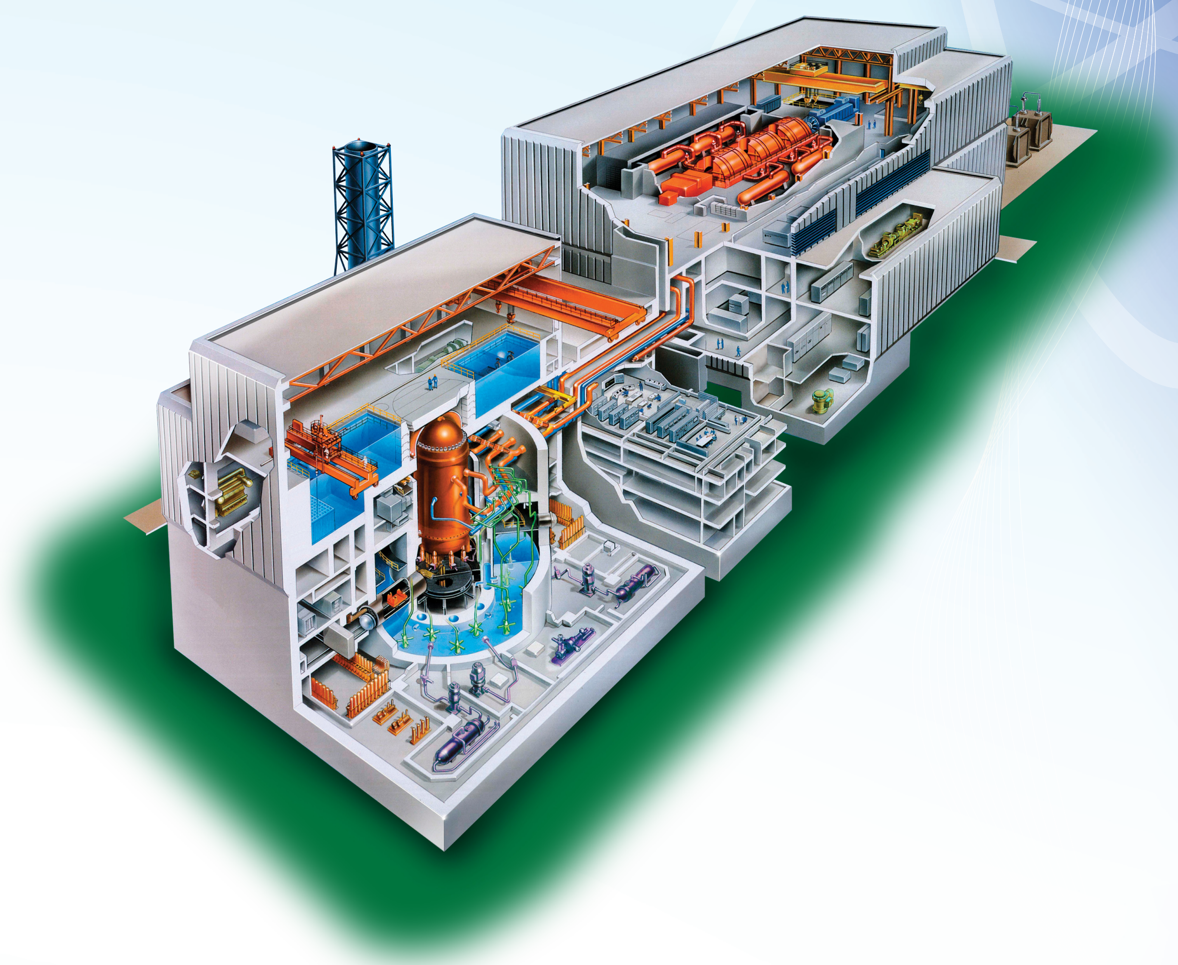
GE Hitachi Nuclear Energy

The ABWR Plant General Description



HITACHI

ABWR
Plant General Description

7.1.2007

GE HITACHI

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Contents

Acronyms

ABWR	Advanced Boiling Water Reactor
ACS	Atmospheric Control System
ADS	Automatic Depressurization System
AIWA	AC-Independent Water Addition System
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
APR	Automatic Power Regulator System
APRM	Average Power Range Monitor
ARD	Anti-Reverse Rotation Device
ARI	Alternate Rod Insertion
ARM	Area Radiation Monitoring
ASD	Adjustable Speed Drive
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 Transients Without Scram
B&V	Black and Veatch
BAF	Bottom of Active Fuel
BOP	Balance of Plant
BWR	Boiling Water Reactor
C&I	Control and Instrumentation
CAM	Containment Atmospheric Montioring System
CB	Control Building
CCC	Control Cell Core
CCFP	Contingent Containment Failure Probability
CDF	Core Damage Frequency
CFS	Condensate and Feedwater System
CO	Commercial Operation
COE	Coost of Electricity
CP	Construction Permit
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDHS	Control Rod Drive Hydraulic System
CRT	Cathode Ray Tube
CSP	Condensate Storage Pool
CST	Condensate Storage Tank
CTG	Combustion Turbine Generator
CWS	Circulating Water System
DAT	Design Acceptance
DBA	Design Basis Accident
DCPS	DC Power Supply
DCV	Drywell Connecting Vent
DG	Diesel Generator
DMC	Digital Measurement Controller
DPS	Diverse Protection System
DRM	Dry Radwaste Management System
DW	Drywell
DWC	Drywell Cooling
ECCS	Emergency Core Cooling System
ECP	Electrochemical Potential
ECW	Emergency Chilled Water
EDG	Emergency Diesel Generator
EHC	Electro-hydraulic Control (Turbine Control System)
EMI	Electro-Magnetic Interference
EMS	Essential Multiplexing System
EPD	Electrical Power Distribution
EPRI	Electric Power Research Institute
ESF	Essential Safeguards Feature
EPRI	Electric Power Research Institute
FCS	Flammability Control System
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
FP	Fire Protection
FPCU	Fuel Pool Cooling and Cleanup
FSAR	Final Safety Analysis Report
FSC	First Structural Concrete
FTDC	Fault Tolerant Digital Controller
FW	Feedwater
FWP	Feedwater Pump
FWC	Feedwater Control System
GE	General Electric Company

GETAB	General Electric Thermal Analysis Basis
GPM	Gallons per minute
HCU	Hydraulic Control Unit
HCW	High-Conductivity Waste
HEPA	High Efficiency Particulate Air
HFF	Hollow Fiber FIlter
HIC	High Integrity Container
HPCF	High Pressure Core Flooder
HPCP	High Pressure Condensate Pump
HPIN	High Pressure Nitrogen Gas Supply
HVAC	Heating, Ventilation and Air-Conditioning
HWC	Hydrogen Water Chemistry
I&C	Instrumentation and Control
IASCC	Irradiation-Assisted Stress Corrosion Cracking
IGSCC	Intergranular Stress Corrosion Cracking
ILRT	Integrated Leak Rate Test
IMS	Information Management System
IRM	Intermediate Range Monitor
ISI	In-Service Inspection
ITAAC	Inspection, Test, Analysis and Acceptance Criteria
LCW	Low Conductivity Waste
LD	Lower Drywell
LDI	Leak Detection and Isolation System
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
LPFL	Low Pressure Flooder
LPRM	Local Power Range Monitor
LRM	Liquid Radwaste Management System
LTP	Lower Tie Plate
MCC	Main Control Console/ Motor Control Center
MCES	Main Condenser Evacuation 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
MOV	Motor-Operated Valve
MRBM	Multi-Channel Rod Block Monitoring System
MS	Main Steam System
MSIV	Main Steam Isolation Valve
MSR	Moisture Separator Reheater
MUW	Makeup Water System
MUX	Multiplexer
MWB	Makeup Water Building
NBS	Nuclear Boiler System
NCW	Normal Chilled Water
NDT	Nil Ductility Temperature
NEMS	Non-Essential Multiplexing System
NMS	Neutron Monitoring System
NRC	Nuclear Regulatory Commission
NRHX	Non-Regenerative Heat Exchanger
NSS	Nuclear Steam Supply
O&M	Operation and Maintenance
OG	Offgas System
OPRM	Oscillation Power Range Monitor
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
PRM	Process Radiation Monitoring System
PRNM	Power Range Neutron Monitor System
PWR	Pressurized Water Reactor
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RBC	Rod Brake Controller
RCCV	Reinforced Concrete Containment Vessel
RCW	Reactor Building Cooling Water System
RCIS	Rod Control and Information System
RCIC	Reactor Core Isolation System
RCPB	Reactor Coolant Pressure Boundary
RFC	Recirculation Flow Control System
RHR	Residual Heat Removal
RHX	Regenerative Heat Exchanger

RIP	Reactor Internal Pump
RMC	Recirculation Motor Cooling
RMISS	Recirculation Motor Inflatable Shaft Seal
RMP	Recirculation Motor Purge
RMU	Remote Multiplexer Unit
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRCS	Redundant Reactivity Control System
RSS	Remote Shutdown System
RSW	Reactor Building Service Water System
RTNDT	Reference Nil Ductility Temperature
RWCU	Reactor Water Cleanup System
RWM	Rod Worth Minimizer
S&PC	Steam and Power Conversion System
SA	Severe Accident
SAR	Safety Analysis Report
SBO	Station Blackout
SBPC	Steam Bypass and Pressure Control System
SCC	Stress Corrosion Cracking
SCRRI	Select Control Rod Run-in
SDV	Scram Discharge Volume
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Ejector
SHE	Standard Hydrogen Electrode
SLCS	Standby Liquid Control System
SOE	Sequence of Events
SP	Suppression Pool
SPCU	Suppression Pool Cleanup System
SPDS	Safety Parameter Display System
SRM	Source Range Monitor
SRNM	Startup Range Neutron Monitor
SRV	Safety/Relief Valve
SSAR	Standard Safety Analysis Report
SSE	Safe Shutdown Earthquake
SSLC	Safety System Logic and Control
SW	Service Water
TAF	Top of Active Fuel
TCW	Turbine Building Cooling Water System
TBS	Turbine Bypass System
TBV	Turbine Bypass Valve
TCCWS	Turbine Component Cooling Water System
TCS	Turbine Control System
TCV	Turbine Control Valve
TCS	Turbine Control System
TEPCO	Tokyo Electric Power Company
TGSS	Turbine Gland Steam System
TIP	Traversing In-Core Probe
TIU	Technician Interface Unit
TMSS	Turbine Main Steam System
TPC	Taiwan Power Company
TRA	Transient Recording and Analysis
TSC	Technical Support Center
TSW	Turbine Building Service Water System
UAT	Unit Auxilliary Transformer
UD	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
VDC	Volts-Direct Current
WDP	Wide Display Panel
WRNM	Wide Range Neutron Monitoring Syste
WW	Wetwell

Chapter 1 Introduction

Nuclear Energy for the New Millennium

Nuclear energy plays a major role in meeting the world's energy needs. At the end of 2005, there were 443 nuclear power plants operating in 32 countries. 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 1600 million tons per year at a time when the world is trying to reduce emissions by 4200 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 Advanced Boiling Water Reactor (ABWR) 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. Four ABWRs have been constructed in Japan and are reliably generating large amounts of low cost electricity. Taiwan is constructing two more ABWRs which will enter commercial operation in 2009 and 2010. 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 ABWR represents an entirely new approach to the way nuclear plant projects are undertaken. The ABWR was licensed and designed in detail before construction ever began. Once construction did begin, it proceeded smoothly from start to finish in just four years. Capital costs amounted to $1600/kW at which level nuclear is very competitive with other forms of power generation.

The successful design, licensing, construction and operation of the ABWR nuclear power plant ushers in a new era of safe, economic and environmentally friendly nuclear electricity. The ABWR 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.

Forty 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 1950's 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, interestingly enough, not a true BWR. The design was based upon dual steam cycle, not the direct steam cycle that 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 ABWR 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 (Oyster Creek) appeared in the mid-1960's 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

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	Cofrentes 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 solid-state microprocessors Fiber optic data transmission / multiplexing Increased number of fuel bundles Titanium condenser Improved ECCS: high/low pressure flooders

Table 1-1. Evolution of the GE BWR

recirculation loops were needed. This change first appeared in the Dresden-2 BWR/3 plant.

The use of reactor internal pumps in the ABWR design has taken this process of simplification to its logical conclusion. 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. This design feature is the source of many of the ABWR's safety and operational advantages. Figure 1-1 illustrates the evolution of the reactor system design.

