DC distribution buses, UPS inverter, and/or DC motor control center.

The plant design and circuit layout of the nonsafety-related DC systems provide physical separation of the equipment, cabling and instrumentation associated with the load groups of nonsafety-related equipment. Each 125-VDC and 250-VDC battery is separately housed in a ventilated room apart from its charger, distribution, and ground detection panels. Equipment of each load group of the DC distribution system is located in an area separated physically from the other load groups.

The nonsafety-related DC power is required for control and switching functions such as control of the main generator circuit breaker, 13.8-kV, 6.9-kV and 480-VAC switchgear, DC motors, control relays, meters and indicators.

Nonsafety-Related Batteries

The 125-VDC non-Class 1E batteries are sized for 2-hour duty cycles at a discharge rate of 2 hours. The DC system minimum battery terminal voltage at the end of the discharge period is 105 volts.

The 250-VDC nonsafety-related batteries are sized for 2-hour duty cycles at a discharge rate of 2 hours. The DC system minimum battery terminal voltage at the end of the discharge period is 210-VDC.

The nonsafety-related batteries have sufficient stored capacity without their chargers to independently supply their loads continuously for at least 2 hours. Each distribution circuit is capable of transmitting sufficient energy to start and operate all required loads in that circuit. The battery banks are designed to permit replacement of individual cells.

Nonsafety-Related Battery Chargers

The nonsafety-related battery chargers are full-wave, silicon-controlled rectifiers, or an acceptable alternate design. The chargers are suitable for float charging the batteries. The chargers operate from a 460-VDC, three phase, 60-Hz supply. Each charger is supplied from a separate power center than its standby charger, which is backed by the standby diesel generator.

Standby chargers are used to equalize battery charging. Standby chargers are supplied from a different power center than its normal charger.

The battery chargers are the constant voltage type, with the 125-VDC system chargers having a voltage adjustable between 125-VDC and 145-VDC and the 250-VDC system chargers having a voltage adjustable between 240-VDC and 290-VDC, with the capability of operating as battery eliminators. The battery eliminator feature is incorporated as a precautionary measure to protect against the effects of inadvertent disconnection of the battery.

The battery charger's output is of a current limiting design. The battery chargers are designed to prevent their AC source from becoming a load on the batteries caused by power feedback from a loss of AC power.

HITACHI

Chapter 10 Radioactive Waste Systems

Overview

The radwaste facility has been significantly improved compared to past designs. The design accommodates skid mounted processing technologies for both liquid and solid radwaste processing, which improves the efficiency of the process and allows new technologies to be readily adapted to the existing ESBWR radwaste facility. The radwaste building is designed to accommodate various processing alternatives. Permanently installed equipment are the collection and sample tanks and the support systems required to use the processing skids. The liquid radwaste system is designed for 100% recycle with no offsite release.

The impact of these improvements in the ESBWR design gives assurance that dewatered solid waste volume is less than 85 m3/year and dry solid waste volume is less than 370 m3/yr, reducing the radwaste volume significantly compared to current U.S. operating plants. Dose rates as a result of annual releases to the environment from the ESBWR radwaste systems are As Low As Reasonably Achievable (ALARA) in accordance with guidelines set forth in 10CFR50, Appendix I. These levels are several orders of magnitude below the NRC-established limits in 10CFR20.

The Radwaste Systems include the Liquid Waste Management System (LWMS), the Offgas System (OGS) and the Solid Waste Management System (SWMS).

Liquid Radwaste Management System

The ESBWR Liquid Waste Management System (LWMS) is designed to control, collect, process, handle, store, and recycle or dispose of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences. All potentially radioactive liquid wastes are collected in sumps or drain tanks at various locations in the plant and are transferred to collection tanks in the radwaste facility.

System components are designed and arranged in shielded enclosures to minimize exposure to plant personnel during operation, inspection, and maintenance. Tanks, processing equipment, pumps, valves, and instruments that may contain radioactivity are located in controlled access areas.

The LWMS normally operates on a batch basis. Provisions for sampling at important process points are included. Protection against accidental discharge is provided by detection and alarm of abnormal conditions and by administrative controls.

The LWMS is divided into several subsystems, so that the liquid wastes from various sources can be segregated and processed separately, based on the most economical and efficient process for each specific type of impurity and chemical content. Cross-connections between subsystems provide additional flexibility in processing the wastes by alternate methods and provide redundancy if one subsystem is inoperable.

The LWMS is housed in the radwaste building and consists of the following four subsystems:

- Equipment (low conductivity) drain subsystem
- Floor (high conductivity) drain subsystem
- Chemical drain subsystem
- Detergent drain subsystem

LWMS has been designed to recycle 100% of the ESBWRs liquid waste.

Equipment (Low Conductivity) Drain Subsystem

The subsystem block flow diagram is shown in Figure 10-1. The equipment drain collection tanks receive low conductivity inputs from various sources within the plant. These waste inputs have a high chemical purity and are processed on a batch basis. The equipment drain subsystem consists of three collection tanks and collection pumps, a processing system featuring a filtration system, reverse osmosis, mixed-bed ion exchangers and the associated piping, instrumentation and electrical systems as required, and two sample tanks and sample pumps.

Additional collection capacity is provided by cross connection to the floor drain collection tanks. Cross-connections with the floor drain subsystem allow processing through the system for floor drain treatment.

A resin trap (strainer or filter) is provided downstream of the last ion exchanger in series to prevent migration of resin in the event of an ion-exchanger screen failure.

The process effluents are collected in one of the two sample tanks for chemical and radiological analysis. If acceptable, the tank contents are returned to the condensate storage tank for plant reuse. A recycle line from the sample tanks allows the sampled effluents that do not meet water quality requirements to be pumped back to an Equipment (Low Conductivity) Drain Collection Tank or Floor (High Conductivity) Drain Collection Tank for additional processing. If the plant condensate inventory is high, the sampled process effluent may be discharged.

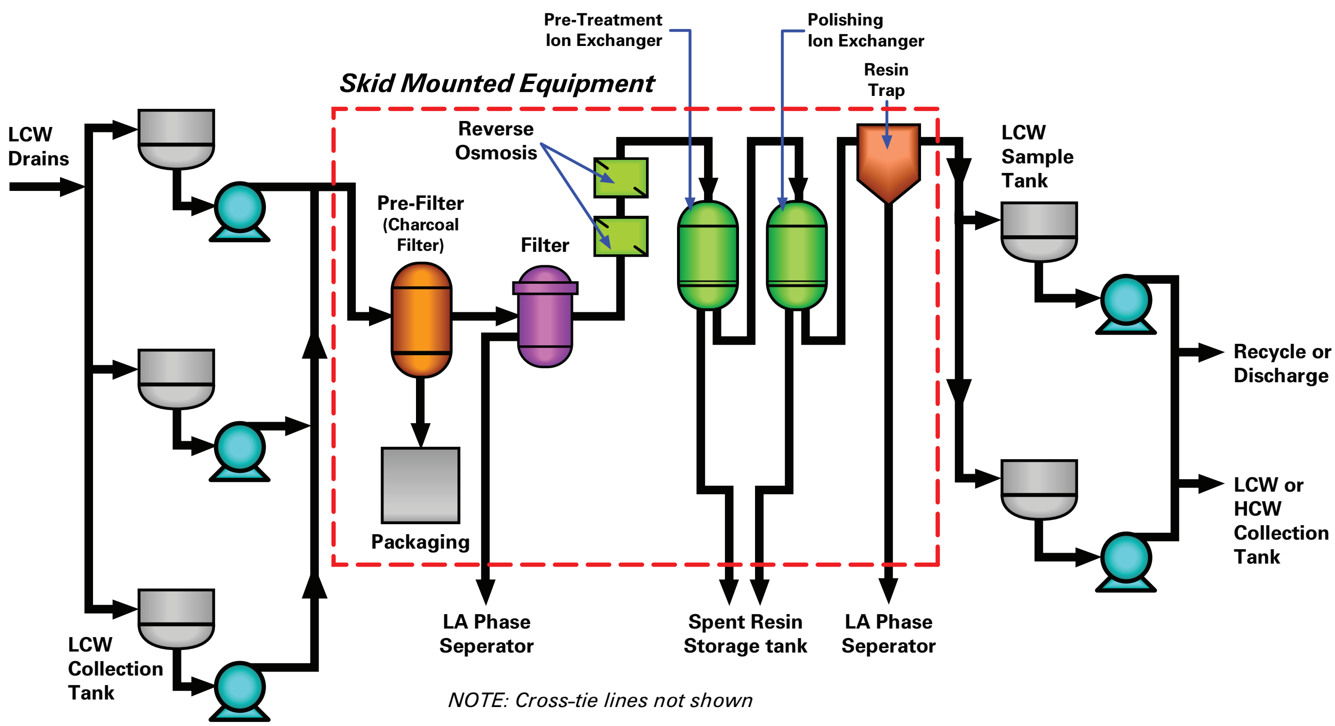

Figure 10-1. Equipment Drain Subsystem Schematic

Filters are backwashed periodically to maintain filtration capacity. Backwash waste is discharged to the Wet Solid Waste Collection Subsystem. Spent mixed-bed ion-exchanger resin is either discharged to a low activity resin hold up tank as a slurry, or sent directly to a High Integrity Container (HIC).

The capability exists to accept used condensate polishing resin in a Condensate Resin Holdup Tank in the SWMS. The used condensate polishing resin from Condensate Purification System is transferred to the Condensate Resin Holdup Tank prior to reuse in the pre-treatment mixed-bed ion exchanger in the equipment drain subsystem.

Floor (High Conductivity) Drain Subsystem

The subsystem block flow diagram is shown in Figure 10-2. The floor drain collection tanks receive high conductivity waste inputs from various floor drain sumps in the reactor building, fuel building, turbine building, and radwaste building. The floor drain collection tanks also receive waste input from chemical drain collection tanks.

The floor drain subsystem consists of two floor drain collection tanks and collection pumps, a processing system featuring a filtration system, reverse osmosis (RO) and mixed-bed ion exchanger and the associated piping, instrumentation and electrical systems as required, and two sample tanks and sample pumps. The waste collected in the floor drain collection tanks is processed on a batch basis. Cross-connections with the equipment drain subsystem also allow for processing through that subsystem. Additional collection capacity is also provided by cross connection to the equipment drain collection tanks.

A resin trap (strainer or filter) is provided downstream of the last ion exchanger in series to prevent migration of resin in the event of an ion-exchanger screen failure.

The floor drain sample tanks collect the process effluent, so that a sample may be taken for chemical and radiological analysis before discharging or recycling. The discharge path depends on plant

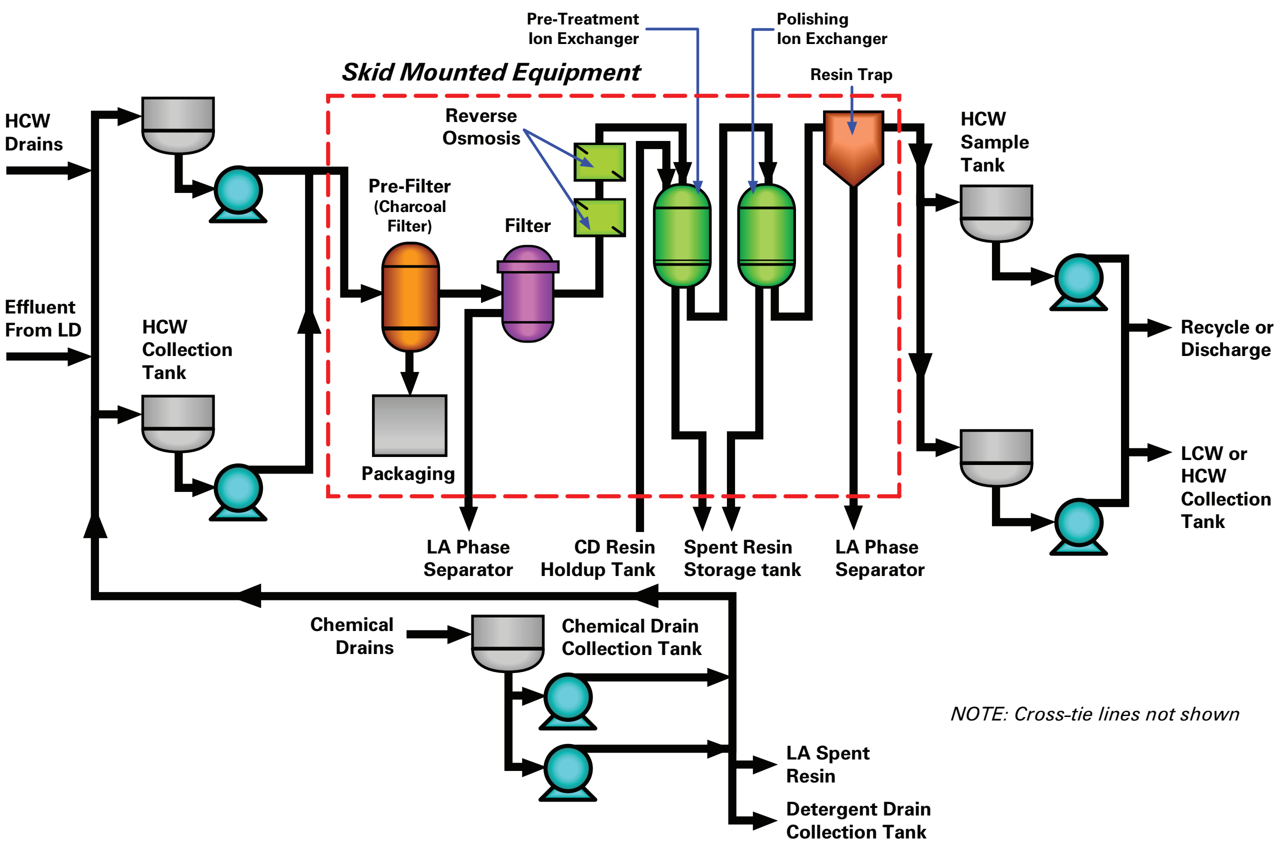

Figure 10-2. Floor Drain Subsystem and Chemical Drain Collection Schematic

water inventory. Off-standard quality effluent can be recycled to floor drain collection tanks or equipment drain collection tanks. If the treatment effluent meets water quality standards and if the water inventory permits it to be recycled, the processed floor drain effluent can be recycled to the condensate storage tank. If plant water inventory does not permit, processed water may be discharged offsite.

RO units create a brine waste stream, which is sent to a concentrated waste tank. Spent mixed-bed ion-exchanger resin is discharged to a Low Activity (LA) spent resin holdup tank as a slurry.

The capability exists to accept used condensate polishing resin in a Condensate Resin Holdup Tank in the SWMS. The used condensate polishing resin from Condensate Purification System is transferred to the Condensate Resin Holdup Tank prior to reuse in the pre-treatment mixed-bed ion exchanger in the floor drain subsystem.

Chemical Drain Subsystem

The subsystem block flow diagram is shown in Figure 10-2. To the greatest extent practicable, waste chemicals will be kept out of the LWMS, including the Chemical Drain Subsystem. The chemical waste collected in the chemical drain collection tank consists of laboratory wastes and decontamination solutions. After accumulating in the chemical drain collection tank, the tank contents are transferred to the low activity spent resin tank, detergent drain tank, or to the floor drain collection tanks. Chemical Control programs ensure that unapproved liquids are not added to chemical drain subsystem or LWMS.

Detergent Drain Subsystem

The subsystem block flow diagram is shown in Figure 10-3. Waste water containing detergent from the controlled laundry and personnel decontamination facilities throughout the plant is collected in the detergent drain collection tanks. The detergent drain subsystem consists of two detergent drain collection tanks and collection pumps, a processing system (consisting of a filtration system, organic pre-treatment equipment, and the associated piping, instrumentation and electrical systems as required), and two sample tanks and sample pumps. The detergent waste treatment includes suspended solid removal processing and organic material removal processing as necessary. The treated waste is collected in sample tanks. A sample is taken, and if discharge standards are met, the waste can be discharged offsite. The intended process path for detergent waste from the Detergent Sample Tanks is to the HCW (Floor Drain) System for further processing and return to the plants water inventory in the CST. Off standard quality water can either be recycled for further processing to the detergent collection tank or to the floor drain collection tank. A cross connection with the chemical drain collection subsystem is also provided.