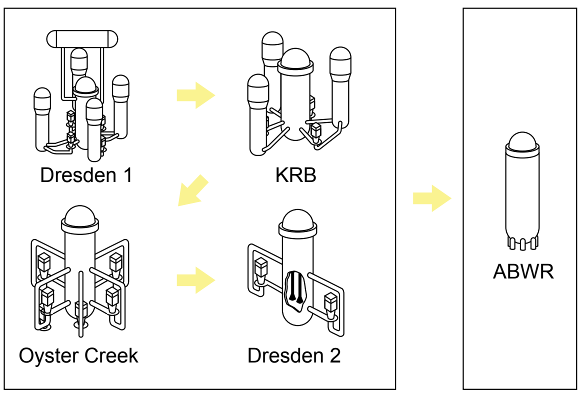


Figure 1-1 Evolution of the Reactor System Design

The first BWR containments were spherical "dry" structures, similar to those still used today in PWR designs. The BWR, however, quickly moved to the "pressure suppression" containment design for its many advantages. Among these are:

- 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 torus used to house a large water inventory in the Mark I gives this design its characteristic light bulb configuration. The conical Mark II design has a less-complicated arrangement, based on steel-lined reinforced concrete. 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 steel 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 free-standing 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. Figure 1-2 illustrates the evolution of the BWR containment from the earliest versions to today's ABWR RCCV design.Where the reactor building is also shown, the containment is outlined

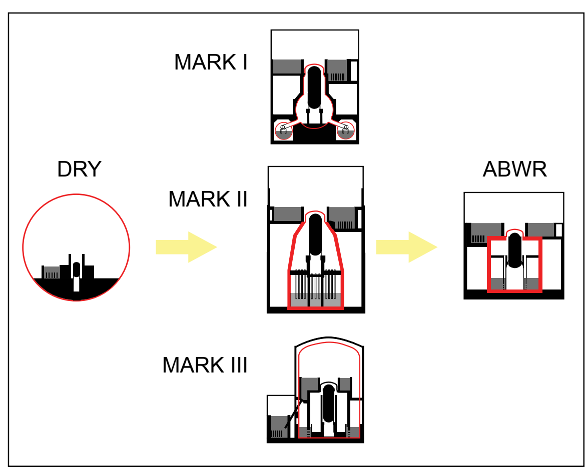


Figure 1-2. Evolution of BWR Containment

in red.

There are 93 BWRs, including four ABWRs, currently operating 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. Japan, for example, has 32 BWR plants, representing nearly two-thirds of its installed nuclear capacity. The Tokyo Electric Power Company (TEPCO) owns 17 nuclear plants, all of which are BWRs. TEPCO’s Kashiwazaki-Kariwa nuclear station, which consists of seven (7) large BWRs, is the largest power generation facility in the world, licensed for 8,200 MWe. Similarly, BWR plants predominate in Taiwan and several European countries. In the United States, there are 37 operating BWRs.

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

ABWR Development and Design Objectives

Development of the ABWR took place during the 1980’s under the sponsorship of the Tokyo Electric Power Company (TEPCO). The stated purpose of the development effort was to design a BWR plant that included a careful blend of (1) the best features of worldwide operating BWRs, (2) available new technologies, and (3) new modular construction techniques. Safety improvements were, as always, the top priority. Anticipating the economic challenges that lay ahead, special attention was paid to systematically reducing the capital cost and incorporating features into the plant design that would make maintenance significantly easier and more efficient.

Development of the ABWR occurred in a series of steps. Phase 1 was a conceptual design study that determined the feasibility of the ABWR concept. Phase 2, in which most of the development work took place, included more detailed engineering and the testing of new technologies and design features. The purpose of Phase 3 was to put the finishing touches on the design and systematically reduce capital costs, which proved to be a highly successful and, in hindsight, fortuitous endeavor. The development phases came to an end in 1988 when TEPCO announced that the next Kashiwazaki-Kariwa units to be constructed would be ABWRs.

With the selection of the ABWR for the K-6&7 project, the detailed, or project, engineering began. Licensing activities with the Japanese regulatory agency, MITI (Ministry of International Trade and Industry), also started at this time and, interestingly, were conducted in parallel for some time with the review of the ABWR in the U.S. by the Nuclear Regulatory Commission (NRC). MITI and the NRC, in fact, held several meetings to discuss their respective reviews.

By 1991, the detailed design was essentially complete and MITI concluded its licensing review. An Establishment Permit, or license, was issued in May 1991. Excavation began later that year on September 17, bringing a decade of development work to a successful conclusion.

Development of an advanced nuclear plant is a major endeavor. The development of the ABWR spanned a decade and cost an estimated $500M. Such an enterprise can only be undertaken in cooperation with many other organizations. The ABWR was developed by GE in cooperation with its technical associates Hitachi Ltd. and Toshiba Corp. The sponsorship and guidance of TEPCO was instrumental. The ABWR development also received financial support from the other Japanese utilities that operate BWRs, as well as from sixteen U.S. utilities.

ABWR 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 on November 7, 1996. Unit 7, the second ABWR, followed shortly thereafter with commercial operation commencing on July 2, 1997.

Both 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 on budget, which is an impressive record of performance, since these were first-of-a-kind units.

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, both plant 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.

Figure 1-3. Kashiwazaki Units 6 & 7

The ABWR in the United States

The licensing of the ABWR has been described as the most exhaustive, and perhaps exhausting, review ever undertaken by the U.S. Nuclear Regulatory Commission. The efforts of the NRC and GE came to fruition on May 2, 1997 when then Chair of the NRC, Ms. Shirley Jackson, approved and signed the ABWR Design Certification 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. See Figure 1-4 for a reproduction of the ABWR Design Certification.

The United States
Nuclear Regulatory Commission
hereby certifies the
Advanced Boiling Water Reactor
standard design as set forth in
Appendix A of 10 CFR Part 52

Dated this 2nd day of May, 1997

Figure 1-4. ABWR Design Certification

The successes continued when the ABWR First-of-a-Kind Engineering (FOAKE) program was completed in September 1996 to the praise and satisfaction of the utility sponsors. FOAKE is an equally significant accomplishment because it represents a major step toward the U.S. industry's other goal—to have a (pre-licensed) design that is 90% engineered prior to the start of construction. At the conclusion of the FOAKE program, approximately 65% of the engineering of the U.S. version of the ABWR was complete. The remaining engineering is being done as part of the Lungmen project, described below.

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 July 2009. The schedule for Unit 2, including the start of commercial operation, is one year later.

Nuclear Plant Projects in the New Millennium

The way in which ABWR nuclear plants are designed, licensed and constructed is vastly different than was the case 10 or 20 years ago.

Design and Licensing

The ABWR nuclear plant is licensed and designed in its entirety prior to the start of construction. Today, long before first concrete is poured, all safety and engineering issues are identified and resolved. This precludes construction delays due to re-engineering, a problem which plagued so many projects in the past and contributed significantly to the high (and in some cases mind-numbing) capital costs.

The ABWR has been designed to higher levels of safety, including being designed to prevent and mitigate the consequences of a Severe Accident. Licensing documents approved by the USNRC indicate that even in the event of a severe accident, there would be no release of radioactive material to the public.

Today's nuclear plants are extensively and exhaustively reviewed by multiple regulatory bodies. In fact, the ABWR has been reviewed and approved in three countries (Japan, U.S. and Taiwan). This ensures that the licensing of the ABWR will proceed on a smooth and timely basis in other countries that choose to deploy an ABWR.

The ABWR design has been captured electronically using the latest state-of-the-art information management technology called POWRTRAK. 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. POWRTRAK is both a 3D model design tool and an extensive database for plant equipment and materials.

The approach described above is being fully utilized for the Lungmen project. The design and licensing of these ABWRs are proceeding smoothly as expected.

Construction of Nuclear Plants in the 2000's

Nuclear plants today are constructed much differently than in the past. The most notable difference is the schedule. The ABWR can be built in only four years, from first concrete to the start of commercial operation. Design simplifications and the use of new construction technologies and techniques make this possible.

Today, the plant owner is spared the concern for schedule delays and cost overruns. Suppliers commit to a fixed schedule and price, largely because the design has been pre-licensed and pre-engineered.

Of course, there is no substitute for experience. The Lungmen ABWRs are being supplied by a team of U.S. and Japanese suppliers, led by GE, that were also involved in the supply of the Japanese ABWRs. This team and the supporting network of equipment sub-suppliers is accustomed to working on an international stage and can readily transplant its experience and know-how to a new host country. This is the basis for the "learning curve" effect, which reduces capital costs with each new unit.

ABWR is Accumulating Operating Experience

The ABWRs in Japan have now accumulated many years of operating experience after having completed a highly successful construction effort. This represents a wealth of information and know-how that is of benefit to subsequent ABWR projects.

Business Risks of the ABWR

It is important to scrutinize a proposed new nuclear project in terms of the business risks it carries. A plant design and its supplier contribute significantly to the owner's ability to manage those risks. The risks that need to be considered are those associated with: the market for electricity, the licensing and construction of the plant itself, the operation of the new facility, the technical aspects of the plant design and financing the project.

In light of these many risks, what characteristics should a potential owner look for in a design and in a supplier?

- Licensing risk-can the plant be licensed on a reasonable and predictable schedule or will this become a lengthy effort that seriously effects the date of commercial operation?

- Engineering risk-is the plant fully designed before construction starts or will there be new surprises that result in costly design changes and construction slowdowns when it is designed during the course of the project?
- Technology risk-will the plant perform as expected or will some unknown technical problem keep the plant shutdown and thus unable to meet its revenue projections?
- Cost risk-will the plant cost more than budgeted, threatening its ability to compete in the open market?
- Schedule risk-will the plant start generating electricity-and revenue-as planned or will the schedule become protracted leading to cost increases?
- Financing risk-will the new plant have predictable revenues and costs or is there uncertainty and lack of confidence by lenders and investors for the new project?

So, how does GE's ABWR measure up?

Licensing Risk

The ABWR has been designed to the highest standards of safety and has already received a Design Certification. Moreover, the ABWR has been licensed in Japan, where two ABWRs have operated safely and successfully for nearly ten years, and in Taiwan. This suggests to us that the process for reviewing a COL based upon an ABWR project should reasonably take one year only.

Engineering Risk

The ABWR is a fully designed plant complete with equipment and manufacturing drawings. Materials, quantities and costs are precisely known. This means there will be no major surprises during construction that would create costly delays and re-designs. This is also the basis upon which GE is able to offer a firm price for its scope of supply, eliminating that element of risk.

Technology Risk

The ABWR is the only advanced nuclear plant being offered that has actually been constructed. In fact, two ABWR units in Japan have a combined ten years of operational experience. Its owner, the Tokyo Electric Power Company, has published information that shows these plants have met or exceeded all of its design and performance goals and no technical problems or design flaws have surfaced. This is directly attributable to the $500M, five year test and development program jointly undertaken by TEPCO, the other Japanese utilities that own BWRs, GE, Hitachi and Toshiba. The TEPCO units are enjoying high availability and capacity factors.

Cost and Schedule Risk

GE has eliminated this risk for new owners by offering the ABWR with both a fixed price and construction schedule, the same basis for the Japanese and Taiwan ABWR projects. This reflects the confidence that comes with the experience of having delivered four ABWR units on a strict budget and schedule.

Financing Risk

Obtaining financing for a new nuclear project is sometimes perceived as a significant obstacle to new construction.

This perception is based upon bad experiences in the past with the U.S. being a case in point. The experience of Wall Street during the late 1970s and early 1980s was that it took 15 years and several billion dollars to complete a nuclear plant. Such perceptions tend to linger even though today nuclear plants are routinely built on budget and schedule.

Lenders and investors, however, are practical people. They will finance a project, including a nuclear project, if they can be assured of a return on their investment. Investors look for a solid project *pro forma* sheet and this means that the potential owner must be able to demonstrate persuasively that revenues and project costs are predictable. A well-managed utility that has over time successfully operated nuclear plants (avoiding safety lapses, achieving high capacity factors, controlling costs) assures the financial community that targets for revenues and on-going operational costs will be achieved.

Likewise, a well-managed engineering and construction team that has experience building advanced nuclear plants assures investors that capital costs projections will not be exceeded and that the plant will be free of technical problems. The GE led team brings this kind of experience and management to a project. This actually has the effect of further reducing the owner's capital costs for the very good reason that strong project fundamentals lead to more favorable financing. A reduction of even 0.25% in the interest rate or a favorable change in debt-to-equity requirements can save tens of millions of dollars.

Chapter 2
Plant Overview

The key design objectives for the ABWR were established during the 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 <100 m^3/yr.
- 48-month construction schedule.
- 20% reduction in capital cost ($/kWh) vs. previous 1100 MWe class BWRs.

At the time, 87% availability was a large improvement over the existing BWR fleet average. Based on more recent experience, however, it is expected that ABWR will achieve availability factors in the mid 90% range.