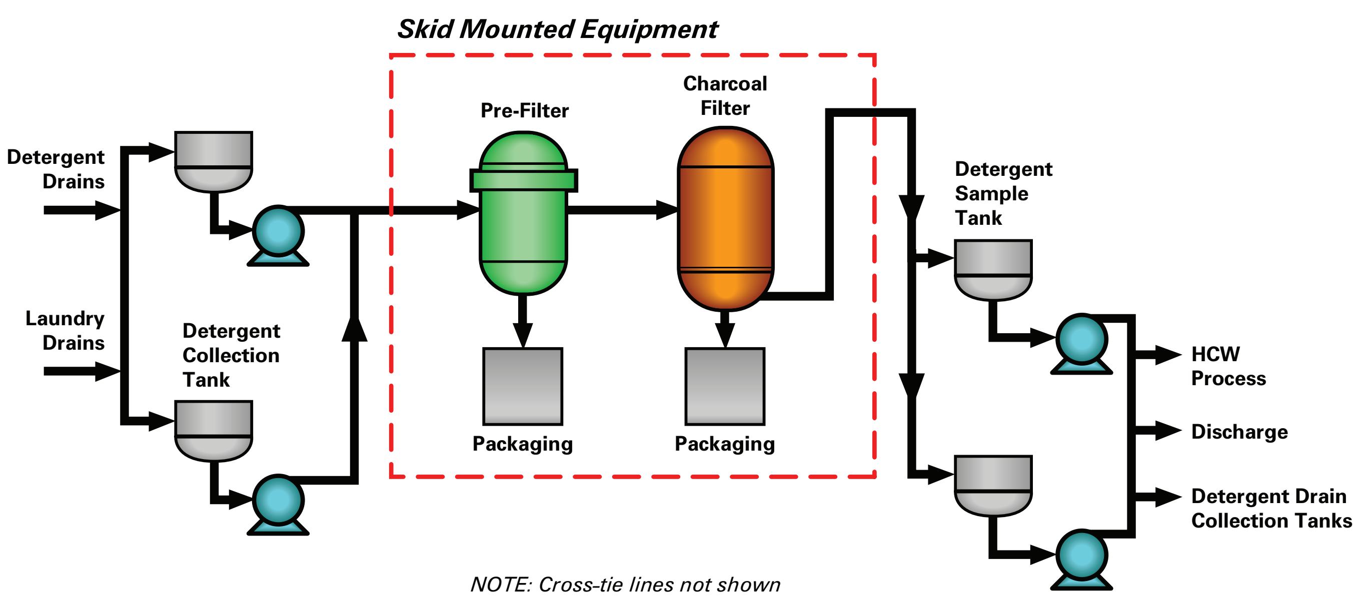

Figure 10-3. Detergent Drain Subsystem Schematic

Processing Subsystems

For Equipment Drain Processing—The equipment drain processing system utilizes filters for removing suspended solid and radioactive particulate material, and charcoal absorption for organic material removal as necessary. Backwash operation for filtration units is performed when the differential pressure across the filter exceeds a preset limit. Filtration backwash waste is discharged to a low activity phase separator or sent directly to a High Integrity Container (HIC).

The Equipment Drain Subsystem consists of a filter for removing large particles, a carbon bed for removing organics, as required, a reverse osmosis membrane for removing submicron particulates and most dissolved ionic compounds, and mixed-bed ion-exchangers for polishing any remaining dissolved ionic compounds. Exhausted resins from a mixed-bed ion-exchange unit are sluiced to the low activity spent resin holdup tank when an effluent purity parameter (such as conductivity) exceeds a preset limit or upon high differential pressure across the unit. Fine mesh strainers with backwashing connections are provided downstream of the ion-exchange vessel to prevent resin from being carried over to the sampling tanks in the event of a screen failure in the ion exchange vessel. Reverse osmosis concentrates are accumulated in the Concentrated Waste Tank to facilitate processing.

The equipment drain processing system is designed and configured for installation ease and process reconfiguration. In-plant supply and return connections from permanently installed equipment to the processing system are provided for operational flexibility.

For Floor and Chemical Drain Processing—The Floor Drain Subsystem consists of a filter for removing large particles, a carbon bed for removing organics, as required, a reverse osmosis membrane for removing submicron particulates and most dissolved ionic compounds, and mixed bed ion-exchangers for polishing any remaining dissolved ionic compounds.

Exhausted ion-exchange resins may be sluiced to the spent resin tank when a chemistry parameter (such as conductivity) exceeds a preset limit or upon high differential pressure. Fine mesh strainers with backwashing connections are provided downstream of the ion-exchange vessel to prevent resin from being carried over to the sampling tanks in the event of a screen failure in the ion exchange vessel. Reverse osmosis concentrates are accumulated in the Concentrated Waste Tank to facilitate processing.

The floor drain processing system is configured for installation ease and process reconfiguration. In plant supply and return connections from other radwaste equipment to the processing system are provided to ensure operational flexibility.

For Detergent Drain Processing—The Detergent Drain Processing Subsystem can utilize organic pretreatment to remove organics and a filter to remove suspended solids. When the differential pressure of the filter exceeds a preset value, the filter performance is rejuvenated in accordance with the design of the filter. Spent filter media are packaged as solid waste.

Offgas System

The Gaseous Waste Management or Offgas System (OGS) processes and controls the release of gaseous radioactive effluents to the site environs so as to maintain the exposure of persons outside the controlled area and personnel working near the system components to ALARA. The OGS process flow diagram is shown in Figure 10-4.

The OGS is a welded, leak-tight system with redundant active components. The OGS process equipment is housed in a reinforced-concrete structure to provide adequate shielding. Charcoal absorbers are installed in a temperature-monitored and controlled vault. The facility is located in the turbine building to minimize piping.

The main condenser evacuation system removes the noncondensable gases from the main condenser and discharges them to the OGS. The evacuation system consists of two 100% capacity, steam jet air ejectors (SJAEs) for normal station operation, and mechanical vacuum pumps for use during startup. The OGS receives air and noncondensable gases from the SJAEs and processes the effluent for the decay

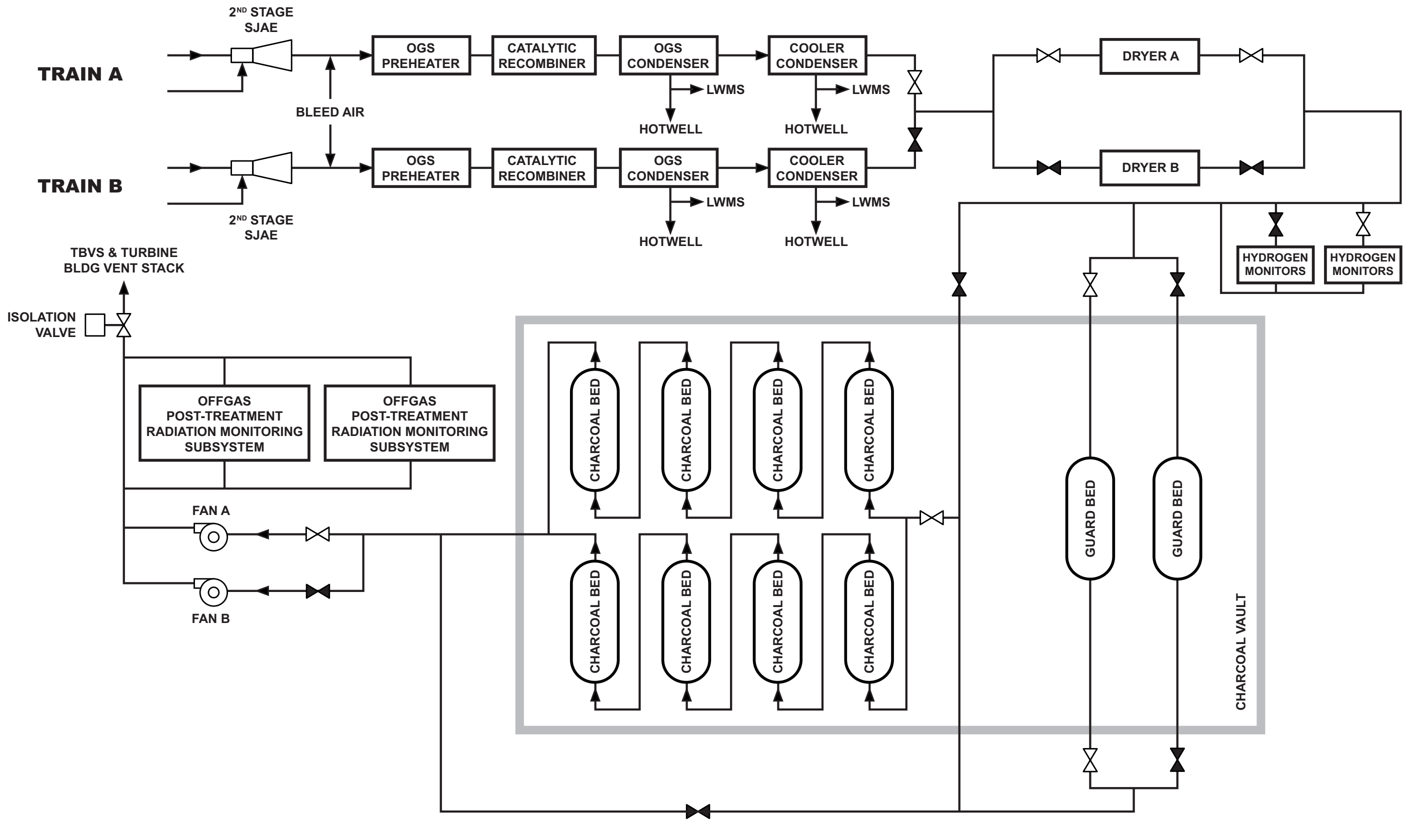

Figure 10-4. Offgas System Schematic

and/or removal of gaseous and particulate radioactive isotopes.

The OGS also reduces the possibility of an explosion from the buildup of radiolytic hydrogen and oxygen. This is accomplished by the recombination of the radiolytic hydrogen and oxygen under controlled conditions within a catalytic recombiner. This process strips the condensables and reduces the volume of gases being processed. Recombiner preheaters preheat gases to provide for efficient catalytic recombiner operation and to ensure the absence of liquid water that suppresses the activity of the recombiner catalyst.

Each recombiner is part of a preheater-recombiner-condenser process train. The preheater uses nuclear quality steam to heat the offgas process stream gases before they reach the catalyst in the recombiner. The recombined hydrogen and oxygen, in the form of superheated steam, which leaves the recombiner is then condensed (by power cycle condensate) to liquid water in the condenser, while the noncondensable gases are cooled. The condensed water in the condenser is drained to a loop seal that is connected to the main condenser hotwell. Condensed preheater steam is also drained to the above loop seal that is connected to the hotwell. No flow paths exist whereby unrecombined offgas can bypass the recombiners.

The gaseous waste stream is then further cooled by chilled water in the cooler condenser. The cooler condenser is designed to remove any condensed moisture by draining it to the main condenser.

The remaining noncondensables (principally air with traces of krypton and xenon) are passed through activated charcoal beds, which are operated at an ambient temperature and provide a holdup volume to allow time for the krypton and xenon to decay. In order to insure enough noncondensable flow, a small quantity of air is deliberately introduced into the system. After processing, the gaseous effluent is monitored and released to the environs through the turbine building stack.

Solid Radwaste Management System

The Solid Waste Management System (SWMS) is designed to control, collect, handle, process, package, and temporarily store wet and dry solid radioactive waste prior to shipment. This waste is generated as a result of normal operation and anticipated operational occurrences. The SWMS is located in the radwaste building. It consists of the following four subsystems:

- Solid waste collection subsystem
- Solid waste processing subsystem
- Dry solid waste accumulation and conditioning subsystem
- Container storage subsystem

Solid Waste Collection Subsystem

The wet solid waste collection subsystem collects spent bead resin slurry, spent charcoal media, filter and tank sludge slurry and concentrated waste into one of the six tanks in accordance with the waste characteristics (see Figures 10-5 and 10-6).

Spent bead resin sluiced from the RWCU, FAPCS, Condensate Purification System and LWMS are transferred to three spent resin tanks for radioactive decay and storage. Spent resin tanks are categorized as follows:

- High Activity Resin Holdup Tank for receiving RWCU and FAPCS spent bead resin
- Low Activity Resin Holdup Tank for receiving LWMS spent bead resin
- Condensate Resin Holdup Tank for receiving Condensate Purification System spent bead resin

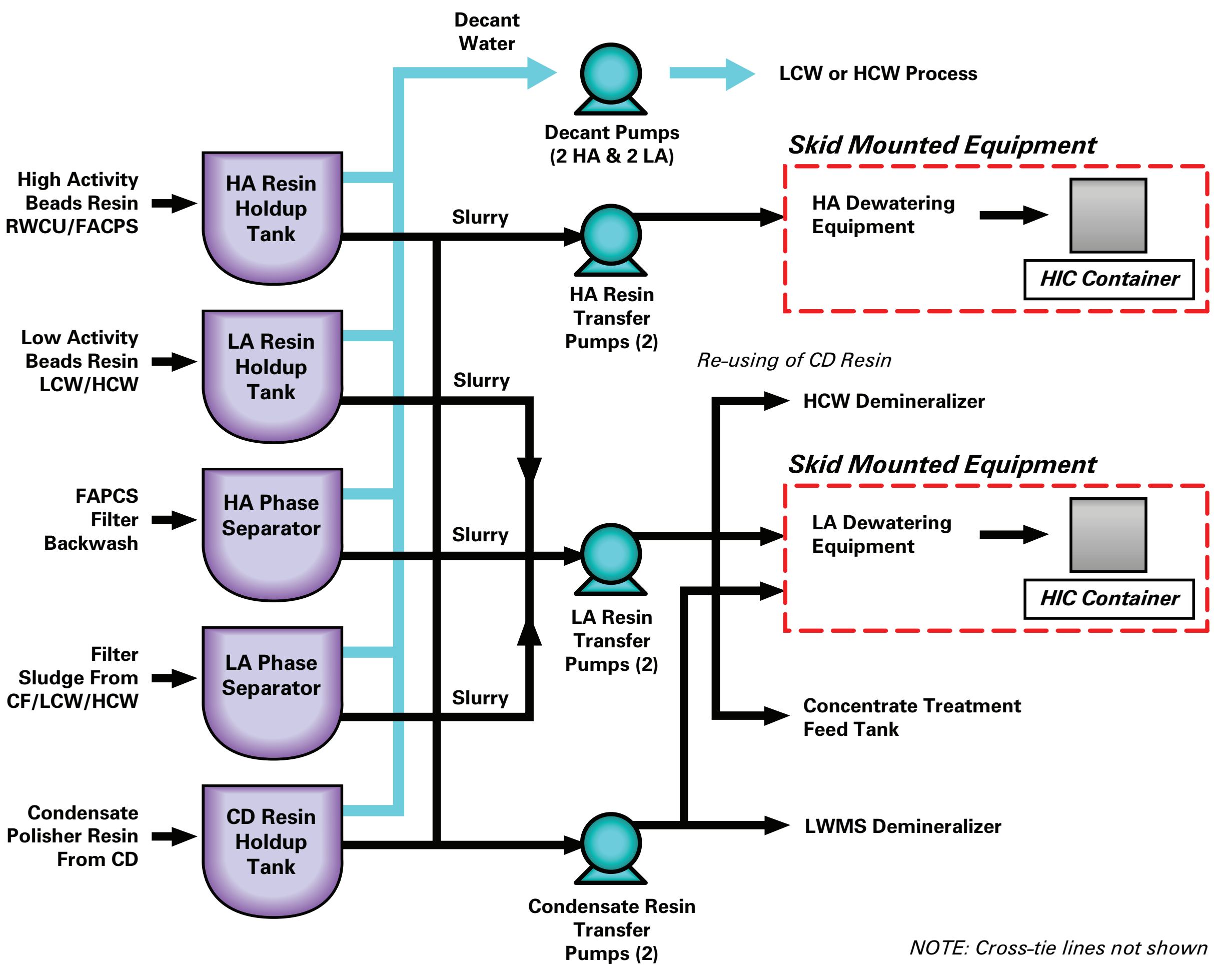

Figure 10-5. Wet Solid Waste Collection Subsystem Schematic

The capability exists to keep the higher activity resins, the lower activity resins and condensate resins in separate tanks. Excess water from a holdup tank is sent to the equipment drain collection tank or floor drain collection tank by a decant pump.

When sufficient bead resins have been collected in the high/low activity resin holdup tank, they are mixed via the high or low activity resin transfer pump and sent to the solid waste processing subsystem. When sufficient bead resins have been collected in the condensate resin holdup tank, they are mixed via the resin transfer pump and sent to the LWMS pre-treatment ion-exchanger for reuse or the solid waste processing subsystem.

A High Activity Phase Separator and a Low Activity Phase Separator receive suspended solid slurries from the FAPCS Filter, the Condensate Purification System, process filtration system of the LWMS and dewatering streams from high integrity containers (HIC). The suspended solids are allowed to settle and the residual water is transferred by the appropriate decant pump to the equipment drain collection tanks or floor drain collection tanks for further processing. When sufficient sludges have been collected in the tank, the sludges are mixed by the appropriate resin pump and sent to the solid waste processing subsystem by the appropriate resin transfer pump.