Summary of the ABWR Key Features

A comparison of key features of the ABWR to the previous model, known as BWR/6, is shown in Table 2-1. The cutaway rendering of the ABWR plant (Figure 2-1) illustrates the general configuration of the plant for a single unit site in the U.S. Shown in the foreground is the Reactor Building, and in the background is the Turbine Building. Between them is located the Control Building.

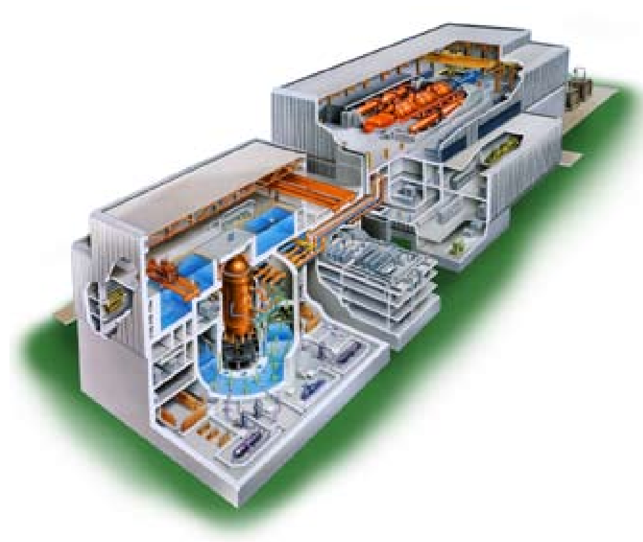


Figure 2-1. Cutaway Rendering of the ABWR

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

Safety Enhancement

Recognizing the desire for the continuous enhancement of safety, one of GE's goals for the ABWR was to reduce calculated core damage frequency by an order of magnitude relative to currently operating plants. The most important design feature contributing to this goal is the adoption of reactor internal pumps (RIPs), which are shown in Figure 2-3. These vessel-mounted pumps eliminate large, recirculation piping on the vessel, particularly involving

Feature	ABWR	BWR/6
Recirculation	Vessel-mounted reactor internal pumps	Two external loop Recirc system with jet pumps inside RPV
Control Rod Drives	Fine-motion CRDs	Locking piston CRDs
ECCS	3-division ECCS	2-division ECCS plus HPCS
Reactor Vessel	Extensive use of forged rings	Welded plate
Primary Containment	Advanced - compact, inerted	Mark III - large, low pressure, not inerted
Secondary Containment	Reactor Building	Shield, fuel, auxiliary & DG buildings
Control & Instrumentation	Digital, multiplexed, fiber optics, multiple channel	Analog, hardwired, single channel
Control Room	Operator task-based	System-based
Severe Accident Mitigation	Inerting, drywell flooding, containment venting	Not specifically addressed
Reactor Water Cleanup	2%, sealless pumps in cold leg	1%, pumps in hot leg
Offgas	Passive offgas with room-temperature charcoal	Active offgas with chilled charcoal filters

Table 2-1. Comparison of Key ABWR Features to a BWR/6

penetrations below the top of the core elevation, and make possible a smaller Emergency Core Cooling System (ECCS) network to maintain core coverage during postulated loss-of-coolant events.

The ABWR ECCS network was designed as a full three-division* system, with both a high and low pressure injection pump and heat removal capability in each division. For diversity, one of the systems, the Reactor Core Isolation Cooling (RCIC) System, includes a steam-driven, high pressure pump. Transient response was improved by designing three available high-pressure injection systems in addition to feedwater. The adoption of three on-site emergency diesel-generators to support core cooling and heat removal, as well as the addition of an on-site gas turbine-generator, reduces the potential for Station Blackout (SBO). The balanced ECCS system has less reliance on the Automatic Depressurization System (ADS) function, since a single, motor-driven high pressure core flooder (HPCF) can maintain core safety for any postulated pipe break.

Response to Anticipated Transients 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 feed-

*The term *division* means that all systems and support systems necessary to complete the safety function are contained within the division, and that division is physically separated from other divisions to avoid any propagating failures, such as threats due to fire or flood.

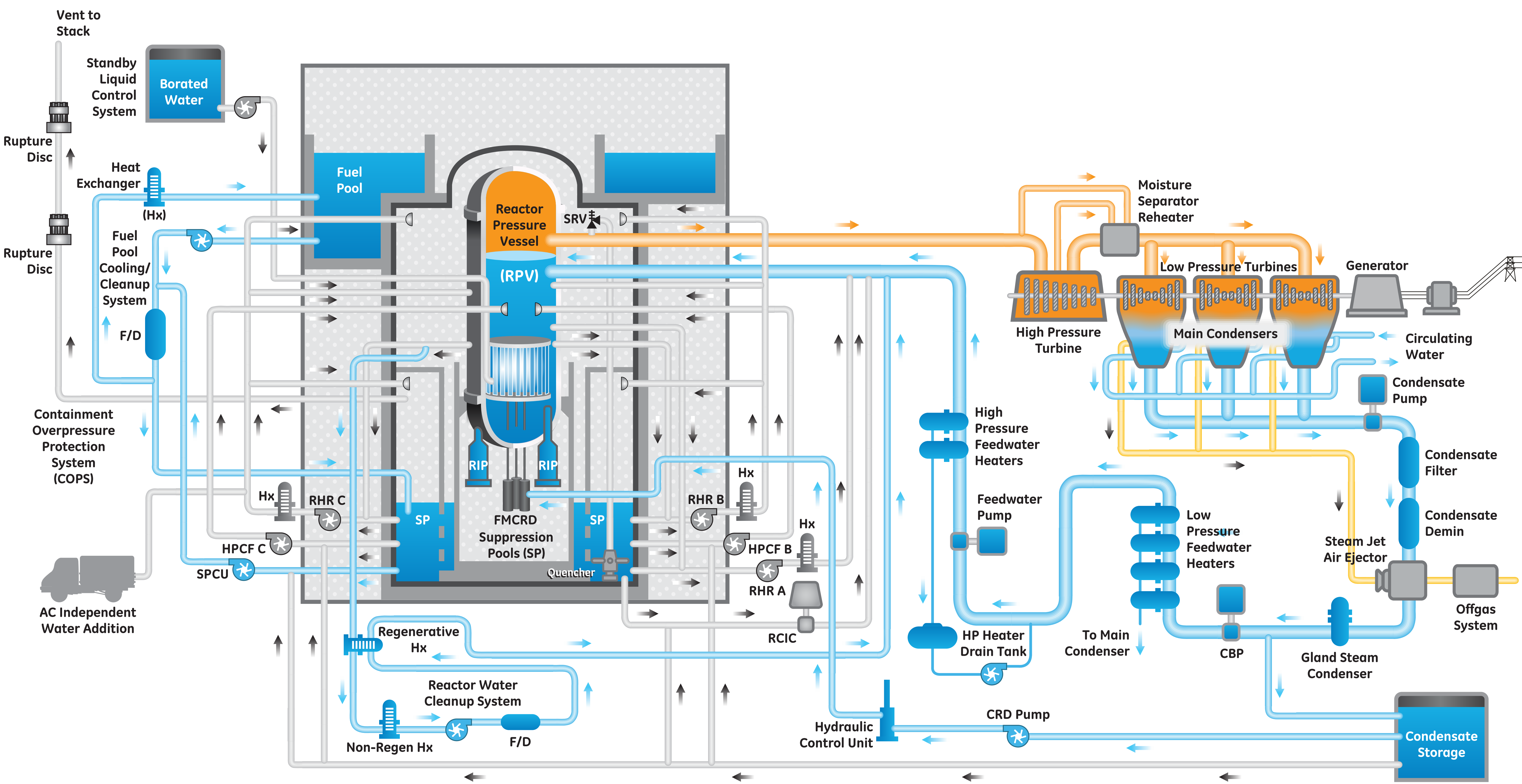


Figure 2-2 ABWR Major Systems

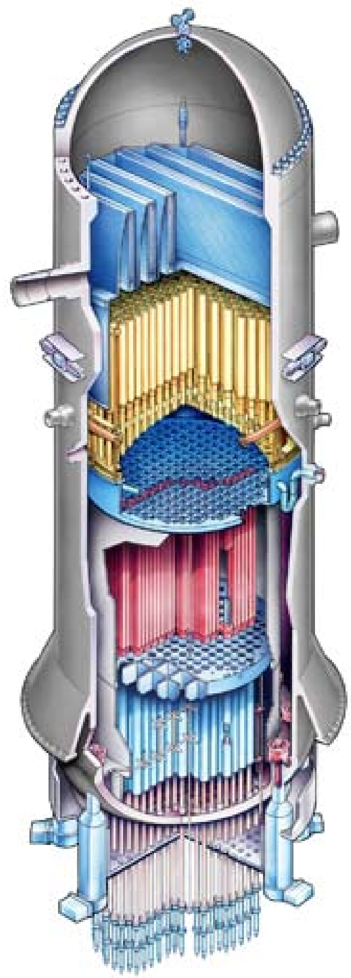


Figure 2-3. ABWR Reactor Pressure Vessel and Internals

water runback and Standby Liquid Control System (SLCS) injection.

Calculated core damage frequency is reduced by more than a factor of ten relative to the BWR/6 design. Furthermore, the ABWR also improved the capability to mitigate severe accidents, even though such events are extremely unlikely. Through nitrogen inerting, containment integrity threats from hydrogen generation were eliminated. Sufficient spreading area in the lower drywell, together with a drywell flooding system, assures coolability of postulated core debris. Manual connections make it possible to use onsite or offsite water systems to maintain core cooling. Finally, to reduce potential offsite consequences, a passive, hard-piped wetwell vent, controlled by rupture disks, is designed to prevent catastrophic containment failure and provide maximum fission product scrubbing. 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 less than 25 Rem. The probability of such an occurrence is calculated at the very low level of 10^{-9}/year. More information on this subject can be found in Chapter 10.

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 ABWR electrical and mechanical system, as well as the layout of equipment in the plant, is focused on improved O&M.

The reactor vessel is 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. Since there are ten RIPs on four power buses, the ABWR's recirculation system is quite robust. Pump speed is controlled by solid-state adjustable speed drives, eliminating the requirement for flow control valves and low-speed motor-generator sets. The wet motor design also eliminates rotating seals.

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 eliminated. The number of hydraulic control units (HCUs) was reduced by connecting two drives to each HCU. 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.

It was possible to significantly downsize ECCS equipment as a result of eliminating large vessel nozzles below the top of the core. Capacity requirements are sized based on operating requirements—transient response and shutdown cooling—rather than on the need for large reflood capability. Inside the reactor vessel, core spray spargers were eliminated, since no postulated LOCA would lead to core uncovery. For transient response, the initiation water levels for RCIC and HPCF were separated so that there is reduced duty on the equipment relative to earlier BWRs. There are three complete shutdown cooling loops, including dedicated vessel nozzles. Complex operating modes of the Residual Heat Removal (RHR) Systems, such as steam condensing, were eliminated. Finally, heat removal, in addition to core injection, was automated so that the operator no longer needs to choose which mode to perform during transients and accidents.

Lessons learned from operating experience were applied to the selection of ABWR materials. Stainless steel materials which 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) is recommended for normal operation to further mitigate any potential for stress corrosion cracking.

The use of material producing radioactive cobalt was minimized. The condenser uses titanium tubing at sea water sites and stainless steel tubing for cooling tower sites. The use of stainless steel in applications that currently use carbon steel was expanded. Depleted Zinc Oxide 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 System (RWCU) capacity to two percent. A more complete summary of materials and water chemistry considerations is given in Appendix B.

The Offgas System was simplified, reflecting lessons learned from operating experience. The charcoal beds are maintained at ambient temperature rather than refrigerated. The desiccant drier was eliminated.

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

Within the containment itself, no equipment

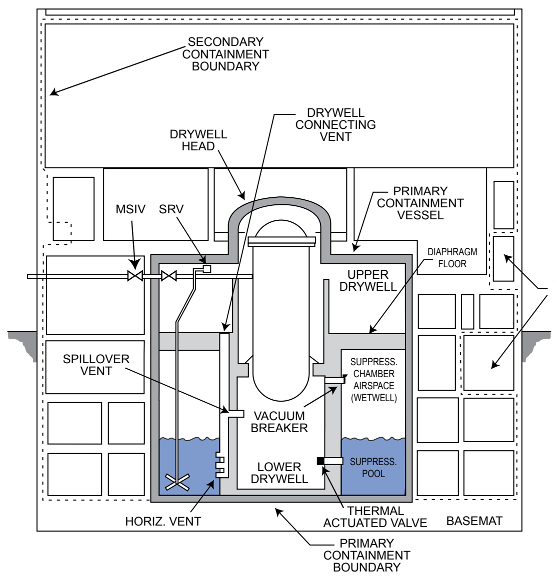


Figure 2-4. ABWR Reactor Building and Containment

requires servicing during plant operation. The containment is significantly smaller than that of the preceding BWR/6. However, primarily due to the elimination of the external 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 RIP and 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 containment, thus minimizing the chance for suppression pool contamination with foreign material.

A new Reactor Building design surrounds the containment and incorporates the same functions as the BWR/6 auxiliary, fuel and diesel-generator buildings. 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. Therefore, the BWR/6-type fuel transfer system has been eliminated.

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.

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 man-machine interface was significantly improved and simplified for the ABWR 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 the main 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. Studies show that less than one unplanned scram per year will be experienced with the ABWR. Increased system redundancies will also permit on-line maintenance. Thus, both forced outages and planned maintenance outages will be significantly reduced.

Minimization of Radiation Exposure and Radwaste

The ABWR 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 have a significant effect on 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 reduce liquid and solid radwaste input. Radwaste reduction in the ABWR can also be facilitated through the use of advanced incineration and super-compaction technologies.

Reduced Capital Cost

Design simplifications and quantities reductions as discussed above, together with an increase in plant electrical output, combine to make a significant improvement in plant capital cost.

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

Chapter 3 Nuclear Boiler Systems

Overview

The Nuclear Boiler Systems (NBS) produce steam from the nuclear fission process, and direct this steam to the main turbine. The NBS is comprised of the reactor vessel, which serves as a housing for the nuclear fuel and associated component, the recirculation system, the control rod drive system, the main steam system and the reactor building portion of the feedwater system. Other supporting systems are described in Chapter 5, Auxiliary Systems.

Reactor Vessel and Internals

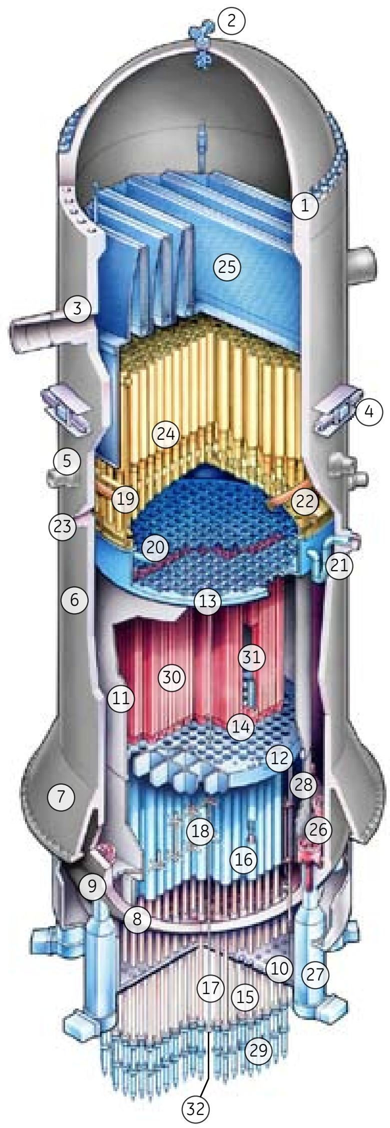
The reactor vessel houses the reactor core that is the heat source for steam generation. The vessel contains this heat, produces the steam within its boundaries, and serves as one of the fission product barriers during normal operation. The ABWR reactor assembly is shown in Figure 3-1. For this size reactor, the diameter of the ABWR RPV is increased but the height is decreased compared to earlier product lines. The increased diameter has resulted in increased wall thickness. The RPV is approximately 21 m in height and 7.4 m in diameter.

The most significant differences between the ABWR RPV and earlier product lines are as follows:

- Inward vessel flange design
- Steam nozzle with flow restrictor
- Double feedwater nozzle thermal sleeve
- Conical vessel support skirt
- Relatively flat bottom head
- Elimination of nozzles below the core
- Reactor internal pump penetrations
- Use of forged shell rings at and below core elevation.

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 ABWR safety systems to keep the core completely and continuously flooded for the entire spectrum of design basis loss-of-coolant accidents (LOCAs).

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. 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. In addition, the larger RPV volume leads to a reduction in the ABWR pressurization rate that would occur after a rapid isolation of the reactor from the normal heat sink.



1 - Vessel flange and closure head
2 - Vent and head spray assembly
3 - Steam outlet flow restrictor
4 - RPV stabilizer
5 - Feedwater nozzle
6 - Forged shell rings
7 - Vessel support skirt
8 - Vessel bottom head
9 - RIP penetrations
10 - Thermal insulation
11 - Core shroud
12 - Core plate
13 - Top guide
14 - Fuel supports
15 - Control rod drive housings
16 - Control rod guide tubes
17 - In-core housing
18 - In-core guide tubes and stabilizers
19 - Feedwater sparger
20 - High pressure core flooder (HPCF) sparger
21 - HPCF coupling
22 - Low pressure flooder (LPFL)
23 - Shutdown cooling outlet
24 - Shroud head and steam separator assembly
25 - Steam dryer assembly
26 - Reactor internal pumps (RIP)
27 - RIP motor casing
28 - Core and RIP differential pressure line
29 - Fine motion control rod drives
30 - Fuel assemblies
31 - Control rods
32 - Local power range monitor

Figure 3-1. ABWR Reactor Assembly

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

Vessel Flange and Closure Head (1) [1]

To minimize the number of main closure bolts, the ABWR RPV has an inside type flange. This is different from the earlier product lines, which had outside type vessel flanges. The inside type vessel flange allows a hemispherical main closure with radius less than the vessel radius. Also, this contributes to minimize the weight of the main closure. The vessel closure seal consists of two concentric O-rings which perform without detectable leakage at all operating conditions, including hydrostatic testing.

1. Numbers refer to Figure 3-1

Vent and Head Spray Assembly (2)

The reactor water cleanup return flow to the reactor vessel, via feedwater lines, can be diverted partly to a spray nozzle in the reactor head in preparation for refueling cooldown. The spray maintains saturated conditions in the reactor vessel head volume by condensing steam being generated by the hot reactor vessel walls and internals. The head spray subsystem is designed to rapidly cool down the reactor vessel head flange region for refueling and to allow installation of steamline plugs before vessel floodup for refueling.

The head vent side of the assembly passes steam and noncondensable gases from the reactor head to the steamlines during startup and operation. During shutdown and filling for hydrotesting, steam and noncondensable gases may be vented to the drywell equipment sump while the connection to the steamline is blocked. When draining the vessel during shutdown, air enters the vessel through the vent.

Steam Nozzle with Flow Restrictor (3)

The ABWR RPV has flow restricting venturi located in the steam outlet nozzles. Besides providing an outlet for steam from the reactor pressure vessel, the steam outlet nozzles will provide for (1) steamline 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.

RPV Stabilizer (4)

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.

Feedwater Nozzle Thermal Sleeve (5)

The feedwater nozzles utilize double thermal 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.

Use of Forged Shell Rings (6)

The ABWR RPV utilizes low alloy forged shell rings, per ASME SA-508, Class 3, adjacent to and below the core belt line region. The flanges and large nozzles are also per ASME SA-508, Class 3. The shell rings above the core belt line region and the main closure are made from low alloy steel plate per ASME A-533, Type B, Class 1. The required

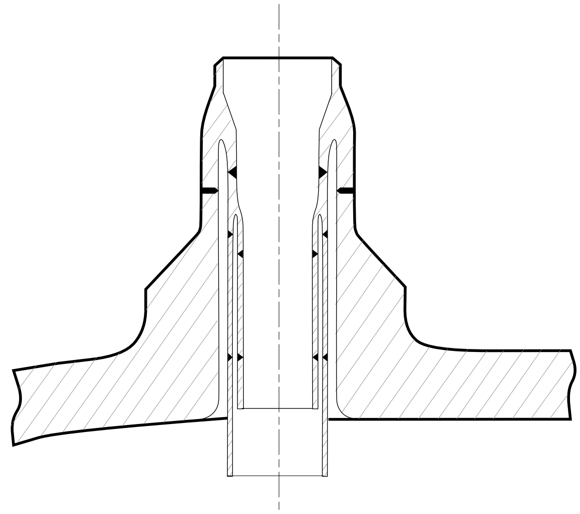


Figure 3-2. ABWR Reactor Pressure Vessel Feedwater Nozzle

Reference Nil Ductility, RTNDT, of the vessel material is -20°C. Figure 3-3 shows one of the RPV forged shell rings during fabrication.

Vessel Support Skirt (7)

The vessel support skirt has a conical geometry and is attached to the lower vessel cylindrical shell course. The support skirt attachment (knuckle) is an integral part of the vessel shell ring. Locating the conical support skirt on the lower shell ring provides:

- Needed space for the reactor internal pump (RIP) heat exchangers.

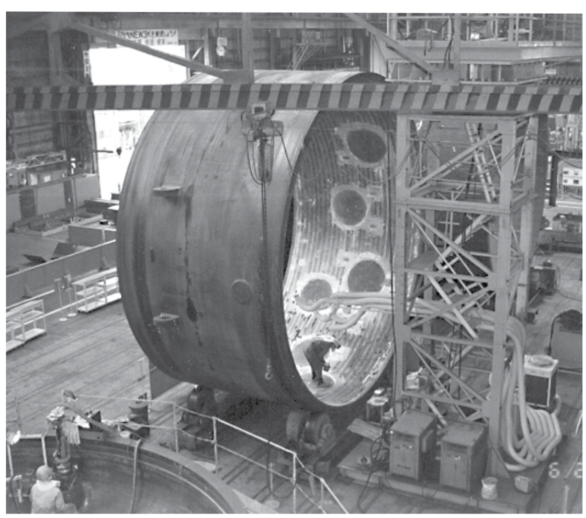


Figure 3-3. RPV Forged Steel Ring

- Access for ISI of the bottom head weld.

Reactor Vessel Bottom Head (8)

The bottom head consists of a spherical bottom cap, made from a single forging, extending to encompass the CRD penetrations and a conical transition section to the toroidal knuckle between the head and vessel cylinder. With bottom head thickness of approximately 250 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 in-service inspection (ISI) requirements.

Reactor Internal Pump Penetrations and Weld (9)

The most significant difference between the ABWR and earlier BWR product lines is elimination of all major pipe connections below the core by incorporating internal recirculation pumps in the reactor. The RIP motor casings are welded to the vessel bottom head by a design as shown in Figure 3-4.

Vertical restraints are provided to prevent the motor casing or the motor cover from blowing out in the unlikely event of a failure of the weld between the RPV and the motor casing or a failure of the motor cover bolts. In addition, if the restraints should fail, the pump impeller is designed to backseat on the stretch tube that keeps the pump diffuser in place and prevent significant leakage through the failed part. More information about the RIP can be found under Recirculation System later in this chapter.

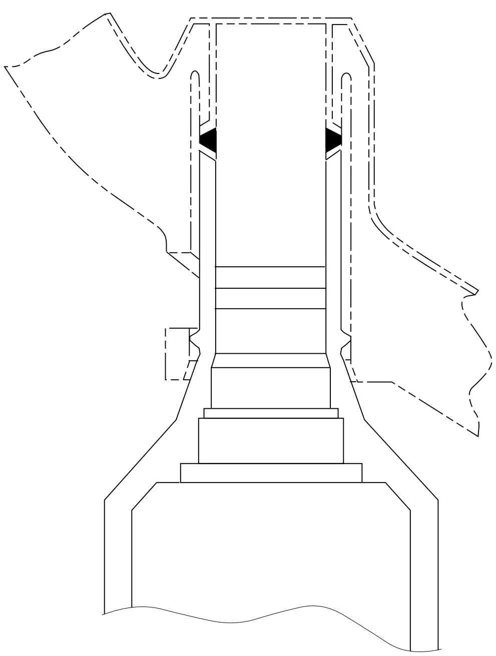


Figure 3-4. Reactor Internal Pump Motor Casing Including Weld to RPV

Thermal Insulation (10)

The RPV insulation is reflective metal type, constructed entirely of series 300 stainless steel. The insulation is made up of a combination of two basic shapes: flat panels and cylindrical panels. The insulation for the bottom head and lower shell course inside the vessel support is a vertical cylindrical panel approximately 75 to 100 mm thick. There is also a horizontal panel of the same thickness which connects across the bottom of the vertical panels. The CRD housings, in-core housings and drain nozzles penetrate this panel.

The insulation for the RPV is supported from the reactor shield wall surrounding the vessel, and not from the vessel shell. Insulation for the upper head and flange is supported by a steel frame independent of the vessel.

At operating conditions, the approximate air temperatures outside the vessel and insulation are 57°C above the top head and 38°C everywhere else.

Core Shroud (11)

The shroud is a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. The volume enclosed by the shroud is characterized by upper and lower regions. The upper portion of the shroud surrounds the active fuel and forms the longest section of the shroud. This section is bounded at the bottom by the core plate. The lower shroud, surrounding part of the lower plenum, is welded to the RPV shroud support. The shroud provides lateral support for the core by supporting the core plate and top guide.