During transfer operations of the spent bead resins and the sludges, the suspended solids are kept suspended by recirculation to prevent them from agglomerating and possibly clogging lines.

The Concentrated Waste Tank receives concentrated waste from the skid-mounted reverse osmosis membranes of the LWMS. When sufficient concentrated waste has been collected in the tank, the concentrated waste is sent to the waste processing subsystem by a concentrated waste pump.

Solid Waste Processing Subsystem

The solid waste processing subsystem consists of a dewatering station for high activity spent resin

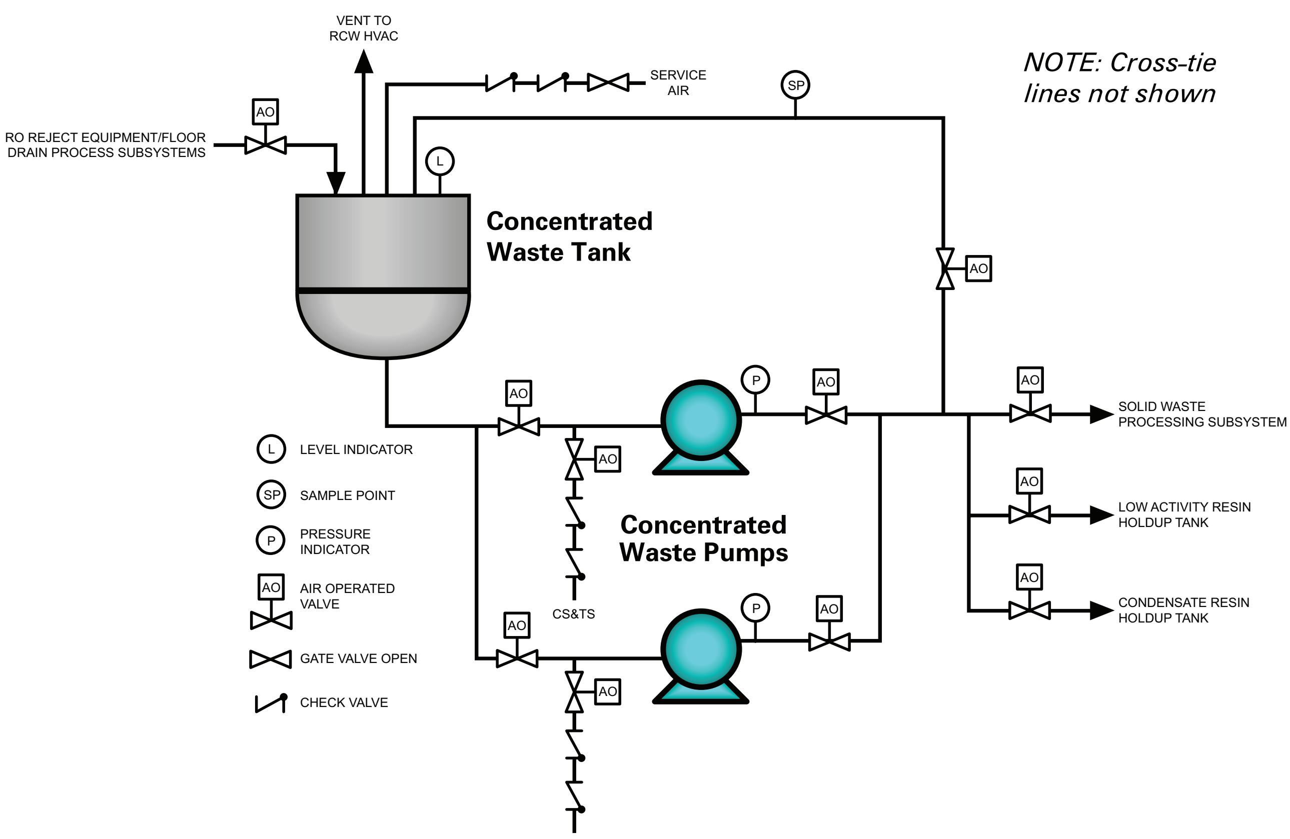

Figure 10-6. Mobile Wet Solid Waste Subsystem Schematic

and a dewatering station for low activity spent resin and sludge and concentrated waste (see Figures 10-5 and 10-6). An empty HIC is lifted off of a transport trailer and placed in each empty dewatering station. The tractor/trailer may then be released. The HIC closure lid is removed and placed in a laydown area. Spent cartridge filters may be placed in the HIC at this point, if they are not shipped in separate containers.

Next, the fill head is positioned over the HIC using a crane. The fill head includes a closed-circuit television camera for remote viewing of the fill operation. The HIC is then filled with each kind of wet solid waste. The capability to obtain samples during the fill operation is provided.

Excess water is removed from the HIC and sent by a dewatering pump to either the high or low activity phase separator or condensate spent resin tanks depending on HIC contents. Sufficient water is removed to ensure there is very little or no free standing water left in the HIC.

The fill head is then removed and placed in a laydown area. The closure head is then placed on the HIC. The HIC is provided with a passive vent to prevent pressure buildup. Radiation shielding is provided around the HIC stations.

Dry Solid Waste Accumulation and Conditioning Subsystem

Dry solid wastes consist of air filters, miscellaneous paper, rags, etc., from contaminated areas; contaminated clothing, tools, and equipment parts that cannot be effectively decontaminated; and solid laboratory wastes. The activity of much of this waste is low enough to permit handling by contact. These wastes are collected in containers located in appropriate areas throughout the plant, as dictated by the volume of wastes generated during operation and maintenance. The filled containers are sealed and moved to controlled-access enclosed areas for temporary storage (see Figure 10-7).

Most dry waste is expected to be sufficiently low in activity to permit temporary storage in unshielded, cordoned-off areas. Dry active waste will be sorted and packaged in a suitably sized container that meets DOT requirements for shipment to either an offsite processor or for ultimate disposal. The dry active waste is separated into three categories: non-contaminated wastes (clean), contaminated metal wastes, and other wastes, i.e., clothing, plastics, HEPA filters, components, etc.

In some cases, large pieces of miscellaneous waste are packed into larger boxes. This waste is stored until enough is accumulated to permit economical transportation to an offsite burial ground for final disposal. The capability exists to bring a shipping container into the truck bay during periods of peak waste generation. Bagged dry active waste can be directly loaded into the shipping container for burial or processing in offsite facilities. A truck scale is provided to ensure optimum shipping/disposal weight of the shipping container.

Cartridge filters that are not placed in HICs are placed in suitability-sized containers meeting DOT requirements.

Container Storage Subsystem

Onsite storage space for 6-months volume of packaged waste is provided. Packaged waste includes HICs, shielded filter containers, 55-gallon (200-liter) drums and other shipping containers as necessary.

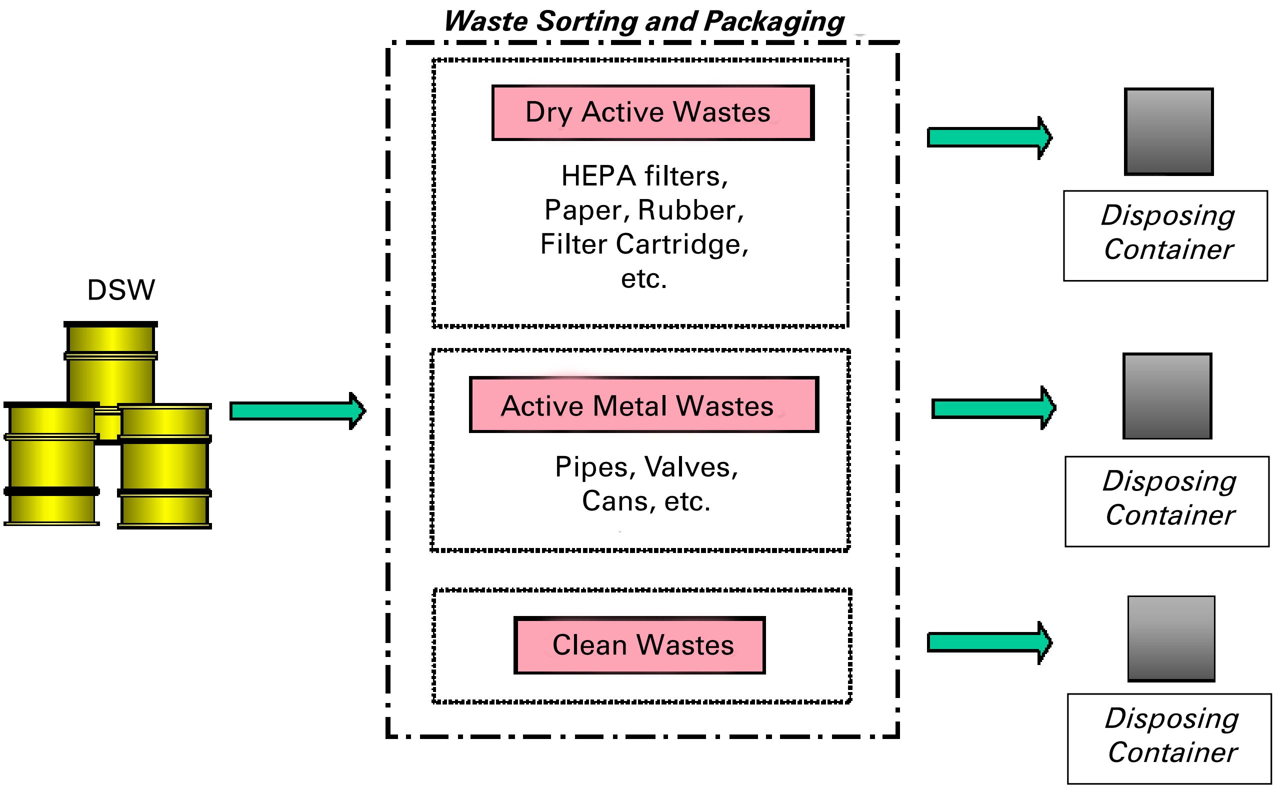

Figure 10-7. Dry Solid Waste Subsystem Schematic

Chapter 11 Safety Evaluations

Overview

The ESBWR represents a new approach to reactor safety. The use of natural circulation and passive ECCS requires a reactor vessel with a relatively large steam volume at power and a relatively large water volume when shutdown. These features permit a reactor design with a more gentle response to design basis transients and accidents. The use of passive containment cooling together with passive drywell flooding and a lower drywell core catcher also leads to a very low containment failure probability for severe accidents.

Transient Performance

Transient performance, in the safety sense, becomes translated into fuel performance and operating margins. The primary BWR measures are minimum critical power ratio (MCPR), and maximum linear heat generation rate (MLHGR). These design parameters vary, depending on the specific fuel design being used (e.g., 9x9 or 10x10). However, the ESBWR was designed to assure flexibility of use of advancing fuel technologies while maintaining significant operating margins to fuel limits (7% or more for the MCPR margin, and 10% or more for the MLHGR margin for 24-month cycles).

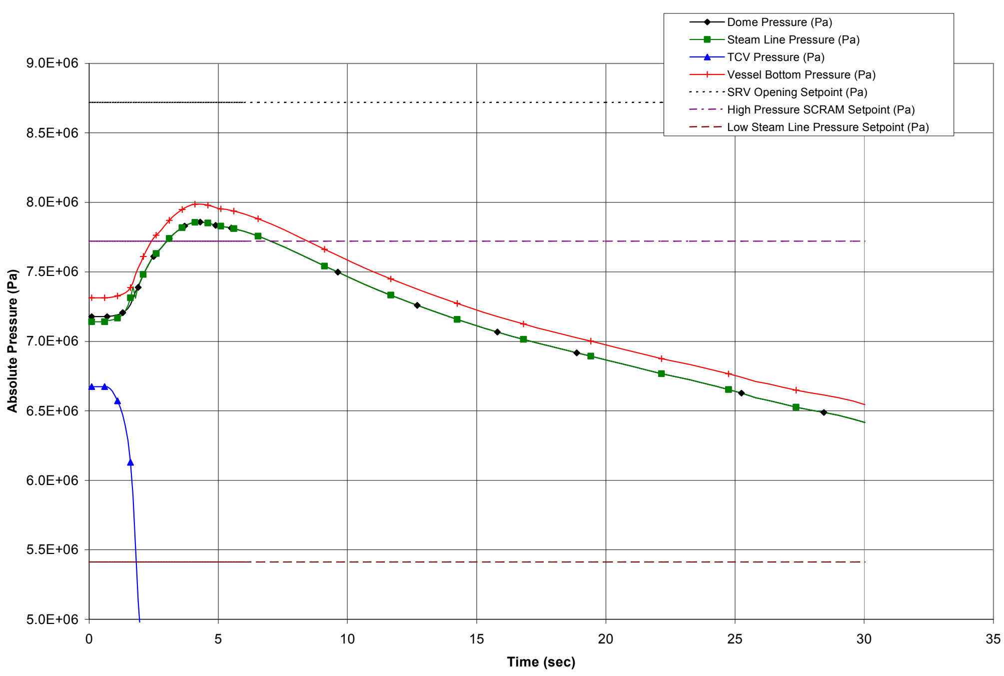

Figure 11-1. MSIV Closure Transient

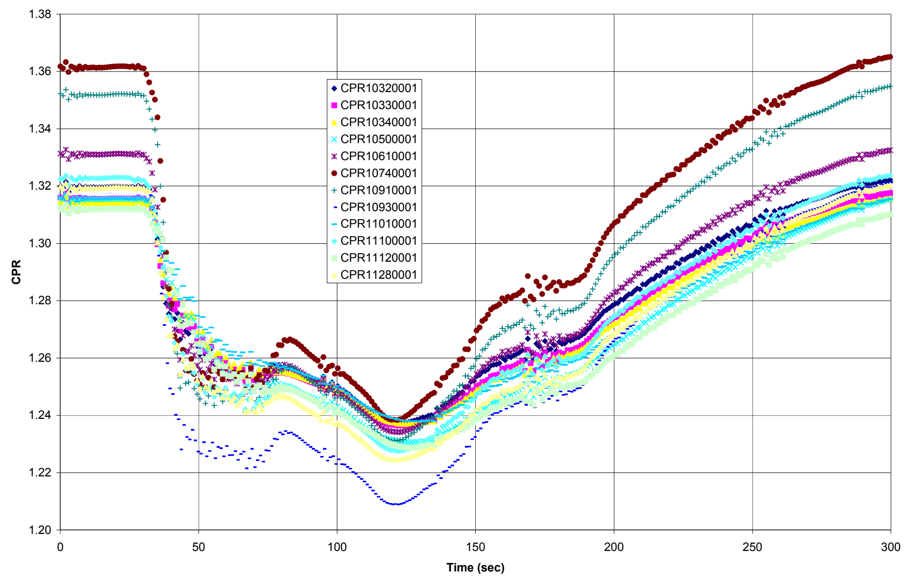

Figure 11-2. Inadvertent Isolation Condenser Initiation

Pressurization transients historically have limited BWR transient performance. With the combination of a large steam volume in the RPV (for transient pressure rise) and the use of isolation condensers (for longer term heat removal), the pressure rise is in ESBWR less than SRV setpoints, so there is no relief valve lift even for isolation transients. In addition, the transient change in MCPR (ΔMCPR) is no longer limiting. Figure 11-1 shows the ESBWR pressure response to an MSIV closure event.

All transients proceed at a slower pace than in previous BWRs. The most limiting transient for ΔMCPR is Inadvertent Isolation Condenser Initiation event, which assumes all four individual isolation condensers startup. Minimum CPR occurs approximately two minutes following event initiation. point with CPR margin restored, see Figure 11-2. The ratio ΔCPR/ICPR is 0.09 for this limiting Anticipated Operational Occurrence event.

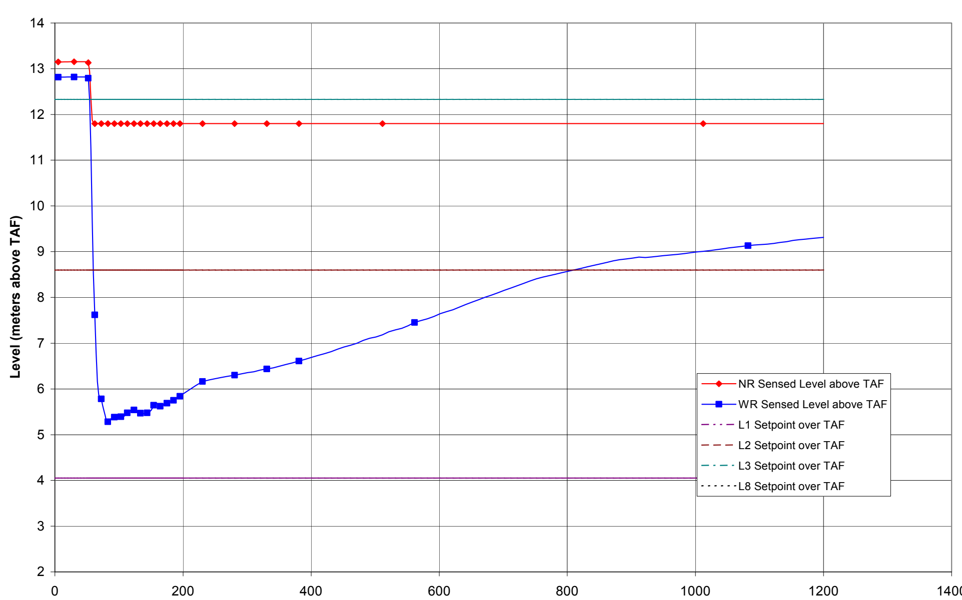

Figure 11-3. Loss of All Feedwater Flow

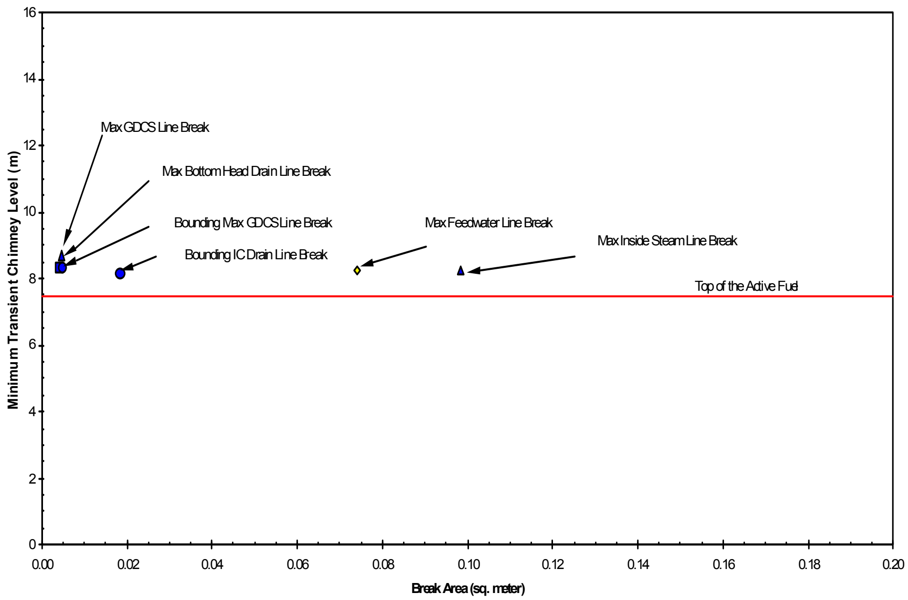

Figure 11-4. Minimum Transient Chimney Water Level vs. Break Area

Another characteristic of a large natural circulation reactor is a larger level swing upon scram due to collapsing voids in the tall chimney. The most limiting of these includes the Loss of Feedwater flow event, where the feedwater and circulating pumps are lost. Even though there is approximately an 8-m water level drop upon loss of feedwater flow, the minimum level is still 5 m above the top of active fuel (TAF). See figure 11-3.

Accident Performance

The ESBWR uses passive systems (GDCS, PCCS, ICS, and SLCS) to mitigate loss-of-coolant accidents (LOCA). More information about these systems can be found in Chapters 3 and 4. Another design feature is provided by the raised suppression pool in the containment and sufficient in containment water to assure long-term core coverage. Finally, there are no large pipes attached to the RPV below

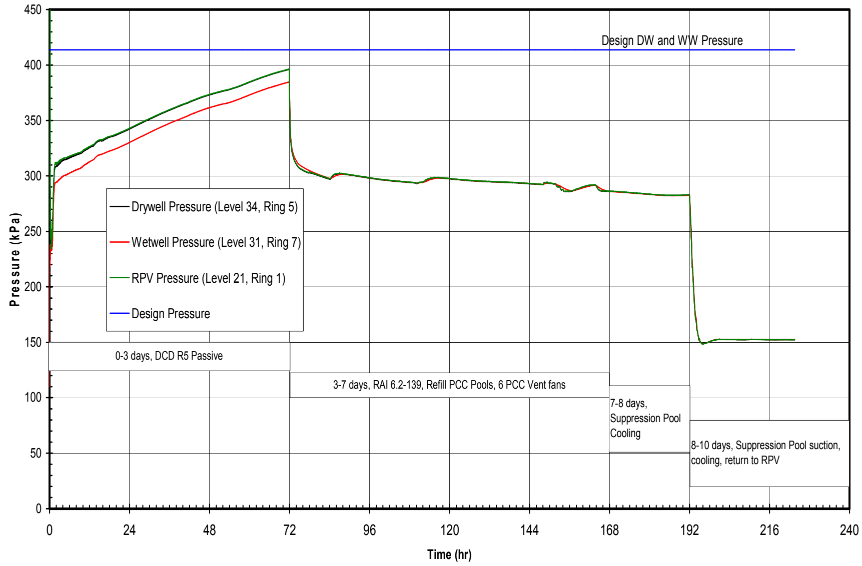

Figure 11-5. Containment Pressure Response, Including Post-LOCA Containment Cooling and Recovery

core elevation. The combination of these features assures the ESBWR has no core uncovery even for the most limiting design-basis/loss-of-coolant accident (DBA/LOCA). Figure 11-4 shows minimum chimney water level relative to top of active fuel in the short term for a spectrum of line breaks. All events are well above the top of active fuel by approximately one meter. Longer term, the water level in the RPV will be at least as high as the spillover vent in the drywell (see Chapter 8).

The ability of the PCCS to remove decay heat and maintain containment pressure within design limits is shown in Figure 11-5. Pressure is controlled for the first 3 days using the PCCS without active assistance of its vent fans. After 3 days, the vent fans are operated, powered by the ancillary diesel generators. Post-LOCA recovery continues after 7 days with suppression pool cooling by operating the FAPCS, powered by the standby diesel generators unless normal or alternate preferred offsite power is available (see Chapters 4 and 5 for details of these post-accident recovery systems).

Calculated doses for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) were done using U.S. NRC Regulatory Guide 1.183 with conservative meteorology, to bound potential sites. For example, the meteorological dispersion coefficient (χ/Q) used for EAB was 2 x 10^{-3} s/m^3, which should allow even poor meteorological sites to establish the EAB at 800 m. The calculated doses are within the regulatory limits.

Special Event Performance

Special events are those that are required by regulation to be analyzed regardless of expected frequency of occurrence. Two of the most challenging events are discussed here - Station Blackout (SBO), and Anticipated Transient Without Scram (ATWS).

Station Blackout events have historically been the most demanding for BWRs to cope with, and have usually been the dominant sequence for Severe Accidents. However, the ESBWR ICS provides means for heat removal and inventory control. Therefore the ESBWR coping time of at least 72 hours (without operator action) far exceeds the regulatory requirements.

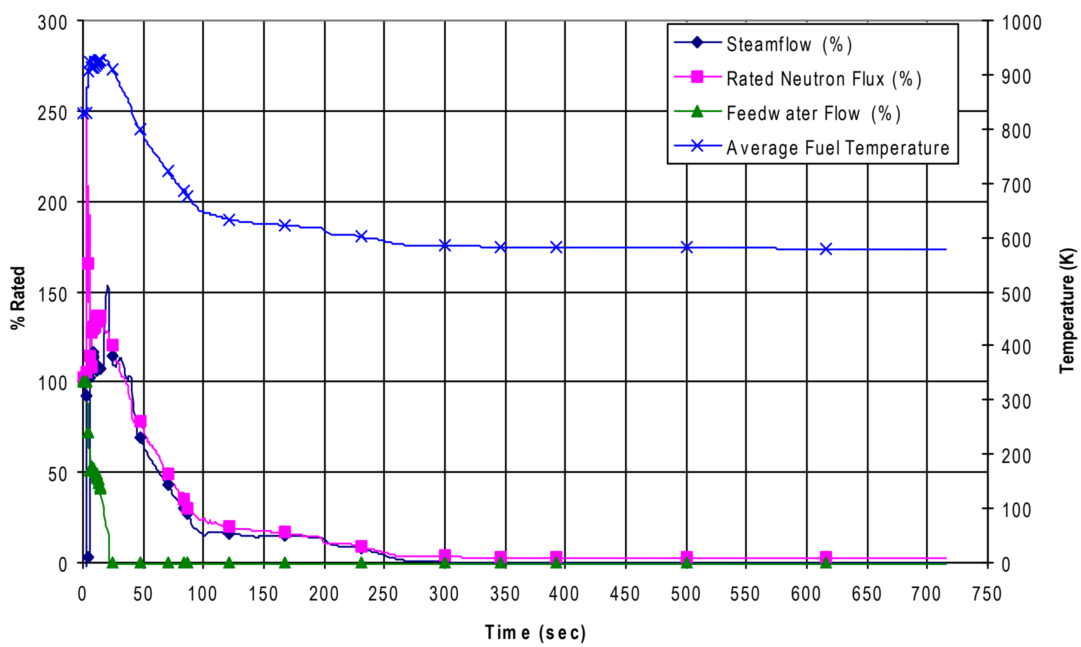

Figure 11-6. MSIV Closure ATWS - Power

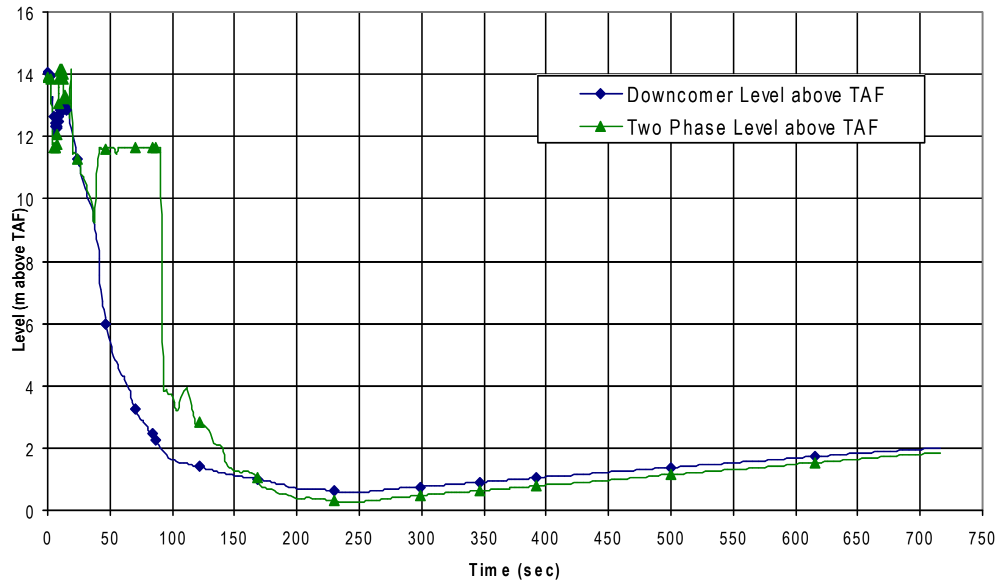

Figure 11-7. MSIV Closure ATWS – Water Level

With operation of the ICS, the containment and suppression pool pressures and temperatures are maintained within their design limits since minimal RPV leakage is expected. However, if the leakage should become significant, ADS, GDCS and PCCS are available to provide cooling.

With the adoption of FMCRDs that provide diverse electrical and hydraulic control rod insertion capability, and with automation of actions needed to mitigate ATWS events, the probability of such events leading to significant consequences has been greatly reduced for ESBWR. Nonetheless, analyses have been performed to demonstrate meeting ATWS acceptance criteria. One of the most limiting of this class of transients is the MSIV Closure ATWS because it challenges peak reactor power, minimum water level, RPV pressure, and suppression pool temperature. The response of ESBWR to an MSIV Closure ATWS shutdown using the SLCS is shown in Figures 11-6 to 11-8. The ESBWR safely mitigates this event.

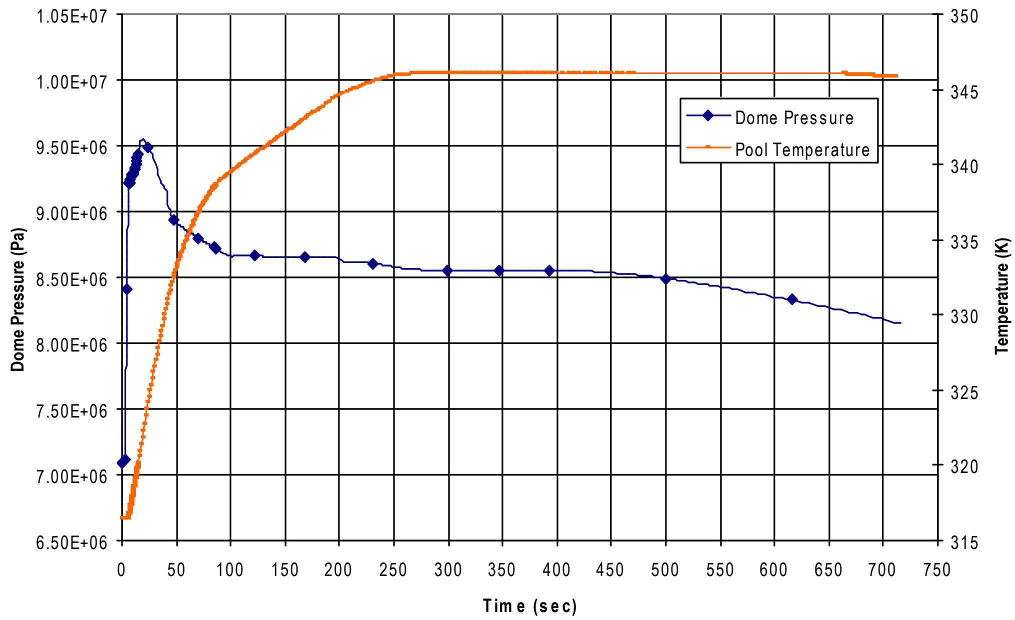

Figure 11-8. MSIV Closure ATWS – RPV Pressure, SP Temperature

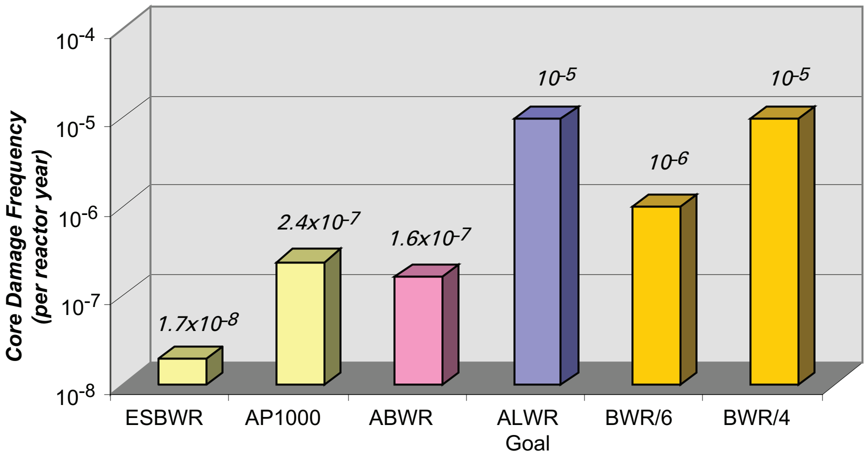

Figure 11-9. Comparison of Internal Event PRAs

ESBWR Probabilistic Risk Assessment (PRA)

PRA studies played a major role in improving the overall plant design. For example, early PRAs were used in deciding to use diverse valves in the ICS. Insights gained from the PRAs were used to improve plant technical specifications, emergency procedure guidelines, and the control room interface. The important insights from the PRA were also collected to provide input into the integrated Reliability Assurance Program. These insights will be used throughout the lifetime of the plant to ensure that plant operations maintain a high level of safety.

A comparison of the internal events PRA for the ESBWR to PRAs performed for other reactors clearly demonstrates the overall improvement in safety (Figure 11-9). The U.S. NRC risk goal for the frequency of core damage events in new reactors is 1 x 10^{-4}/(reactor year), and the Utility requirement for ALWR plants is 1 x 10^{-5}/ry. The core damage frequency (CDF) for the ESBWR was found to be approximately 1.7 x 10^{-8}/ry. This represents a factor of ten improvement compared to ABWR, and

Severe Accident Performance

Although demonstration of performance for the traditional set of design basis transients and accidents is important, in recent years regulatory emphasis has shifted toward performance for beyond design basis events, classified as severe accidents. The ESBWRs capability to prevent severe reactor accidents from capability to withstand a severe accident in the extremely unlikely event that one should occur, were evaluated with several probabilistic risk assessments (PRAs) during the design and development process. These evaluations influenced the design choices and certain design features in the final product. The final evaluation indicates that events resulting in damage to the reactor core are extremely unlikely, with low offsite doses even for severe accidents (Figure 11-12).