Core Plate (12)

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 support and through the fuel assemblies.

Top Guide (13)

The top guide consists of a grid that gives lateral support of the top of the fuel assemblies, a cylinder supporting core flooder spargers, and a top flange for attaching the shroud head. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, one, 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 shroud.

Fuel Supports (14)

The fuel supports are of two basic types; namely, peripheral fuel supports and orificed 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 orificed 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 orificed 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 orificed fuel support.

Control Rod Drive Housing (15)

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, orificed fuel support and four fuel assemblies.

Control Rod Guide Tubes (16)

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 an orificed fuel support. This locates the four fuel assemblies surrounding the control rod drive housing, 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 FMCRD to restrain a hypothetical ejection of the FMCRD.

In-core Housing (17)

The in-core housings provide extensions of the RPV at the bottom head for the installation 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.

In-Core Guide Tubes and Stabilizers (18)

The in-core guide tubes extend from the top of the in-core housing to the top of the core plate. They provide the in-core instrumentation with protection from flow of water in the bottom head plenum, and guidance for insertion and withdrawal from the core. The in-core guide tube stabilizers provide lateral support and rigidity to the in-core guide tubes.

Feedwater Spargers (19)

The feedwater spargers are attached to brackets on the vessel wall and deliver makeup water to the reactor during plant startup, power generation and plant shutdown modes of operation. Nozzles in the spargers provide uniform distribution of feedwater flow within the downcomer flow passage.

High Pressure Core Flooder Sparger Assembly (20)

The high pressure core flooder (HPCF) spargers inside the cylinder of a top guide are arranged to provide emergency coolant injection over the upper end of the core. The spargers have the function of a standby liquid control solution injection. The HPCF spargers are connected to the HPCF nozzles by means of an HPCF coupling (21).

Low Pressure Flooder Spargers (22)

The two flooding spargers that are attached to the vessel wall deliver flow at low pressure from the RHR System and distribute it in the upper plenum above the shroud head of the reactor. Flow is delivered in either of two modes: (1) for the flooding of the reactor in the event of an abnormal drop in water level, or (2) in the circulation of cooling water to remove residual and core decay heat from the reactor during shutdown.

Shutdown Cooling Nozzles (23)

Suction for the RHR System in shutdown cooling mode is provided by three shutdown cooling nozzles.

Shroud Head and Steam Separator Assembly (24)

The steam separator assembly consists of a slightly domed 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 core shroud and forms the cover of the core discharge plenum region. The seal between the separator assembly and core shroud flanges is metal-to-metal contact and does not require a gasket or other replacement sealing devices. The separator assembly is bolted to the core shroud flange, by long holddown bolts which, for ease of removal, extend above the separators. During installation, the separator base is aligned on the core shroud 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 top guide 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 3-5). The separated water 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.

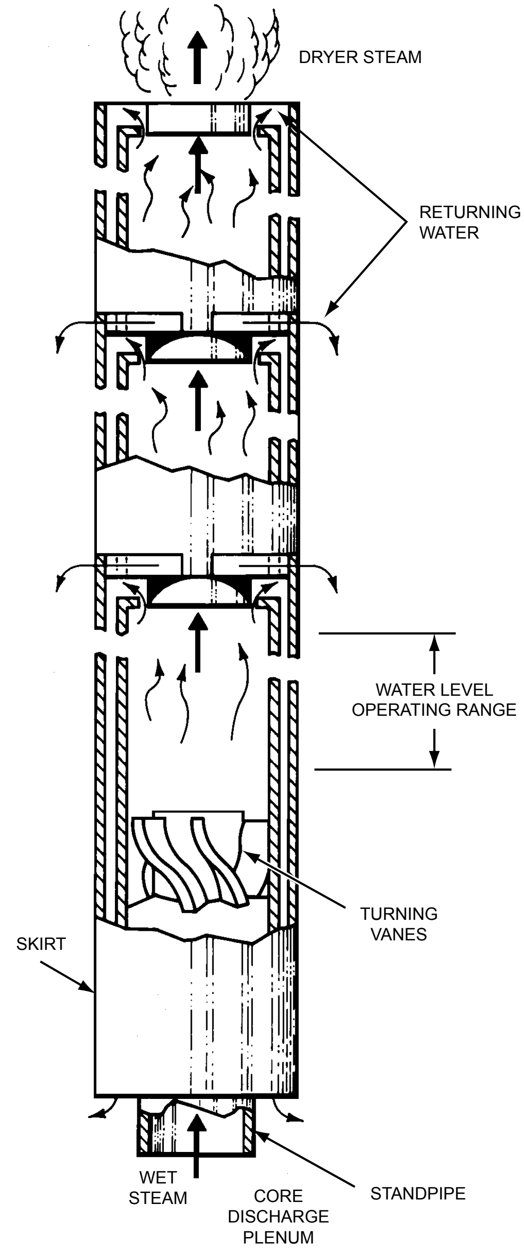


Figure 3-5. Schematic of Steam Flow through Separator

Steam Dryer Assembly (25)

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 trough, drain ducts, and a skirt which forms a water seal extending below the separator reference zero elevation. Steam from the separators flows upward and outward through the drying vanes (Figure 3-6). 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 recirculation downcomer annulus between the core shroud 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 holddown brackets inside the main closure. The assembly is arranged for removal from the vessel as an integral unit on a routine basis.

Reactor Internal Pumps (RIP) (26,27)

Refer to the next section for information on Reactor Internal Pumps (RIP).

Core and RIP Differential Pressure Lines (28)

These lines comprise the core flow measurement subsystem of the Recirculation Flow Control System (RFCS) and provide two methods of measuring the ABWR core flow rates. The core ΔP lines and internal pump ΔP lines enter the reactor vessel separately through reactor bottom head penetrations. Four pairs of the core ΔP lines enter the head in four quadrants through four penetrations and terminate immediately above and below the core plate to sense the pressure in the region outside the bottom of the fuel assemblies and below the core plate during normal operation. Similarly, four pairs of the internal pump ΔP lines terminate above and below the pump deck and are used to sense the pressure across the pump during normal pump operation.

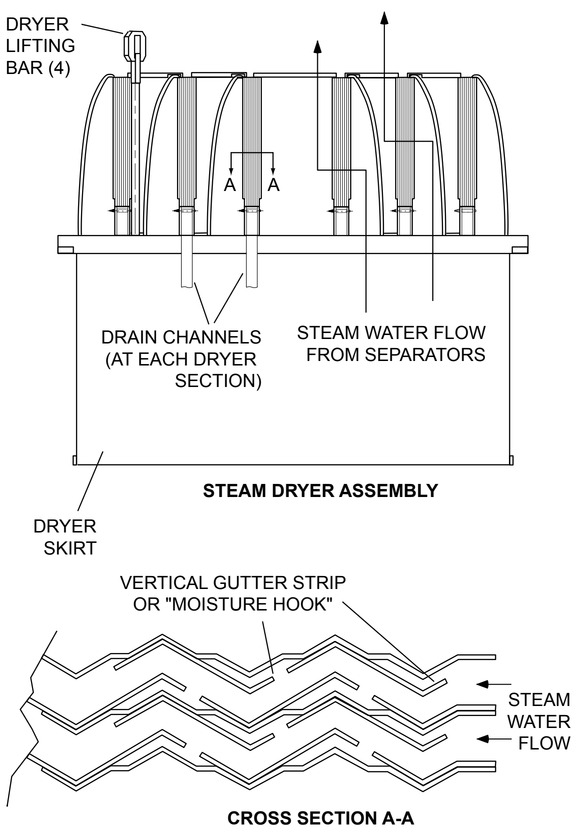


Figure 3-6. Schematic of Steam Flow Through Dryer

Fine Motion Control Rod Drives (29)

Refer to the discussion on the Control Rod Drive System later in this chapter.

Fuel Assemblies, Control Rods and Local Power Range Monitors (30-32)

Refer to the Chapter 6 discussion for fuel and related hardware.

Recirculation System

The function of the Reactor Recirculation System (RCIR) is to:

- Provide forced circulation of reactor coolant for energy transfer from fuel to the cooling fluid and, as a result, generate a larger amount of steam.
- Control the reactor power by changing the recirculation flow; the flow is controlled by the use of adjustable speed pumps.

The RCIR System provides forced circulation of reactor water through the core, removing the heat produced by the fuel. The reactor water is made up of water removed from the two-phase reactor coolant (core flow) in the moisture separators and steam dryers and the incoming feedwater flow. The RCIR System uses an arrangement of ten pumps to provide the motive force for core flow. The pumps are mounted internally in the reactor vessel and are called reactor internal pumps (RIPs). The RIPs function collectively to force the reactor coolant through the lower plenum of the reactor and upward through openings in the fuel support castings, through the fuel bundles, steam separators, and down the annulus to be mixed with feedwater and recirculated through the core. Figure 3-7 shows the RIPs and the pumped flow path.

Recirculation flow rate is variable over a range from natural circulation flow of 20% to above the rated flow required to achieve rated core power. In fact, the RCIR design can produce rated core flow rate at 100% reactor power with nine of its ten pumps operating. The flow control range allows automatic regulation of reactor power output between ~70 to 100% without control rod movement. Core flow (RCIR pumping capacity) is regulated by the Recirculation Flow Control System (RFC). The RFC System provides conditioned control and logic signals, which regulate the RIP speed, which, in turn, regulates the pump flow. Because the core flow affects reactor power and fuel thermal margins, the RCIR System is also used to mitigate the effects of transient, upset and emergency modes of reactor operation.

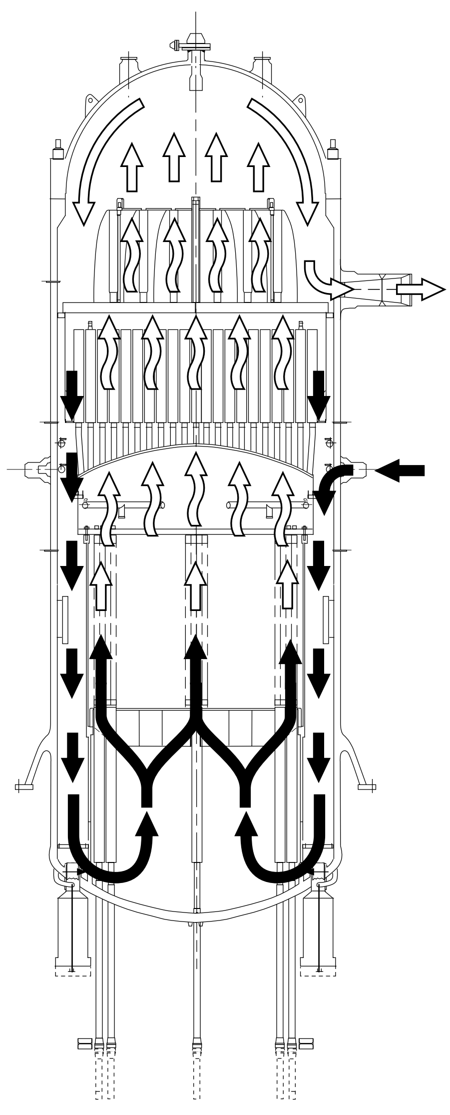


Figure 3-7 Recirculation Flow Schematic

There are three RCIR subsystems which are used in conjunction with the reactor internal pump:

- Recirculation Motor Cooling Subsystem (RMC Subsystem)
- Recirculation Motor Purge Subsystem (RMP Subsystem)
- Recirculation Motor Inflatable Shaft Seal Subsystem (RMISS)

Recirculation Motor Cooling Subsystem (RMC)

Each RIP has its own external heat exchanger (Figure 3-8). Each RIP motor casing and the RIP heat exchanger is connected via stainless steel piping. The heat exchanger is a typical shell-tube type with U-tubes supported by baffles. The hot water coming from the motor enters from the upper end of the heat exchanger shell side and leaves from the lower end of the shell side and returns back to the motor. The connecting piping is welded to the RIP motor casing and also to the heat exchanger shell to prevent any leakage during the plant operation.

Recirculation Motor Purge Subsystem (RMP)

The purge system provides a source of clean control rod drive (CRD) water that flows up the annulus between the stretch tube and the shaft and prevents the intrusion of reactor water with its associated contamination into the motor. The purge system normally operates continuously even when the RIPs are tripped or the reactor is shutdown for the refueling and/or maintenance.

Recirculation Motor Inflatable Shaft Seal Subsystem (RMISS)

During normal operation, the purpose of this system is to prevent any leakage of reactor water (escaping from the primary seal) during plant outages and to assist in maintenance or inspection of motors. During the maintenance of the RIPs, a portable pump is used to pressurize the seal using water from the Makeup Water System (MUW). The seal is made of elastomeric material and seals tightly between the pump shaft and the pump motor casing.

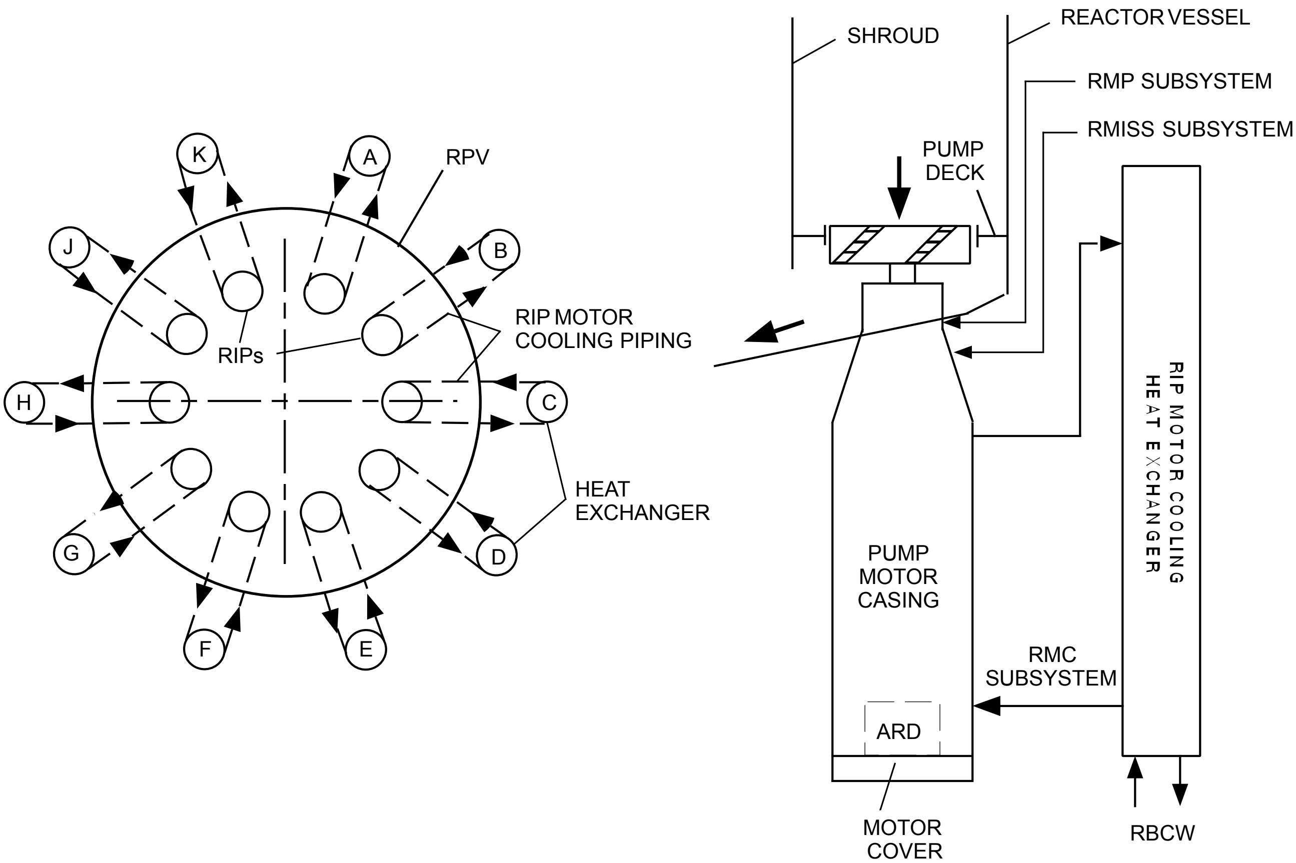


Figure 3-8. RCIR Subsystems

The pump pressurizes the seal and maintains it at shutoff head conditions.

Reactor Internal Pumps

The vessel-mounted RIPs simplify the RPV by eliminating all large nozzles below the core, significantly reducing piping in-service inspection (ISI) and personnel exposure, and allowing for a compact containment design due to elimination of the external recirculation system piping. Use of the RIP feature allows for the optimization of the Emergency Core Cooling System (ECCS) and assures no core uncovery for postulated pipe breaks. One of the goals for the ABWR is to reduce calculated core damage frequency by an order of magnitude relative to GE's previously designed BWR operating plants. One of the most important design features contributing to this goal was the adoption of RIPs in place of externally pumped reactor recirculation system/pumps.

The internal pumps are an improved version of a European designed RIP that is in operation in many European nuclear power plants. About 9 million pump hours of successful operating experience has been accumulated, with some pumps having been in service since the mid-1970's.

The general design details of the RIP, motor, and heat exchanger are as follows:

Number of Pumps:	10
Type of Pump:	Vertical shaft, single stage, mixed flow
Rated Flow:	7700 m^3/hr/pump
Rated Head:	40 m
Rated Pump Speed:	1500 rpm
Overall Height (Impeller & Motor):	3 m
Overall Weight:	5000 kg
Motor Type:	3-Phase, Wet Induction Motor

Rated Output Power:	830 kW
Rated Voltage:	~ 3300 V
Heat Exchanger Type:	Shell and Tubes
Hx Cooling Capacity:	1.15 kcal/hr

Reactor Internal Pump Component Description

There are 10 RIPs arranged circumferentially between the shroud and the RPV near the RPV bottom head. Figure 3-9 shows a cross section of the RIP used in the ABWR and key components are described below.

Diffuser: The RIP has the impeller and diffuser inside the RPV. The diffuser is installed in the pump deck and sealed by a piston ring arrangement. The RIP diffuser is removable. The diffuser is retained on the RPV nozzle by the stretch tube.

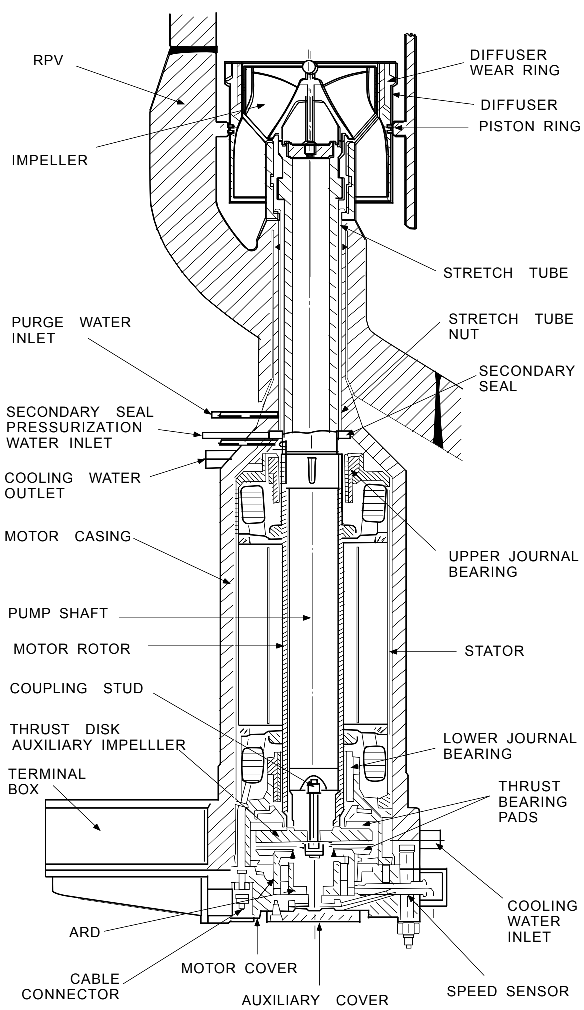


Figure 3-9 Cross-Section of RIP

Stretch Tube: The stretch tube is essentially a long hollow bolt which passes through the RPV nozzle penetration from the diffuser to the top of the RIP motor casing where it is held in tension by a large nut. The stretch tube is preloaded by use of a stud tensioner similar to that used for the main closure studs of the RPV. The pump shaft passes through the center of the stretch tube and motor rotor. The pump shaft key fits in a slot in the motor rotor tube.

Impeller and Pump Shaft: The impeller and the pump shaft are connected by the impeller bolt. The pump shaft passes through the stretch tube, rotor and is connected to the thrust bearing disk at its lower end. The motor rotor keyway slot fits with the key on the pump shaft and transmits motor torque.

Radial Bearings: The motor upper radial bearing is below the secondary seal. This bearing design has been tested and proven to eliminate bearing instability due to half speed rotation.

The lower radial bearing is located below the motor rotor and the stator. The lower radial bearing is similar to the upper radial bearing.

Thrust Bearing: The thrust bearing is of an offset tilting pad configuration. The rotating portion of the thrust bearing is integral with the cooling water auxiliary impeller, which circulates water through the motor and bearing to provide cooling and cleaning via the purge system.

Anti-Reverse Rotation Device: Below the cooling water auxiliary impeller is the Anti-Reverse Rotation Device (ARD). This is a cam clutch arrangement that prevents the RIP from rotating in the reverse direction when one RIP is tripped while the others are running (which can result in backflow through the tripped RIP). The purpose of the ARD is to prevent reverse rotation of the pump shaft and minimize the backflow through an idle/tripped RIP.

Motor (Stator and Rotor): The RIP motor is a 3-phase, 4-pole wet induction motor. The cooling water flows upward through the windings of the stator and the rotor. The motor stator is attached to the motor cover.

Terminal Box: The electrical terminal box is bolted to the motor cover. The motor winding cable penetrations pass through the motor cover coolant pressure boundary and are connected to the power supply leads at this location. Each motor is driven by its own variable frequency power supply known as the Adjustable Speed Drive (ASD).

Speed and Vibration Sensors: There are 2-pump speed sensors and 2-motor casing vibration sensors on each RIP motor casing. There is an additional sensor on each RIP to detect rubbing of internal parts of the pump.

RIP Operation

Whenever the RIP motor is started, it is controlled to reach its minimum speed. Similarly, one by one, the other 9 RIPs are started and brought to the minimum speed level. From this condition, the speed of all 10 RIPs can be increased individually when in the individual speed control mode, or as a group when in the automatic ganged mode of operation, with the ganged mode being the normally preferred mode after all 10 RIPs have been started. The RFC System controls the speed of the RIPs as described earlier in this section. A change in RIP speed conditions will vary the core flow in the reactor, which, in turn, will change the reactor power during normal power range operation.

The RFC operational modes also include the core flow control mode and the automatic load-following mode. The core flow mode controls the speed of the RIPs selected for gang speed operation to maintain the steady-state core flow equal to the core flow demand signal. For the automatic load-following mode, the RFC System controls the speed of those RIPs selected for gang speed operation to reduce the load demand error signal (from the turbine control system) to zero.

Individual RIP speed control operation mode and the ganged speed mode of operation provide significant flexibility during normal plant operation. If, for any reason, one RIP develops a problem, then either speed can be reduced to eliminate the problem or that RIP can be tripped, if necessary, without affecting the continued operation of other RIPs.

During normal plant rated power operation (in either the core flow control mode or the automatic load-following mode), if one RIP is lowered in speed or tripped, then the speed of the remaining 9 RIPs is increased by the control system to maintain the demanded core flow; thus, steady-state plant output power remains unaffected.

RIP Power Supply

The RIP motor is driven by a solid-state variable-frequency power supply known as the Adjustable Speed Drive (ASD). The ASD is a proven product with wide industrial applications as well as experience in the European nuclear plants. The ABWR application uses ~3000 V for the output voltage rating. The ASD power supply provides extremely low maintenance, high reliability, and provides excellent RIP speed maneuverability.

Each RIP is driven by its dedicated ASD. Six RIP ASDs receive power from constant speed Motor-Generator (M-G) sets and the other four directly from medium voltage buses. A representative simplified power distribution one-line diagram is shown in Figure 3-10. Each M-G set provides power to 3 associated RIP ASDs. The other four RIP ASDs are divided into two sets receiving power directly from two separate main buses.

The assignment of the power distribution

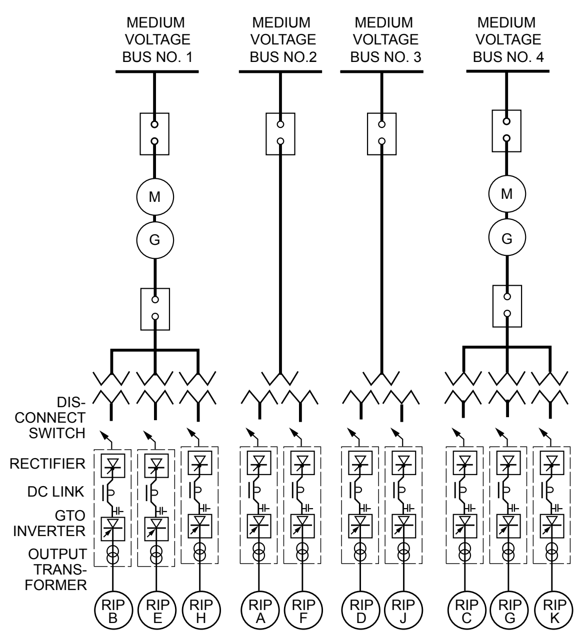


Figure 3-10. RIP Power Supply Diagram

to individual RIP ASDs is chosen to balance the azimuthal distribution within the vessel (e.g., when a M-G set trips or one medium voltage bus is lost). The M-G sets have inertial flywheels to provide continued operation of the associated RIPs during either the momentary or complete loss of incoming power. After complete loss of the main bus power, continued operation of these RIPs for at least 3 seconds is provided via the M-G sets.