In the ESBWR, GEH has provided passive severe accident mitigation features to protect the containment from overpressurization and to limit the consequences to the public.

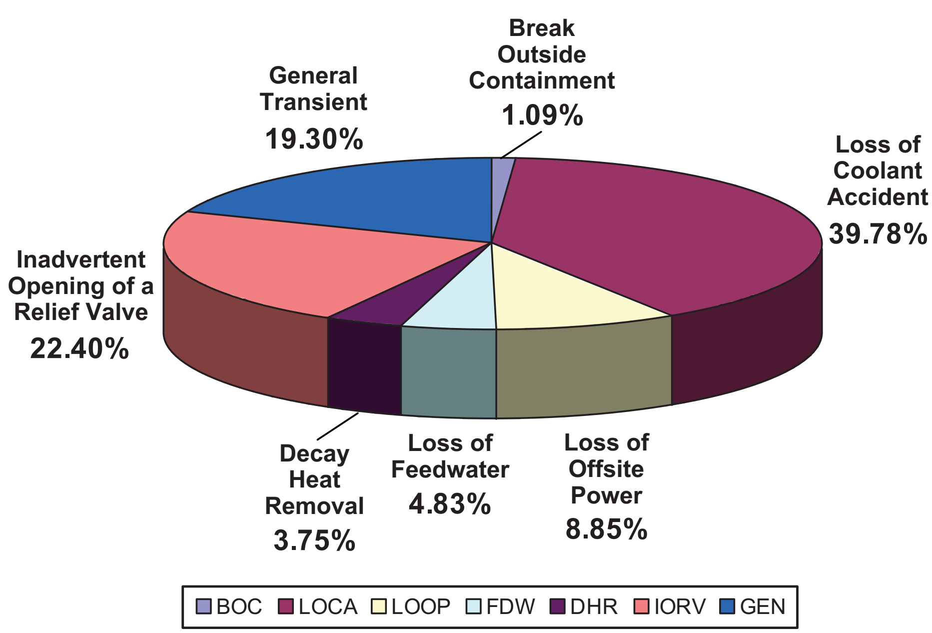

Figure 11-10. Core Damage Risk by Event

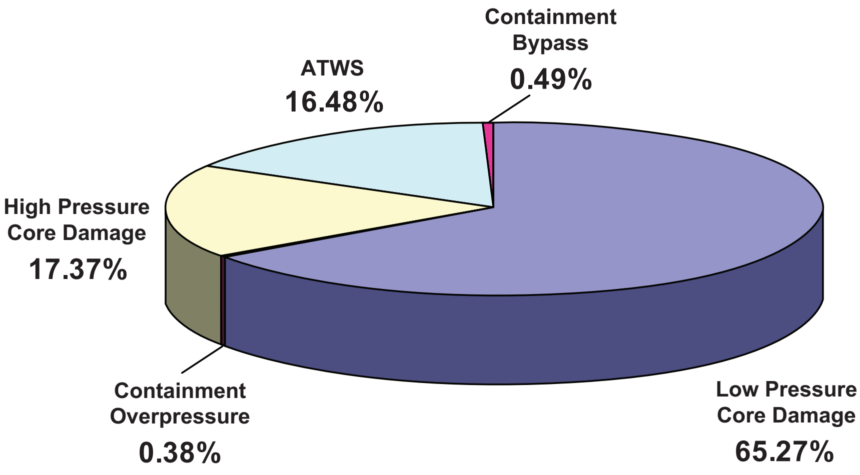

Figure 11-11. Core Damage Risk by Accident Class

a factor of 100 or more improvement compared to most operating light water reactors.

The contribution to core damage by initiating event is shown in Figure 11-10. Another perspective is given in Figure 11-11, which shows the breakdown by Accident Class. It can be seen that low-pressure core melt sequences dominate.

With low-pressure core melt sequences dominating, the Conditional Containment Failure Probability satisfies a probabilistic goal of being less than 0.1. The PCCS and BiMAC core catcher (see Chapter 8) also contribute to a low CCFP.

In addition to events at power, the risk of core damage during shutdown was also evaluated. This can occur primarily because of operator errors or small line breaks which might drain the RPV while the lower containment hatches are open. The mitigating factor in these scenarios is the significant amount of time available to correct the situation. The quantitative evaluation of the risk while in shutdown is the same (1.7×10^{-8}) as at power.

Probabilistic methods were also applied to events initiated externally (e.g., tornado, flood, fire and earthquake). The important design features to ensure plant safety for each of these events were identified in a manner similar to that for the internal events PRA.

Tornado risk: The CDF due to a tornado was found to be extremely low because, with the exception of a limited number of fail-safe safety-related instruments in the Turbine Building used by the Reactor Protection System, safety-related components are located in the Seismic Category I reinforced-concrete Reactor Building. The ESBWR internal events PRA already evaluates the Fuel Building and Control Building, probabilistically, for loss-of-offsite power due to other causes.

Flood risk: The objective of the ESBWR internal probabilistic flood analysis is to identify and provide a quantitative assessment of the core damage frequency due to internal flood events. Internal floods may be caused by large leaks due to rupture or cracking of pipes, piping components, or water containers such as storage tanks. Other possible flooding causes are the operation of fire protection equipment and human errors during maintenance.

The results of the ESBWR bounding analysis show that the CDF for internal flooding is considerably less than the total plant CDF. The risk from internal flooding is acceptably low. The following insights concerning the flooding mitigation capability of the ESBWR are identified:

- Safety system redundancy and physical separation for flooding by large water sources along with alternate safe shutdown features in buildings separated from flooding of safety systems give the ESBWR significant flooding mitigation capability
- A small number of location-specific design features must be relied on to mitigate all potential flood sources. The flood specific features are: watertight doors on the Control and Reactor Buildings, floor drains in the Reactor and Control Buildings, Circulating Water System (CIRC) pump trip and valve closure on high water level in the condenser pit
- While timely operator action can limit potential flood damage, all postulated floods can be adequately mitigated (from a risk perspective) without operator action

Fire risk: The evaluation of fires was based on the Fire-Induced Vulnerability Evaluation methodology developed by the Electric Power Research Institute (EPRI). This conservative methodology provides procedures for performing quantitative screening analyses of fire risk. The results from these conservative screening analyses show that all the screening cases analyzed have a CDF typically lower than the internal events CDF and therefore do not require a further detailed fire analysis. The following insights are provided on the fire mitigation capability of the ESBWR:

- Safety-system redundancy and physical separation by fire barriers ensure that one fire limits damage to one safety system division. PIP system commonality is limited and is only affected by a few fire areas
- Fires in the control room have the capacity to affect the execution of human actions. One feature relevant to the design is that a fire in the control room does not affect the automatic actuations of the safety systems. The remote shutdown panels allow the mitigation of any accident condition as if the operator were in the main control room

Seismic Risk: The risk of seismic events was evaluated using the seismic margins method. The ESBWR was designed for a Safe Shutdown Earthquake (SSE) of at least 0.3g. In the margins method, the margins implicit in the system designs are evaluated to determine a somewhat conservative estimate of the actual capacity of each system. Then, using fault trees and event trees similar to those developed for the internal events analysis, the system capacities are combined to determine the overall plant capacity. The ESBWR was shown to have a factor of margin at least 1.67 times the SSE of the ESBWR Certified Seismic Design Response Spectrum. This ensures that there is very little possibility of a core damage event as a result of an earthquake.

ESBWR Features to Mitigate Severe Accidents

In the event of a core damage accident, the ESBWIn the event of a core damage accident, the ES-BWR containment has been designed with specific mitigating capabilities. These capabilities not only mitigate the consequences of a severe accident but also address uncertainties in severe accident phenomena. The capabilities are listed below.

Isolation Condenser System (ICS): Although the ICS responds to transients and prevents lifts of SRVs during reactor isolation transients (see Chapter 3), the ICS pool has been sized to provide capability for approximately 72 hours. This will provide a heat sink outside of containment during SBO events.

Passive Containment Cooling System (PCCS): The PCCS heat exchangers are located directly above the containment in water pools and form part of the containment boundary. There are no valves in the system and they act totally passively to remove heat added to the containment after any accident.

Independent Water Addition: The Fire Protection System (FPS) and Fuel and Auxiliary Pools Cooling System (FAPCS) not only play an important role in preventing core damage through common lines for injection of water from the FPS water supply system into the RPV, but they are the backup source of water for flooding the lower drywell should the core become damaged and relocate into the containment (primary source is the deluge subsystem pipes of Gravity Driven Cooling System). The primary point of injection for these systems is the LPCI injection, through the feedwater pipeline, to the reactor pressure vessel. Flow can also be delivered through the drywell spray header to the drywell. The drywell spray mode of this system not only provides for debris cooling, but it is capable of directly cooling the upper drywell atmosphere and scrubbing airborne fission products.

The flow path for independent water addition draws water from the FPS tank using a dedicated FAPCS Adjustable Speed Drive Motor driven pump located in the Fire Pump Enclosure and powered by the ancillary diesel generators.

Inerted Containment: The ESBWR containment is normally inerted with nitrogen containing < 3% oxygen (see discussion of the Containment Inerting System in Chapter 5). In addition to inerting the containment, the Passive Autocatalytic Recombiner System, a nonsafety-related system, is provided as defense-in-depth against the potential buildup of combustible gases generated by the radiolytic decomposition of water post-LOCA.

Igniters in the lower drums of the PCCS condensers also recombine the hydrogen and oxygen at low concentrations during a severe accident, thereby keeping the resultant internal pressure of the PCCS condensers within acceptable limits to ensure there is no plastic deformation during a potential detonation under severe accident conditions. During the initial stages of a severe accident, there is essentially no water in the vicinity of the core, so radiolysis is greatly reduced. However, large quantities of hydrogen are released into the drywell due to metal-water reactions. The high abundance of hydrogen relative to oxygen effectively reduces the potential for detonation in the PCCS. Later in the postulated event, after the core melts through the vessel and interacts with the concrete, the deluge valves open and the core once again has the potential to resume radiolysis. Thereafter, relative concentrations of hydrogen and oxygen trend closer to a stoichiometric ratio at pressures much higher than during a DBA. The igniters are nonsafety-related and are activated by the existing GDCS deluges (BiMAC) control system implemented in a nonsafety-related technology programmable logic controller.

Basemat Internal Melt Arrest and Coolability Device (BiMAC): The ESBWR design uses a passively-cooled boundary that is impenetrable by the core debris in whatever configuration it could possibly exist on the lower drywell (LDW) floor. For ex-vessel implementation, this boundary is provided by a series of side-by-side inclined pipes, forming a jacket which can be effectively and pas-

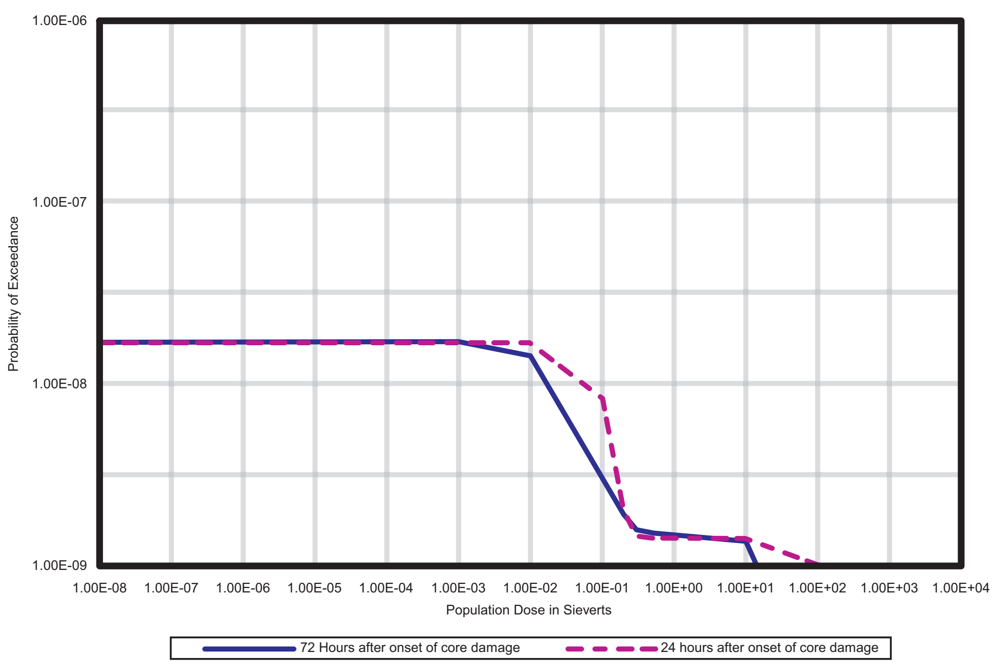

Figure 11-12. ESBWR TEDE Dose at 800 meters

sively cooled by natural circulation when subjected to thermal loading on any portion(s) of it. Water is supplied to this device from the GDCS pools via a set of squib-valve-activated deluge lines. The timing and flows are such that: (a) cooling becomes available immediately upon actuation, and (b) the chance of flooding the LDW prematurely, to the extent that this opens up a vulnerability to steam explosions, is very remote. The jacket is buried inside the concrete basemat and would be called into action only in the event that some or all of the core debris on top is non-coolable. More details can be found in Chapter 8.

Analyses have shown that the containment will not fail by Basemat melt-through or by overpressurization as long as the BiMAC functions.

Manual Containment Overpressure Protection (MCOPS): If an accident occurs that increases containment pressure to a point where containment integrity is threatened, this pressure will be relieved through a line connected to the wetwell atmosphere by opening the wetwell atmosphere to the plant stack via air-operated valves that are opened by manual operator action. Providing a relief path from the wetwell airspace precludes an uncontrolled containment failure. Directing the flow to the stack provides a monitored, elevated release. Relieving pressure from the wetwell, as opposed to the drywell, takes advantage of the decontamination factor provided by the suppression pool.

Protection of the Public

The low core damage frequency combined with low failure probability of the containment leads to very low offsite doses, even after severe accidents. Figure 11-12 shows the offsite dose at 800 m (0.5 mile) as a function of probability for a nominal U.S. site. It can be seen that large releases do not occur even at a one-in-a-billion probability per year.

Appendix Key Design Characteristics A

This appendix lists key design characteristics for the ESBWR, using the standard design licensed in the U.S. as a reference. Further details can be obtained from the ESBWR Design Control Document, Tier 2.