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 (RCIS). The CRD System provides rapid control rod insertion in response to manual or automatic signals from the Reactor Protection System (RPS). Figure 3-11 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. Thus, the hydraulic scram action is backed up by an electrically energized insertion of the control rods.

The CRD System consists of three major elements:

- Electro-hydraulic 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

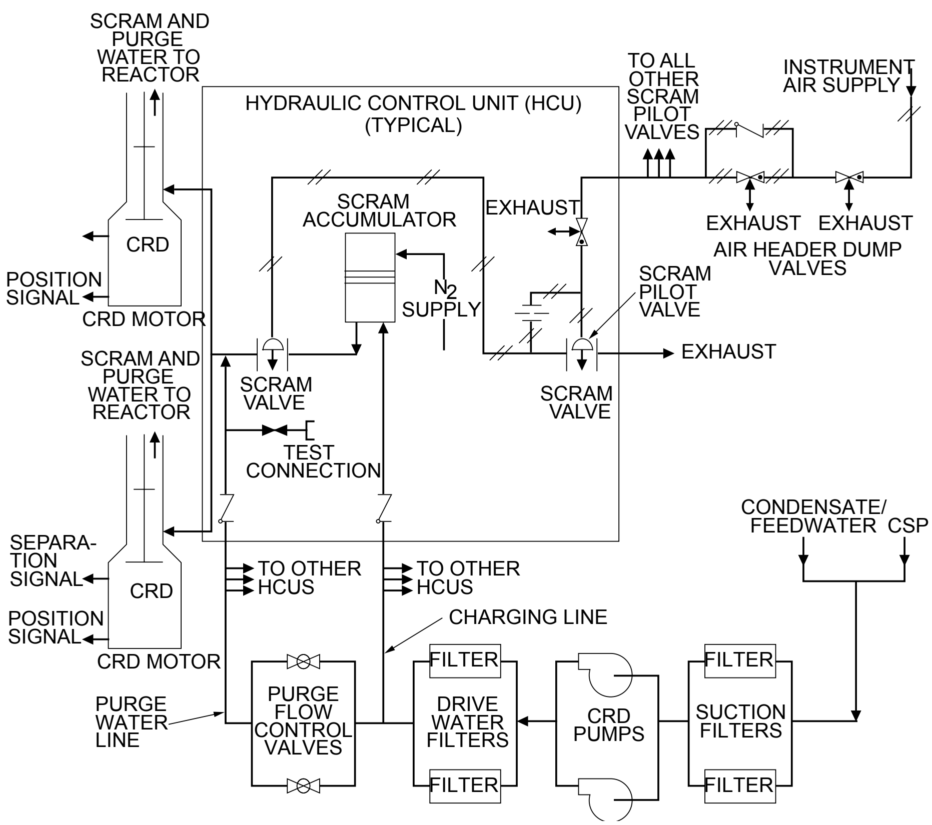


Figure 3-11. CRD System Schematic

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. 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.

Fine Motion Control Rod Drives

The ABWR FMCRDs are distinguished from the locking piston CRDs, which are in operation in all current GE 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 ATWS events. The FMCRD design is an improved version of similar drives that have been in operation in European BWRs since 1972.

Figure 3-12 shows a cross-section of the FMCRD as used in the ABWR. The FMCRD consists of four major subassemblies: the drive, the spool piece, the brake and the motor/synchros. The spool piece and motor may be removed without disturbing the drive and this allows maintenance with low personnel exposure.

The drive consists of the outer tube, hollow piston, guide tube, buffer, labyrinth seal, ball check valve, spindle adaptor and splined spindle adaptor back seat.

The coupling is a bayonet 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. 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.

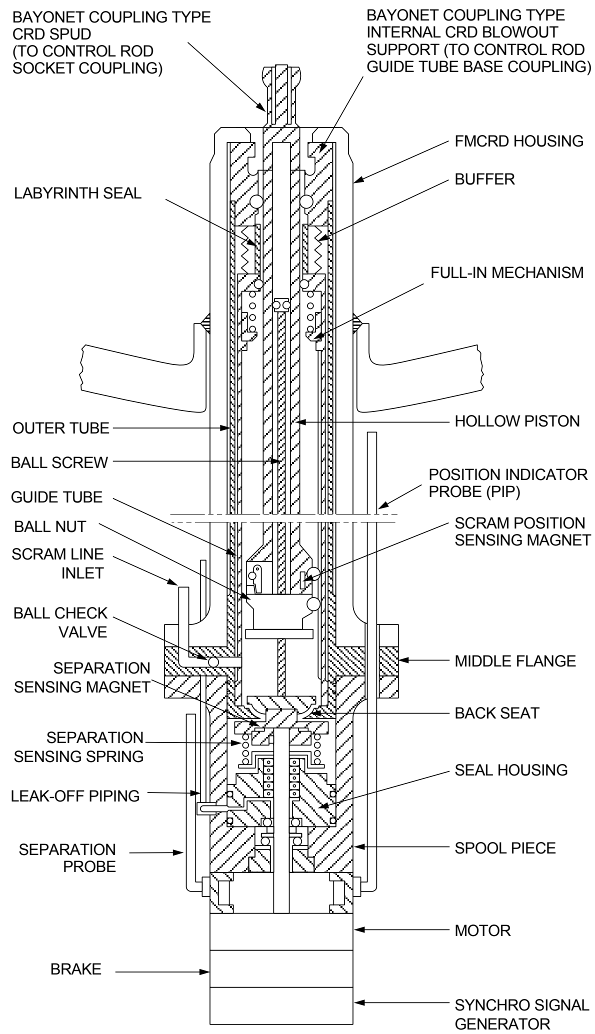


Figure 3-12. Fine Motion Control Rod Drive Cross-Section

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 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 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 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 along the ball screw resulting from rotation of the ball spindle. 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 spindle adaptor and splined spindle adaptor back seat provide a back seat type anti-withdrawal latch feature, which consists of a gear that automatically engages whenever the spool piece is lowered. This prevents the ball spindle from rotating and withdrawing the rod.

The spool piece contains a redundant packing type shaft seal to preclude leakage from the drive. The spool piece also contains radial and thrust bearings for the stub shaft that is also part of the spool piece. The stub shaft transmits torque from the motor to the ball screw via a slip coupling. The spool piece also contains the weighing platform.

The weighing device is a spring-loaded platform with two magnets located on it. In normal service, 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. The two external reed switches are called separation switches and, 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.

The spool piece is bolted to the CRD housing by bolts which pass through the outer tube flange. As mentioned above, the outer tube 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 bolts to the spool piece through a motor bracket. The motor is a stepping motor similar to that used in robotic and precision positioning applications. The use of the stepping motor is the major change from the configuration used in Europe. The European drives used a gear motor and, because of this, had less precise positioning capability than the ABWR FMCRD. In addition, the gear motor required increased maintenance compared to the stepping motor. The stepping motor design is based on motors that have had successful experience in industrial applications.

There are two synchro-type position indicators located below the stepping motor. The synchros provide a continuous readout of the rod position during normal operation and are driven by gears from the motor shaft.

The brake is mounted between the motor and the synchros. The brake 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. Some key FMCRD parameters are:

Step Size	18.3 mm
Movement Speed	30 mm/sec
Scram Time 60%	1.7 sec

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 Class 1E.

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

Power to the FMCRD is provided by a solid state, thyristor-driven power supply. The power supply is based on proven products with successful experience in industrial applications. The power supply is a variable voltage and frequency device which starts the motor at low speed, accelerates it to the normal run speed and then slows it when approaching the specified position. The power supply interfaces with the Rod Control and Information System (RCIS). The power supply interfaces with the Rod Control and Information System (RCIS). The power supply includes the capability to move the individual drive, while the RCIS provides the logic and control for overall control rod motion.

The DC power for the brake, which is an energize-to-release model, is supplied by an inverter which is integrated with the motor power supply cabinets.

Hydraulic Control Units

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 ABWR, there is one HCU for every two FMCRDs rather than the one HCU per CRD as in past GE plants. 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 ABWR 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 RIP and the Reactor Water Cleanup (RWCU) System pump.

The CRD pump is basically the same as that used in BWR/6 plants (i.e., a multi-stage centrifugal pump). The filtration system is basically also the same as that used on BWR/6.

Main Steam System

The purpose of the Main Steam System (MS) is to direct steam flow from the RPV steam outlet nozzles to the main turbine. A main steamline flow restrictor is provided in each steam outlet nozzle. It is designed to limit the flow rate in the event of a postulated steamline break. The system also incorporates provisions for relief of over-pressure conditions in the RPV.

In the ABWR design, four 28-inch steamlines transport steam from the steam outlet nozzles on

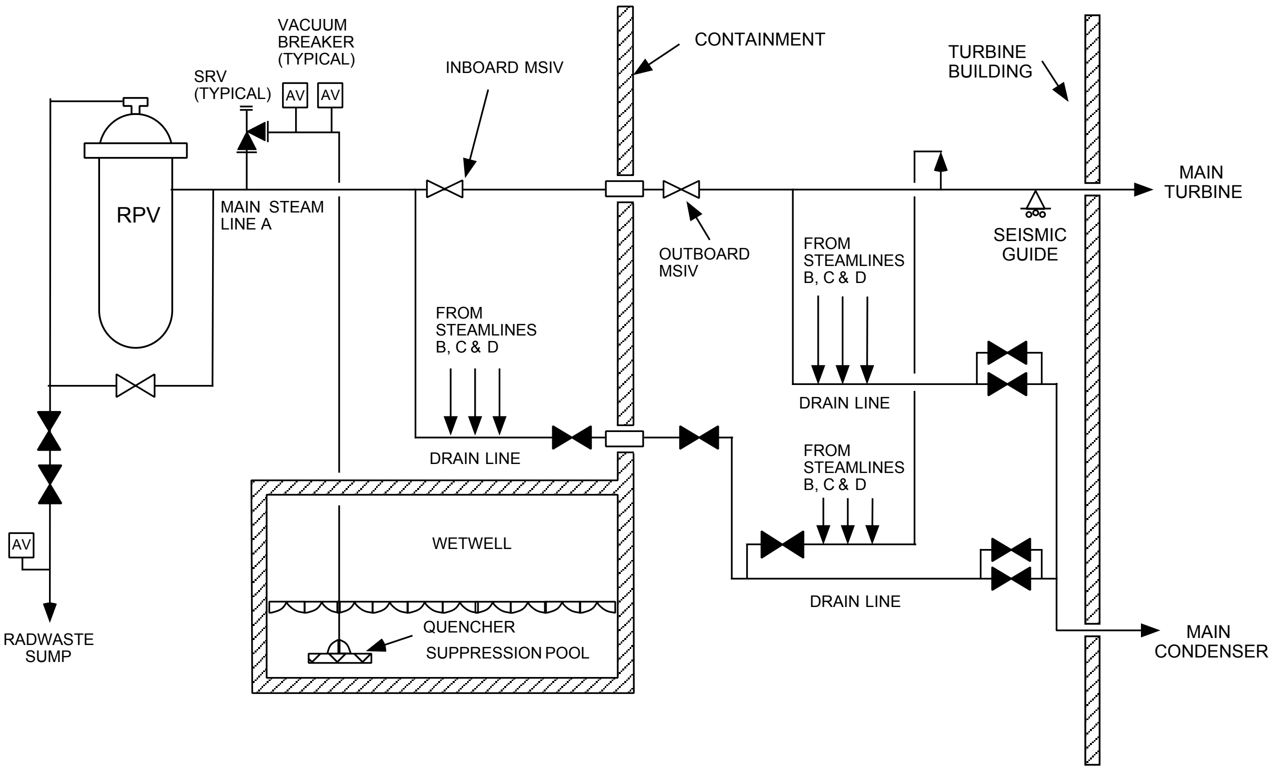


Figure 3-13. Main Steam System

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 steamline inboard and outboard of the RCCV penetrations. Eighteen safety/relief valves (SRVs) are installed on horizontal steamline headers and the discharge from each SRV is routed through the associated SRV discharge line to quenchers located in the suppression pool. Of the 18 SRVs, 8 provide the Automatic Depressurization System (ADS) function during an accident condition. Figure 3-13 is a simplified piping diagram of the MSS, and Figure 3-14 illustrates the three dimensional configuration of the piping, MSIVs and SRVs.