Overall Design	
Site Envelope	
Safe shutdown earthquake, g	0.3 envelope
Wind design, km/h	242
Maximum tornado, km/h	531
Max dry bulb/wet bulb ambient temperature, °C	47.2/27.8
Thermal and Hydraulic	
Rated Power, MWt	4500
Generator Output, MWe	1600
Steam flow rate, Mkg/h	8.76
Core coolant flow rate, Mkg/h	34.5
System operating pressure, MPa	7.17
Average core power density, kW/l	54.3
Maximum linear heat generation rate, kW/m	44.0
Average linear heat generation rate, kW/m	15.1
Minimum critical power ratio (MCPR)	1.4 - 1.5*
Core average exit quality, %	25
Feedwater temperature, °C	215.6

* Depending on the cycle length

Core Design	
Fuel Assembly	
Number of fuel assemblies	1132
Fuel rod array	10 x 10
Overall length, cm	379
Weight of UO_2 per assembly, kg	163
Number of fuel rods per assembly	92
Rod diameter, cm	1.026
Cladding material	Zircaloy-2
Fuel Channel	
Thickness corner/wall, mm	3.05/1.91
Dimensions, cm	14 X 14
Material	Zircaloy-2
Reactor Control System	
Method of variation of reactor power	Moveable control rods and Feedwater temperature
Number of control rods	269
Shape of control rods	Cruciform
Pitch of control rods, cm	31
Type of control rod drive	Bottom entry electric hydraulic fine motion
Rod step size, mm	36.5
Number of hydraulic accumulators	135
Hydraulic scram speed, sec to 60% insert	1.15
Electric drive speed, mm/sec	30
Type of temporary reactivity control	Burnable poison; gadolinia urania fuel rods
High pressure coolant injection 1/2 pumps, m^3/h	118/235
Incore Neutron Instrumentation	
Total number of LPRM detectors	256
Number of incore LPRM penetrations	64
Number of LPRM detectors per penetration	4
Number of SRNM penetrations	12

Reactor Vessel and Internals	
Reactor Vessel	
Material	Low-alloy steel/ stainless and Ni-Cr-Fe alloy clad
Design pressure, MPag	8.62
Inside diameter, m	7.1
Inside height, m	27.6
Steam Separators and Dryers	
Separator type	AS-2B
Number of separators	379
Dryer type	Chevron
Main Steam	
Number of steam lines	4
Diameter of steam lines, cm	70
Number of safety/relief valves	18
Number of depressurization valves	8
Isolation Condenser	
Number of loops	4
Capacity of each loop, MWt	34
Number of safety/relief valves	18

Emergency Core Cooling	
Gravity Driven Core Cooling	
Number of loops	4
Number of pumps	0
Flow rate per loop, m^3/h	500*
Automatic Depressurization	
Number of relief valves	10
Number of depressurization valves	8
Passive Containment Cooling System	
Number of loops	6
Heat removal duty per loop, MWt	7.8
Standby Liquid Control	
Number of accumulators	2
B10 enrichment, %	94
Capacity per accumulator, m^3	7.8
Initial flow rate per accumulator, m^3/h	66

* At runout

Containment	
Primary	
Type	Pressure suppression
Construction	Reinforced concrete with steel liner, steel structure
Drywell	Concrete cylinder
Wetwell	Concrete cylinder
Design pressure, MPaG	0.31
Design leak rate, % free volume/day	0.35*
Drywell free volume, m^3	7206
Wetwell free volume, m^3	5467
Suppression pool water volume, m^3	4383
Number of vertical vents	12
Vertical vent diameter, m	1.2
Number of horizontal vents per vertical vent	3
Horizontal vent diameter, m	0.7
Reactor Building	
Type	Low leakage
Construction	Reinforced concrete/steel
Design in leakage rate at 6.4 mm water, % free volume/day	50

* Excluding MSIV leakage

Auxiliary Systems	
Reactor Water Cleanup/Shutdown Cooling	
Number of trains	2
Number of pumps per train high/low capacity	1/1
Type	Canned rotor
Flow rate per train (cleanup mode), m^3/h/ % of feedwater	116/1
No. of regenerative heat exchangers per train	2
No. of non-regenerative heat exchangers per train	2
Return water temperature (cleanup mode), °C	227
Flow rate (shutdown cooling), m^3/h	1365
Heat removal duty (shutdown cooling), MWt	55.4
Fuel and Auxiliary Pools Cooling	
Number of trains	2
Number of pumps/train	1
Flow rate per pump, m^3/h	284
Number of heat exchangers/train	1
Total heat removal capability, MWt	16.6
Reactor Component Cooling Water	
Number of trains	2
Capacity of each train, %	100
Number of pumps per train	3
Number of heat exchangers per train	3
Flow rate per loop (normal), m^3/h	1250
Heat removal duty (normal), MWt	32.5
Flow rate per loop (shutdown), m^3/h	2700
Heat removal duty (shutdown), MWt	88
Plant Service Water	
Number of trains	2
Capacity of each train, %	100
Number of pumps per train	2
Flow rate per loop (normal), m^3/h	9072
Drywell Cooling	
Number of trains	2
Flow rate per train, m^3/h	72800
Number of fans per train	4
Heat removal duty per train, MWt	2.0

Appendix B
Frequently Asked Questions

What proof is there that natural circulation works in such a large reactor?

History

Natural circulation in Boiling Water Reactors (BWR) is a proven technology. Some of the early GE BWRs employed natural circulation. These were small plants (e.g., Dodewaard at 183 MWt and Humboldt Bay at 165 MWt), but they clearly demonstrated the feasibility of the BWR natural circulation and provided valuable operating data and experience. GE moved to forced circulation plants to achieve higher power ratings in a compact pressure vessel. Pressure vessel fabrication capability at the time was a factor in this decision. Now, after several decades, GEH is returning to natural circulation for the ESBWR.

Natural circulation provides major simplification by removal of the recirculation pumps and associated piping, heat exchangers and controls. It is also synergistic with two other requirements that GEH considered to be important in the design of a new reactor: large safety margins with a very reliable passive Emergency Core Cooling System (ECCS) and avoidance of safety/relief valve (SRV) opening for pressurization transients such as turbine trips or main steam line isolation events. Both of these features need a tall pressure vessel with large water volumes. The tall vessel leads to enhanced natural circulation flow, so the natural circulation capability comes with no additional cost.

Evolutionary Design

ESBWR builds on the design features of operating BWRs. Figure B-1 shows a cutaway of the ESBWR reactor pressure vessel. Most components in the ESBWR are standard BWR components that have been operating in the field for many years (steam separators, control rods and guide tubes, core support structure, etc.). The main difference is the taller reactor vessel with the addition of a partitioned chimney above the core and a correspondingly taller downcomer annulus. The fluid in the taller downcomer provides the additional driving head for natural circulation flow through the core, as well as a large water inventory for a Loss-of-Coolant Ac-

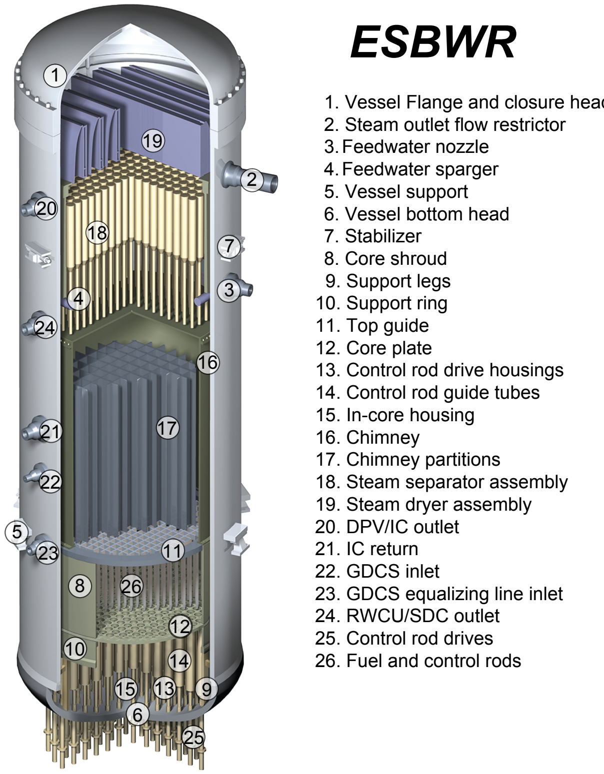

Figure B-1. Cut-away of ESBWR Reactor Assembly cident (LOCA). Steam in the chimney also provides a cushion to dampen void collapse in the core during pressurization transients, leading to a softer response with no SRV discharges.

Figure B-2 shows a view of the partitioned chimney and its layout above the core. The core consists of conventional BWR fuel bundles, shortened from 12 ft to 10 ft to improve pressure drop and stability characteristics. The absence of hardware in the downcomer (jet pumps or internal pumps) reduces flow losses and further enhances natural circulation. Figure B-3 shows how the taller, open downcomer and reduced core resistance lead to a great enhancement of natural circulation flow in the ESBWR relative to operating BWRs.

Operating Experience

Valuable operating experience was gained from the early natural circulation BWRs. It was demonstrated that BWRs could operate in natural circulation without problems. The plants were extremely stable and presented no unusual characteristics relative to noise in the instrumentation. Power was raised by control rod withdrawal. The ESBWR will also adjust output using control rods, but with electrically driven control rod drives that move slower and have finer positioning capability than the locking piston design for conventional BWRs.

Present-day BWRs can operate at about 50% of rated power in natural circulation; however, stability considerations prevent steady operation in this region. There have been recirculation pump trip events in operating plants, which led to a natural circulation state at around 50% of rated power. The operating conditions in present day BWRs (power, flow, power distribution) in natural circulation following the pump trip were well predicted by the calculational models used for ESBWR performance analysis.

The operating parameters for the ESBWR, such as the power density, steam quality, void fraction, and void coefficient are within the range of operating plant data. Figure B-4 shows a comparison of the ESBWR power-flow operating map with those of operating BWRs. This figure is based on the power per bundle and flow per bundle so that a meaningful comparison can be made. The power per bundle and flow per bundle for the ESBWR are both lower than for a modern jet pump plant at rated operating conditions, but the ratio of power to flow is similar to that for an uprated BWR at Maximum Extended Load Line Limit Analysis-Plus (MELLLA+) conditions. This means that the core exit steam quality (ratio of steam flow to core flow) is also similar.

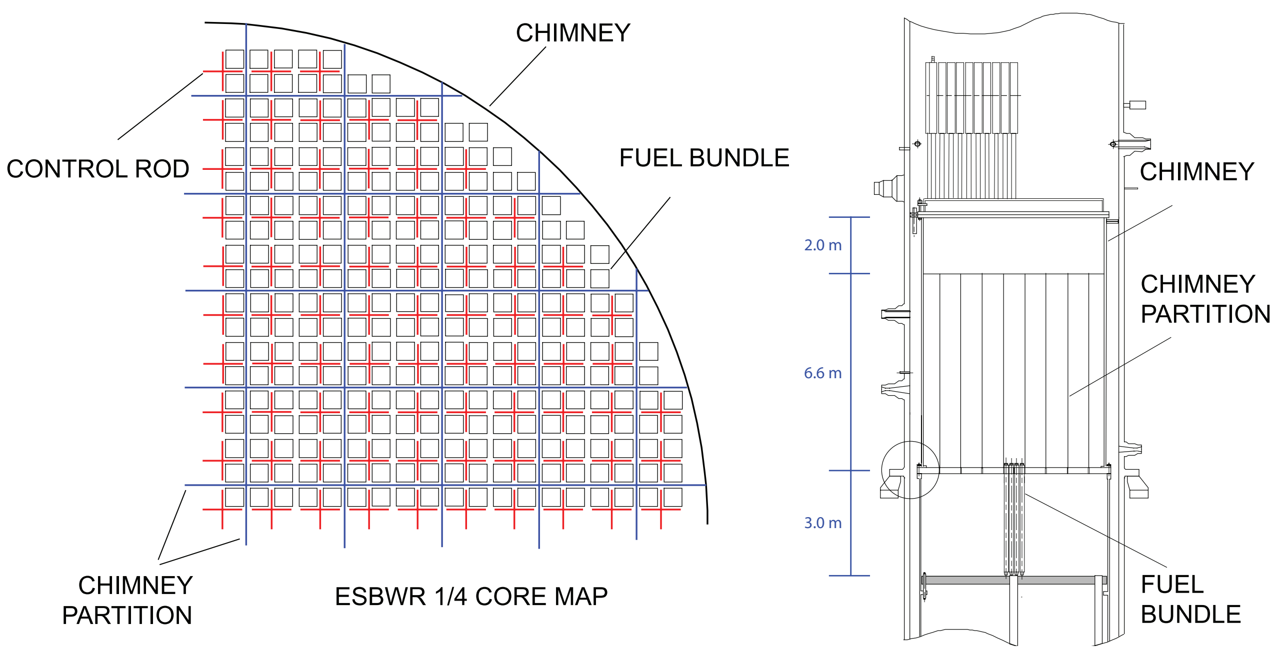

Figure B-2. ESBWR Core and Chimney

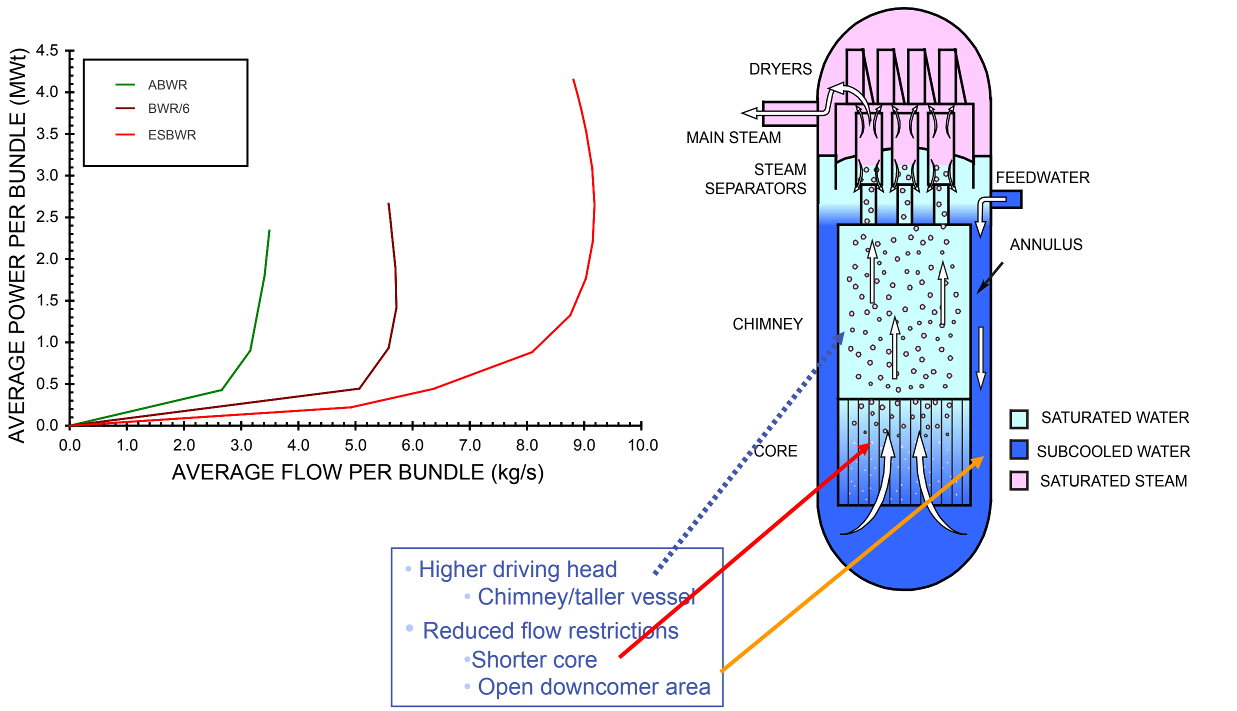

Figure B-3. ESBWR Enhanced Natural Circulation

Test Data and Code Validation

For analysis of the ESBWR, GEH uses the state-of-the-art TRACG code. GEH has invested over a hundred man-years in the development and validation of this technology, which originated in the national laboratories. A systematic strategy was adopted, using separate effects tests, component performance tests, integral system tests, and BWR data to validate TRACG.

Natural circulation flow in ESBWR is driven by the difference in the static head between the down-comer annulus outside the core shroud and the static head in the core and chimney inside the core shroud. The flow is governed by the flow loop losses, primarily in the core and separators. There is a large database for the pressure drop in the fuel channels and separators. Extensive data are also available for the void fraction in the fuel bundles.

Additionally, the test database was augmented for the chimney region by tests performed at Ontario Hydro. These tests measured the void fraction in pipes with diameter and height similar to a cell of the partitioned chimney. These data were obtained at pressures, flow rates, and void fractions representative of ESBWR operation. All these test data have been used to validate the TRACG code. Figure B-5 shows a schematic of the Ontario Hydro test facility and a comparison of the measured and calculated void fraction in a 0.53-m-diameter pipe. A chimney cell in the ESBWR has a similar hydraulic diameter (0.6-m). The good agreement with data provides confidence in the ability of TRACG to correctly calculate the flow regimes and void fractions inside the partitioned chimney cells.

Uncertainties in the calculation of the natural circulation flow are small, and can readily be accommodated in the design process. The uncertainties in parameters that govern the natural circulation flow, such as the core frictional losses, separator fictional losses and chimney void fraction, were statistically combined through a Monte Carlo process, resulting in an overall uncertainty (1σ) of approximately 3% in the calculation of core flow. Calculations of natural circulation flow for operating plants following pump trips (Oyster Creek, Hatch, LaSalle) confirm the accuracy of the TRACG predictions. Figure B-6 shows a comparison of measured and calculated flows for the LaSalle plant following a trip of the recirculation pumps. The magnitude of the flow rate is seen to be

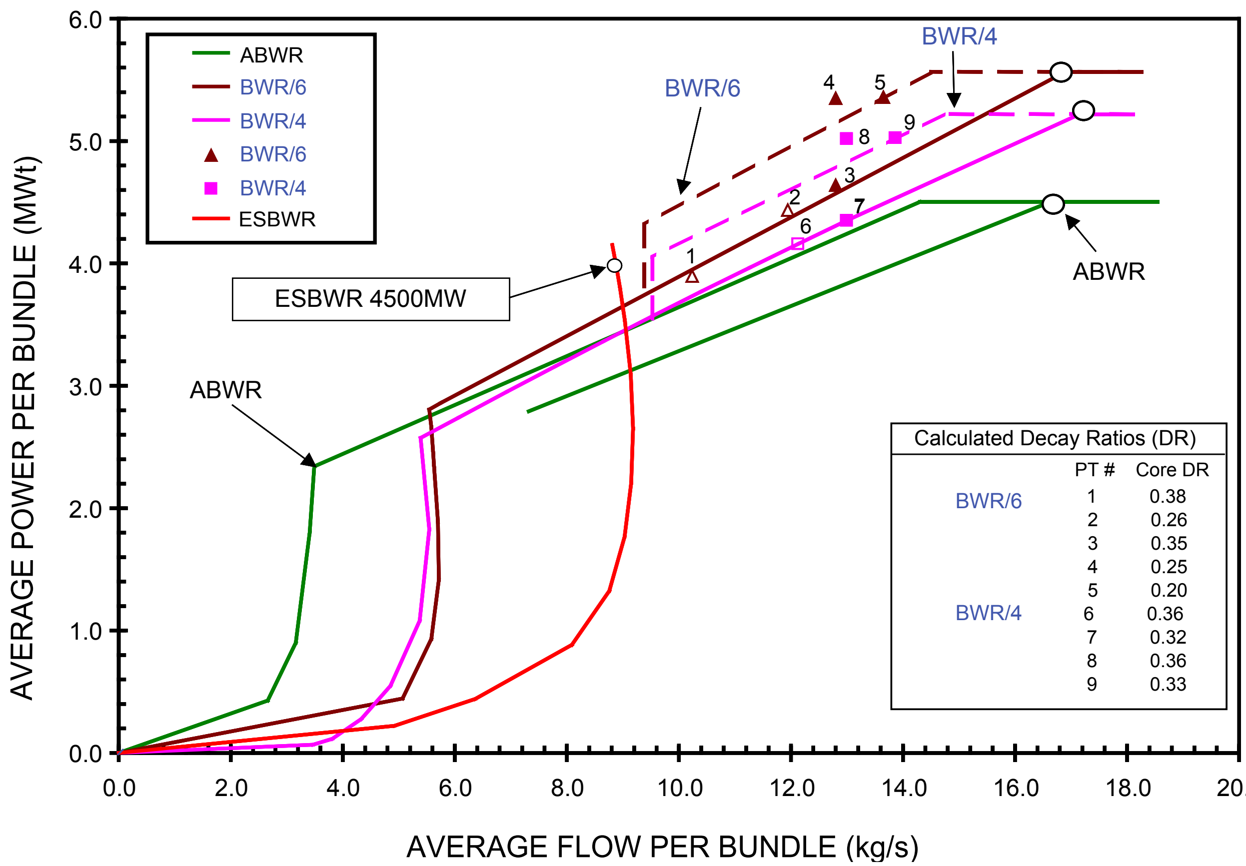

Figure B-4. Comparison of ESBWR Operating Map with Operating BWRs

calculated very accurately.

Stability data have been obtained at test facilities and in operating BWRs. The phenomena that govern stability are the same in the ESBWR and operating BWRs. The FRIGG facility in Sweden tested hydrodynamic stability performance; the SIRIUS test facility in Japan was scaled to SBWR and investigated stability under startup conditions as well as higher pressures. Figure B-7 shows a comparison of the measured and calculated oscillations in the test facility at 7.2 MPa and the measured and calculated stability maps (regions of instability). TRACG is able to capture the characteristics of these oscillations as well as the onset of instability. Note that the region of instability is far removed from the ESBWR operating state. Operating plant data from LaSalle, Leibstadt, Forsmark, Cofrentes, Nine Mile Point 2 and Peach Bottom 2 have been used to benchmark TRACG predictions of stability performance. As an example, Figure B-8 shows a comparison of the regional oscillation profile seen in test data from a European BWR with the corresponding TRACG calculation. The agreement is excellent. The shape of the oscillation profile corresponds closely to the shape of the first azimuthal harmonics of the core neutronics.

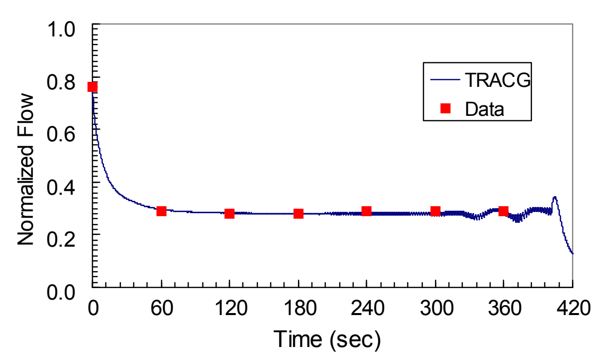

Figure B-6. Comparison of BWR Natural Circulation Flow Following Pump Trip to TRACG

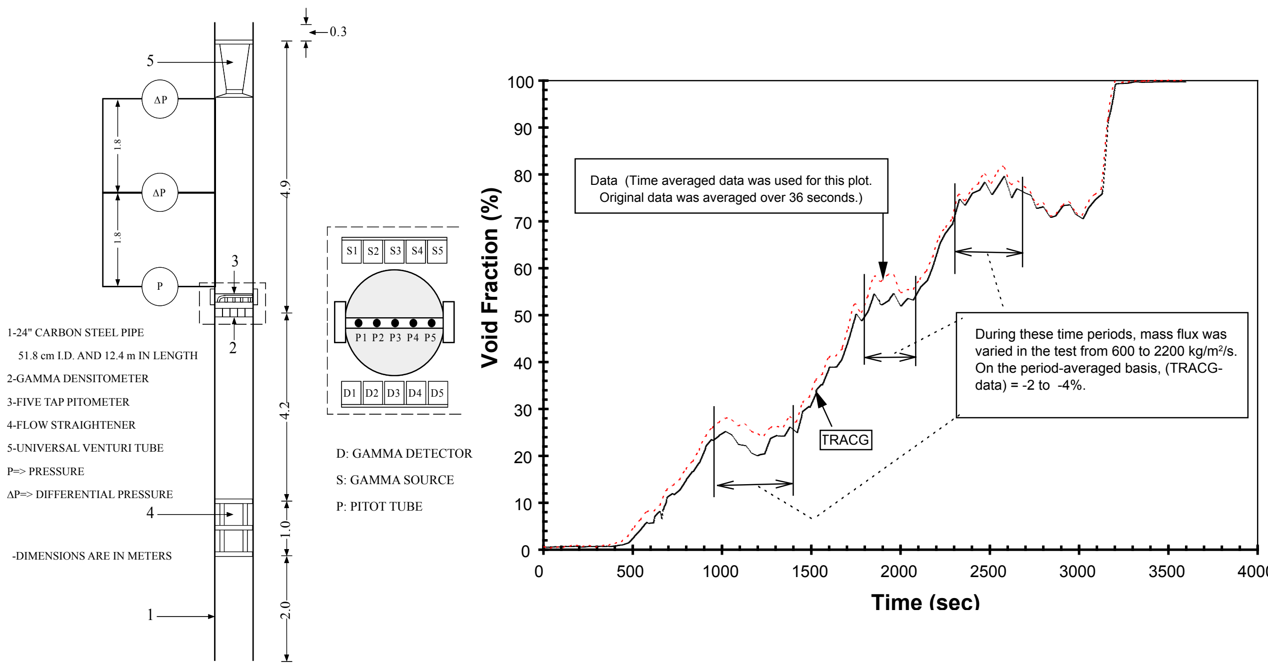

Figure B-5. Comparison of Ontario Hydro Test Data to TRACG

The NRC and ACRS have reviewed the test database and TRACG validation, and have concluded that additional tests are not required. The NRC staff has also completed a favorable Safety Evaluation of the applicability of TRACG for analyzing ESBWR stability.

ESBWR Stability

The enhanced natural circulation flow rate in the ESBWR greatly improves stability performance relative to operating BWRs at natural circulation conditions. Figure B-4 shows that the rated operating conditions for the ESBWR are closer to an uprated BWR in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) region, rather than at natural circulation.

Additionally, two other factors improve ESBWR stability significantly: the ratio of single-phase pressure drop (stabilizing) to two-phase pressure drop (destabilizing) is higher for the shorter ESBWR fuel; and the ratio of fuel time constant to the time period for a stability resonance is higher. This reduces the destabilizing neutronic feedback resulting from void perturbations. The net result is a very stable reactor that easily meets the very conservative design/licensing criteria. Stability performance is measured in terms of decay ratios for the channel, core-wide and regional stability modes. A lower decay ratio implies a more stable plant. Table B-1 compares the calculated decay ratio for the ESBWR for these modes of stability with the licensing limits of 0.8 for each stability mode. The table shows that even at the 2 σ (95% confidence) level, there is substantial margin to the stability limits. At power levels lower than rated, ESBWR stability improves further.

	Decay Ratio – Baseline Result	**Decay Ratio – One Sided Upper Tolerance Limit (95/95)**	**Decay Ratio - Design Limits**
Channel	0.23	0.36	0.8
Core	0.33*	0.50	0.8
Regional	0.60	0.80	0.8

* Core decay ratio result is for a cycle exposure near Peak Hot Excess consistent with uncertainty analysis exposure.

Table B-1. ESBWR Stability Margins

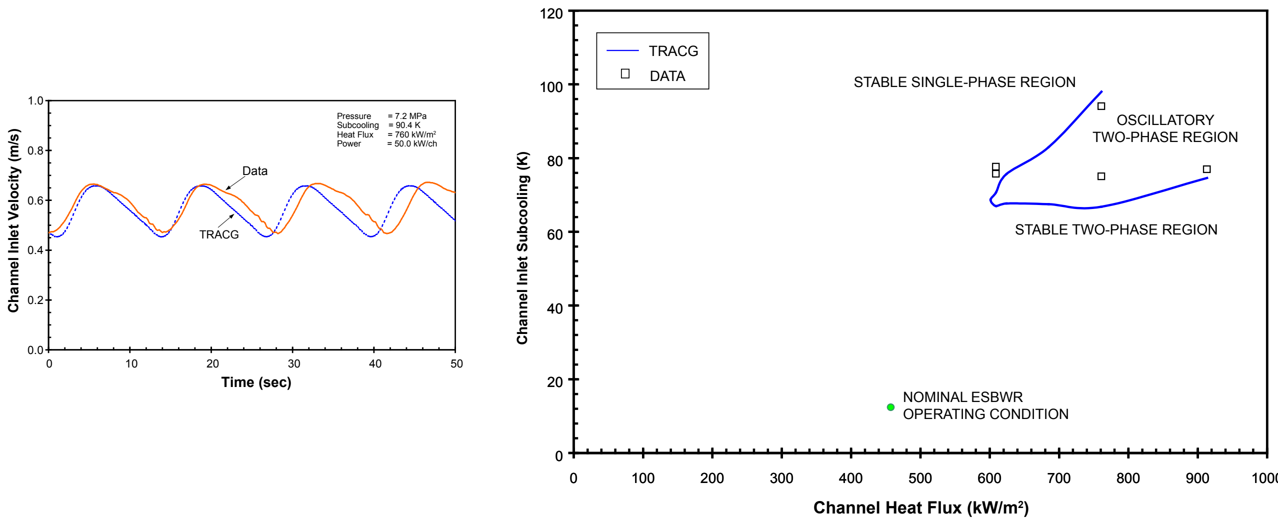

Figure B-7. Comparison of SIRIUS Test Data to TRACG

Plant Startup and Load Following

The startup procedure for ESBWR will follow the established process at Dodewaard. The Dodewaard plant operated through twenty-three (23) cycles for 30 years with no problems during startup. TRACG calculations for Dodewaard startup 22 are shown in Figure B-9. TRACG captured the main trends of the startup sequence. TRACG calculated some noise in the downcomer flow at the initiation of voiding in the chimney. This occurs at very low power levels, before the start of boiling in the core. However, this phenomenon was not noticeable in the measurements of the downcomer flow or on the in-core flux instrumentation. TRACG may be magnifying the flow noise in its calculations. In any event, at the low power levels, no thermal margins are approached. TRACG simulations of ESBWR startup demonstrate that the plant can be started up and reach rated conditions without difficulty, while maintaining large margins to thermal limits.

Because of its size, ESBWR will not normally be operated in a load-follow mode. However, changes in core power can be readily accomplished through movement of the Fine Motion Control Rod Drives (FMCRD). The FMCRDs can accommodate a duty corresponding to daily load-following cycles for 10 years.

Natural Circulation Benefits

In summary, natural circulation is a proven technology that provides numerous benefits. Natural circulation allows for the elimination of several systems, including recirculation pumps and associated piping, valves, heat exchangers, motors, adjustable speed drives and controllers. The larger RPV employed for natural circulation provides synergy with the use of passive ECCS and improves the response to operational transients and increased safety margins. Flow transients resulting from recirculation pump anomalies are not present; i.e., no runbacks or trips that would challenge stability.

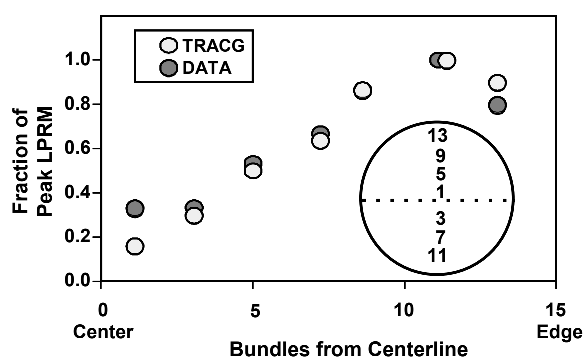

Figure B-8. European BWR Measured and Calculated Regional Oscillation Contour

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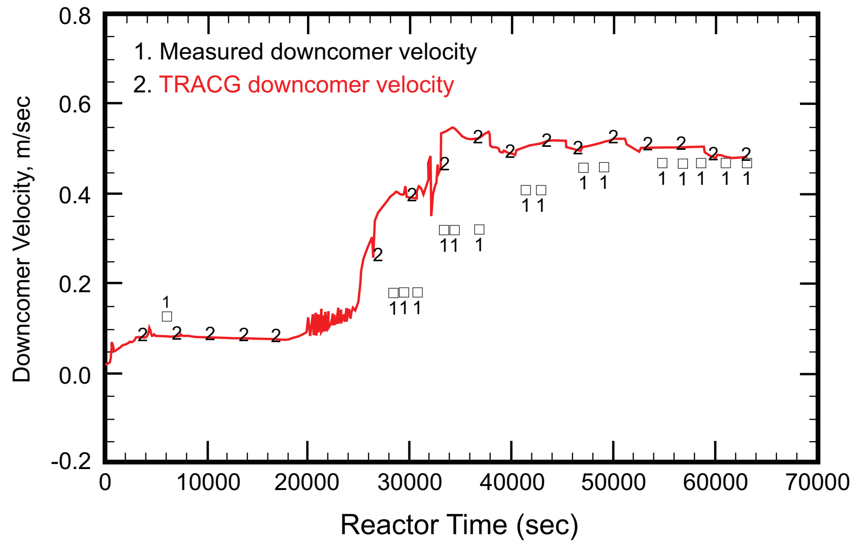

Figure B-9. Comparison of Measured and Calculated Downcomer Velocities for Dodewaard Startup

Conventional BWRs adjust reactor power using recirculation flow; how is reactor power adjusted in the ESBWR?

There are two ways to adjust reactor power for the ESBWR: (1) with control rods, and (2) with final feedwater temperature.

Just as control rods are used during startup and to compensate for burnup reactivity effects in conventional BWRs, they are used to adjust reactivity and reactor power in the ESBWR as well. However, two features improve maneuverability for the ESBWR when using control rods. The ESBWR uses Fine Motion Control Rod Drives (FMCRD) introduced in the GEH ABWR, which moves the control blades using an electric motor with a slower movement rate and finer positioning capability than the hydraulic movement and positioning capability of the conventional BWR Locking Piston Control Rod Drive. Furthermore, the ESBWR can move a single control rod or a ganged group of control rods. These features provide greater flexibility in the rate and magnitude that reactor power can be adjusted using control rod movement and positioning.

A second capability to adjust reactor power for the ESBWR is provided using feedwater temperature. By varying the final feedwater temperature, core inlet subcooling is varied, which in turn results in a change in the average void fraction in the reactor core. Increasing feedwater temperature results in a higher core average void fraction that feeds back to reduce reactor power through the negative void coefficient of reactivity. Decreasing feedwater temperature results in a lower core average void fraction, thereby producing higher reactor power.

Changing the core average void fraction produces a similar reactivity change as changing the void fraction with recirculation flow in a conventional BWR. Instead of a power-flow map to guide operation, however, a power-feedwater temperature map is constructed for the ESBWR. Figure B-10 illustrates the power-feedwater temperature map, conceptually, for the ESBWR. (The map is typical because the boundaries of the operating domain are adjusted on a cycle-by-cycle basis.)

The ESBWR rated operating condition on the power-feedwater temperature map is denoted as SP0. Six feedwater heaters receiving turbine extraction steam produce a nominal final feedwater temperature of 215.6ºC (420ºF) for this rated condition. In addition to these six feedwater heaters, a seventh feedwater heater receiving extraction steam from the main steam line is available to increase the final feedwater temperature for maneuverability purposes. (Note that since this feedwater heater receives steam from the main steam line before useful work is produced by the turbine, it does not cause an increase in the Rankine steam cycle efficiency and therefore would normally be in a standby condition during plant steady-state, baseload operation.) Heater bypass lines are provided to facilitate a reduction in feedwater temperature as well.

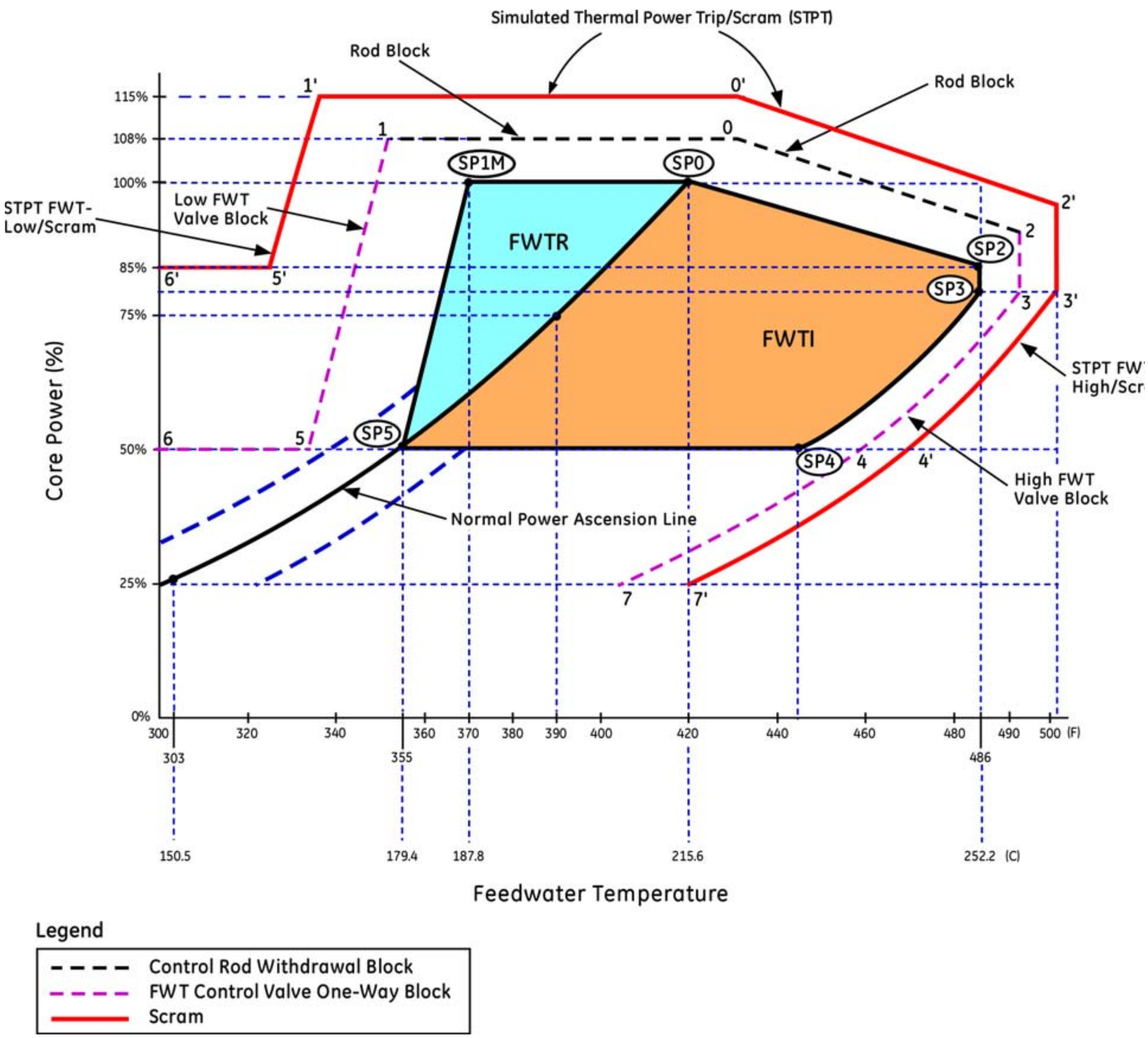

Figure B-10. Typical ESBWR Power-Feedwater Temperature Operating Map

Increasing feedwater temperature by 36.7ºC (66ºF) when starting at the nominal SP0 rated condition will result in a 15% reduction in reactor power (end state SP2 in Figure B-10). Various other trajectories are possible within the power-feedwater temperature domain, wither feedwater temperature increase (FWTI) or feedwater temperature reduction (FWTR).

Thus, in conjunction with initial power ascension using control rods to approximately 50% rated output (SP5), the power-feedwater temperature domain is useful for implementing startup, fuel preconditioning and control rod swap operating strategies for the ESBWR

What has ESBWR done to reduce worker radiation exposure?

Overview

Historically, operating BWRs have had higher workeHistorically, operating BWRs have had higher worker occupational radiation exposures than PWRs. Improvements in plant operations and lower forced outage rates have improved all LWRs and reduced the gap between PWRs and BWRs. In 2004, the average annual U.S. power reactor exposures were 156 person-Rem per reactor at BWRs and 71 person-Rem per reactor at PWRs (per NRC publication NUREG-0713). Figure B-11 shows the trend.

Starting with ABWR, a lot of design attention was paid to improving occupational exposure, based on operating experience. Since commencement of operations in 1996, the two ABWR units at Kashiwazaki-Kariwa in Japan have averaged about 36 person-Rem per reactor each year, for refueling and inspection work. The factors involved in this dramatic improvement are discussed below. ESBWR will benefit from ABWR technology. In addition, there are design improvements which should even further reduce occupational exposure.

Materials Considerations

With the increasing focus on managing radiation buildup, the ESBWR focuses on both alloy selection and water chemistry modification and control to minimize this effect, just as ABWR did. Cobalt, as activated cobalt-60, is well known as the major contributor to gamma radiation levels in nuclear plants during shutdown. This presence of Co-60 significantly affects personnel exposure to gamma radiation during maintenance activities. There are many potential sources of cobalt in a water-cooled reactor system. cobalt-based alloys are commonly used in valves as seats and other components where wear and galling resistance is vital. Stainless steel, commonly used for reactor internal structures and some piping systems, also contains some cobalt as a trace element. Also, carbon steels contribute dissolved iron to the reactor water, which acts as an accumulator of cobalt and promotes its concentration.

With respect to materials improvements, three broad areas are addressed: (1) elimination of cobalt-based alloys wherever possible; (2) reduction of cobalt in stainless steels of high surface area, high flow rate components; and (3) reduction of iron input to reactor water by use of more corrosion-resistant ferritic piping materials. Cobalt-based alloys are almost completely eliminated inside the reactor pressure vessel of the ESBWR. One reason is that, by eliminating jet pumps found in conventional BWRs, stellite hard-facing used in jet pump slip joints will not be present. Another significant improvement is the use of iron- and nickel-based alloys for control blade guide rollers and pins rather than cobalt-based alloys. Outside the pressure vessel, valve seats will be considered for cobalt alloy replacement. Stainless steel and nickel alloy components in which cobalt content will be controlled include feedwater heaters, control blades, fuel bundle hardware, steam separators and dryer, and some core support components. To reduce iron input, extensive use of low alloy steel and atmospheric corrosion-resistant steel in piping systems and other components will be implemented. In addition to other measures, the RWCU/SDC System includes filtration features to minimize the amount of cobalt-containing corrosion products retained in the system. The other approach to minimizing radiation buildup is through water chemistry changes. Small additions of depleted zinc oxide (DZO) to the coolant drastically change the oxide films, thereby minimizing cobalt content and reducing surface radiation buildup.

Minimization of Radiation Exposure

The ESBWR combines advanced facility design features and administrative procedures designed to keep the occupational radiation exposure to personnel as low as reasonably achievable (ALARA). During the design phase, layout, shielding, ventilation and monitoring instrument designs were integrated with traffic, security, and access control. Operating plant results were continuously integrated during the design phase. Clean and controlled access areas are separated.

Reduction in the plant personnel radiation exposure was achieved by: (1) minimizing the necessity for and amount of personnel time spent in radiation areas and (2) minimizing radiation levels in routinely occupied plant areas in the vicinity of plant equipment expected to require personnel attention.

Minimizing the time required by plant operators in high radiation areas was achieved through design features such as described in these examples:

- Controls and instrumentation are located in areas that are accessible during normal and abnormal operating conditions. Remote operation is incorporated where feasible (e.g., backwashing operations in the RWCU/SDC and FAPCS)
- Equipment is designed to facilitate maintenance. The FMCRDs require greatly reduced maintenance compared to the locking piston drives used in conventional BWRs today. Most of the maintenance is performed on the motors, which are magnetically-coupled to the drives through the pressure boundary, thus eliminating the need to expose the plant personnel to reactor water. In addition, the shoot-out steel, which complicates access to the drives in conventional BWRs, was eliminated by design
- Equipment such as the RWCU/SDC heat exchanger is designed with an excess of tubes to permit plugging of some tubes. Some valves have stem packing of the cartridge type that can be easily replaced. Refueling tools are designed for drainage and with smooth surfaces to reduce contamination. Vessel and piping insulation is easily removable

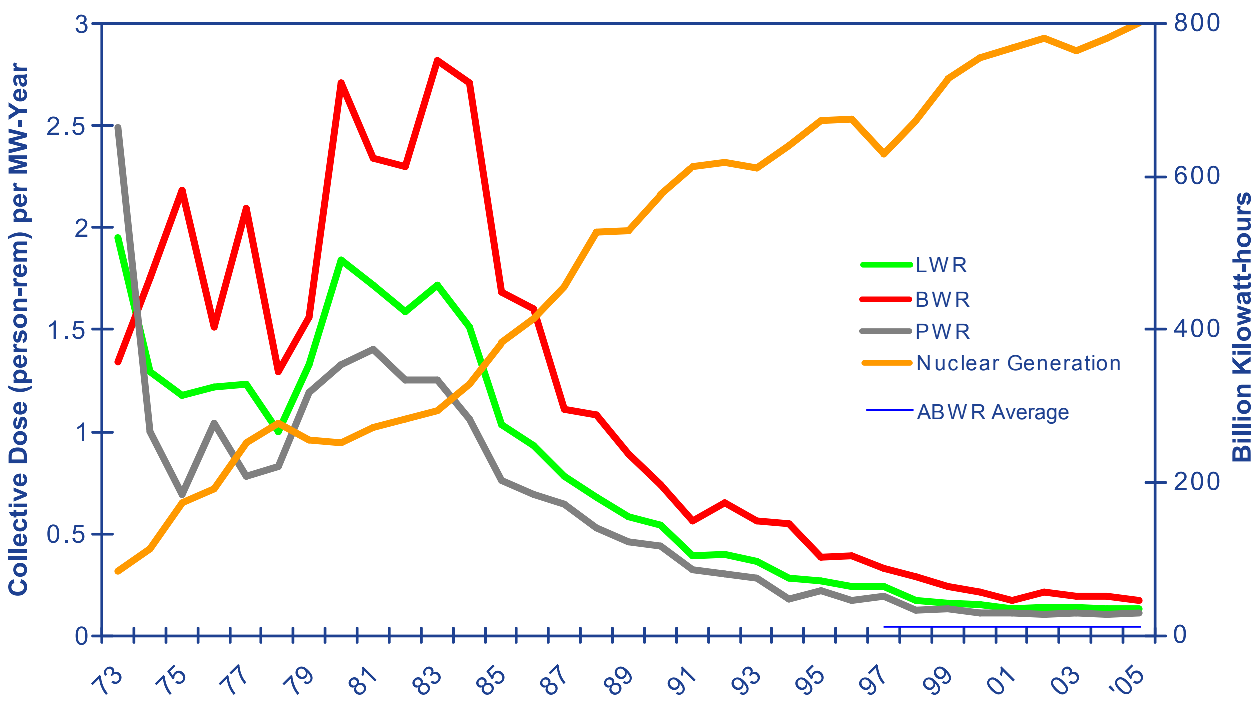

Figure B-11. US LWR Nuclear Plant Occupational Exposure

- Materials selection considers ALARA requirements
- Examples of design considerations to minimize equipment radiation levels include
- Equipment and piping are designed to reduce accumulation of radioactive materials. Piping is constructed wherever possible of seamless pipe, and filters and demineralizers are backwashed and flushed prior to maintenance
- Leakage from equipment is piped to sumps and floor drains
- The materials used in the primary coolant system consist mainly of austenitic stainless steel, carbon steel, and low alloy steel components. The use of cobalt is minimized to reduce the potential for gamma radiation
- Radioactive isotopes in the primary coolant are limited by the use of RWCU/SDC and condensate demineralizer on the reactor feedwater
- External recirculation pumps and recirculation piping have been replaced with internal natural circulation
- Clean purge water is continuously supplied to the FMCRDs to keep the equipment from accumulating radioactive contaminants
- The retractable SRM and IRM neutron detectors have been replaced with fixed in-core SRNMs
- The Traversing In-core Probe (TIP) system has been replaced by fixed in-core gamma thermometer calibration devices

In addition to minimizing radiation levels and time spent in radiation areas, other features were incorporated to further reduce personnel radiation exposure.

Radiation zones were established in all areas of the plant as a function of both the access requirements and radiation sources in that area. Operating activities, inspection requirements of equipment, maintenance activities, and abnormal operating conditions were considered in determining the appropriate zoning for a given area.

Extensive consideration was given to implementation of radiation shielding. The primary objective of radiation shielding is to protect operating personnel and the general public from radiation emanating from the reactor, power conversion systems, radwaste process systems and auxiliary systems. Radiation shielding also is designed to limit the radiation exposure of critical components within specified limits to assure that their performance and design life are not impaired.

For all areas potentially having airborne radioactive materials, the ventilation systems were designed such that during normal and maintenance operations, airflow is from an area of low potential contamination to an area of higher potential contamination. This was achieved by keeping specific zones at higher or lower pressure, as required, in respect to their adjacent compartments.

Overall, it is expected that the annual worker dose for the ESBWR will be significantly reduced relative to conventional light water reactors due to these materials technologies. Based on feedback from the operating ABWRs in Japan, on which the ESBWR is based, it is expected that the U.S. Utility Requirements Document goal of less than 100 person-Rem/yr (1 person-Sv/yr) will be met by a wide margin. Current Japanese experience for ABWRs is 36 person-Rem/yr (0.36 person-Sv/yr) for refueling and inspection work.

Turbine Radiation Exposure

Because the BWR uses a direct-power cycle, there is a concern about potential contamination of turbine equipment and subsequent radiation exposure to workers. Forty years of plant operations have shown fears of equipment contamination to be unfounded.

The important radiation source is N16, created in the reactor, and carried over with the steam to the turbine. It releases a high energy 6 MeV gamma ray upon decay, but has a very short 7-second half-life. The design implication of this requires significant turbine building shielding in areas where steam is present (e.g., main turbine and moisture separator-reheaters) to allow access to key areas during plant operation, but the rapid decay means no residual radiation after shutdown. Only about 20% of occupational exposure to workers comes from maintenance activities in the Turbine Building.

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Index

Italics represents page number of illustration, photo, or table.

A

B

C

D

O

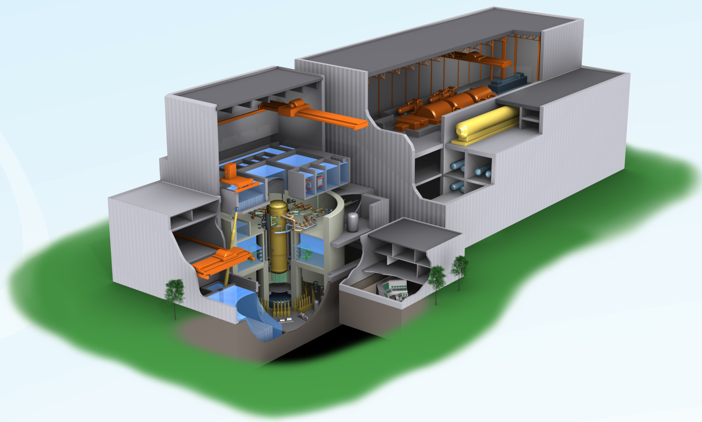
P

Q

R

S

ESBWR General Description



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