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, SRV and ADS, reactor head vent subsystem, and system instrumentation.

Main Steam Isolation Valves

Two MSIVs are welded in a horizontal run of each of the four main steam pipes. The MSIVs are designed to isolate primary containment upon receiving an automatic or manual closure signal, thus limiting the loss of coolant and the release of radioactive materials from the nuclear system.

Each MSIV is a Y-pattern, globe valve and is powered by both pneumatic pressure and compressed spring force (Figure 3-15). The main disk assembly is attached to the lower end of the stem. Normal steam flow tends to close the valve and the pressure is over the disk. The bottom end of the valve stem or a stem disk attached to the stem closes a small pressure balancing hole in the main disk assembly. When the hole is open, it acts as an opening to relieve differential pressure forces on the main disk assembly. Valve stem travel is sufficient to provide flow areas past the wide open main disk assembly greater than the seat port area. The main disk assembly travels approximately 90% of the valve stem travel to close the main seat port

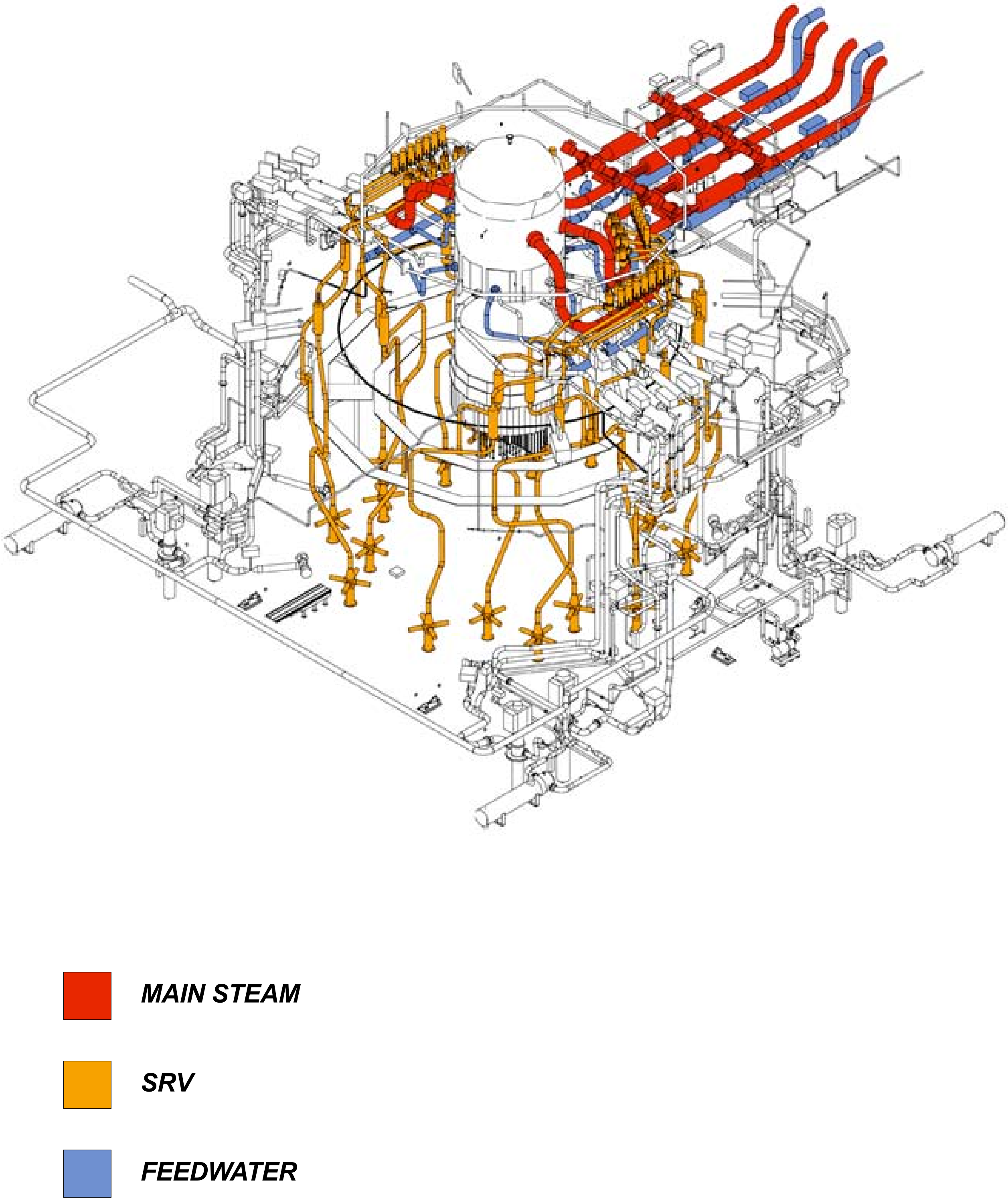


Figure 3-14. Main Steam System Piping Schematic

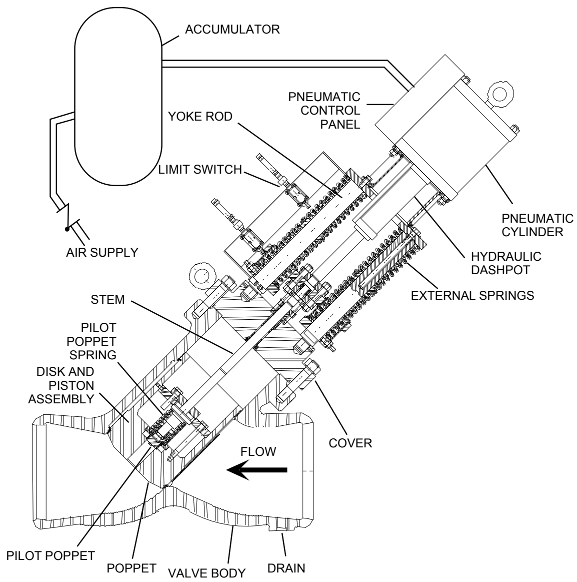


Figure 3-15. Main Steam Isolation Valve

area; approximately the last 10% of the valve stem travel closes the pilot seat. The air cylinder actuator can open the main disk assembly with a maximum differential pressure of 1.38 MPaG (200 psig) across the isolation valve in a direction that tends to hold the valve close. The Y-pattern valve permits the inlet and outlet passages to be streamlined; this minimizes pressure drop during normal steam flow and helps prevent debris buildup on the valve seat.

Attached to the upper end of the stem is an air cylinder that opens and closes the valve and a hydraulic dashpot that controls its speed. The speed is adjusted by hydraulic control valves in the hydraulic return lines bypassing the dashpot piston.

The valve is designed to close quickly when nitrogen or air is admitted to the upper piston compartment to isolate the MS in the event of a LOCA, or other events requiring containment or system isolation to limit the release of reactor coolant. The valve can be test closed one at a time at a slow closing speed by admitting nitrogen or air to both the upper and lower piston compartments. This is to ensure that the slow valve closure does not produce a transient disturbance large enough to cause a reactor scram.

When all the MSIVs are closed, the combined leakage through the MSIVs for all four steamlines is monitored to within the offsite radiation dose release limit.

Nitrogen is used for the inboard MSIV operation because of the inerted drywell environment where the inboard MSIVs are located. Instrument air is used for the outboard MSIV operation.

A separate pneumatic accumulator is provided and located close to each MSIV and supplies pressure as backup operating gas to assist in valve closure in the event of a failure of pneumatic supply pressure to the valve actuator.

Safety/Relief Valves

The SRV is a dual function, direct-acting valve and is classified as safety-related (Figure 3-16). The SRV is considered as part of the RCPB because the inlet side of the valve is connected to the steamline prior to the inboard MSIV. The SRV logic and solenoids are also classified and qualified as Class 1E per the IEEE Standards. This classification is also applied to the ADS function and other associated systems.

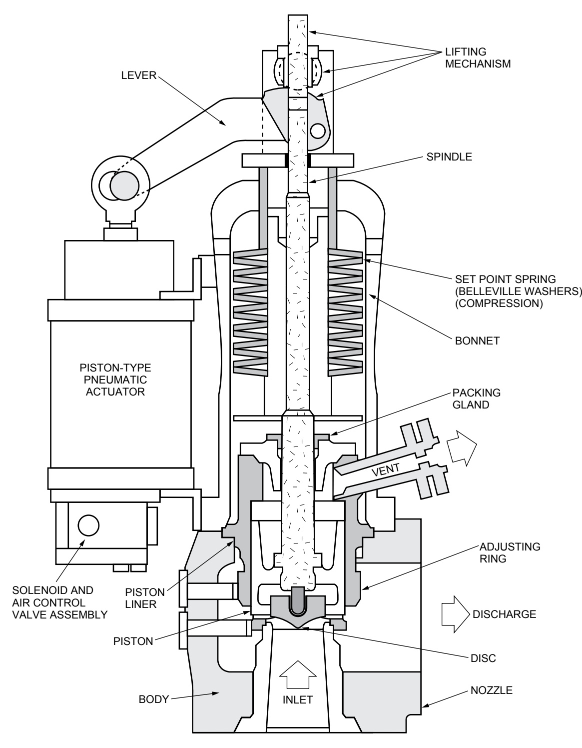


Figure 3-16. Safety/Relief Valve

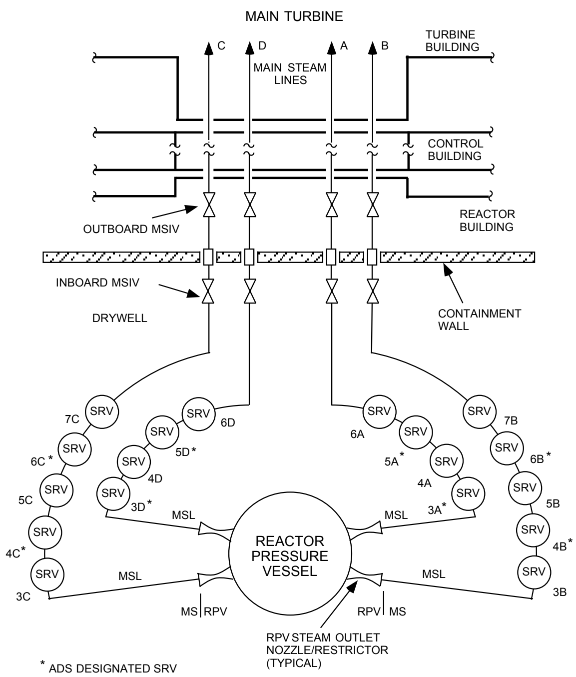


Figure 3-17 MSIV and SRV Configuration

The SRV capacity is sized to maintain primary system pressure below the ASME Code design limits. Figure 3-17 illustrates the SRV and MSIV configuration.

The SRVs are located on the main steamlines between the RPV and the inboard MSIV. These valves provide three main protection functions:

- **Overpressure Safety Operation:** The valves function as spring-loaded safety valves and open to prevent RCPB overpressurization. The valves are self-actuated by inlet steam pressure.
- **Overpressure Relief Operation:** The 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. This action pulls the lifting mechanism of the main disk, thereby opening the valve to allow 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.
- **Automatic Depressurization System (ADS) Operation:** The ADS valves open automatically or manually in the power actuated mode when required during a LOCA. The ADS designated SRVs open automatically as part of the Emergency Core Cooling System (ECCS) as required to mitigate LOCA when it becomes necessary to reduce RCPB pressure to admit low pressure ECCS coolant flow to the reactor.

Eight of the 18 SRVs are designated for the ADS function, and are equipped with three separate solenoid valves powered by 125 VDC. The other non-ADS SRVs are each equipped with one solenoid valve powered by 125 VDC.

The SRVs are divided into five setpoint groups to relieve the RPV pressure in accordance with the RPV overpressure protection evaluation.

A separate pneumatic accumulator is provided for each SRV function and is located close to each SRV to supply pressure for the purpose of valve actuation. SRVs that are designated for ADS function are provided with one additional accumulator for each valve.

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 SRV 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. The SRV discharge lines terminated at the quenchers located below the surface of the suppression pool (SP).

Feedwater System (Nuclear Island)

Two 22-inch 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 12-inch riser lines that connect to nozzles on the RPV (see Figure 3-18). Isolation check valves are installed upstream and downstream of the RCCV penetrations, and manual maintenance gate valve are installed in the 22-inch lines upstream of the horizontal headers. Also shown in the figure are the interconnections from the RCIC, RHR and RWCU systems.

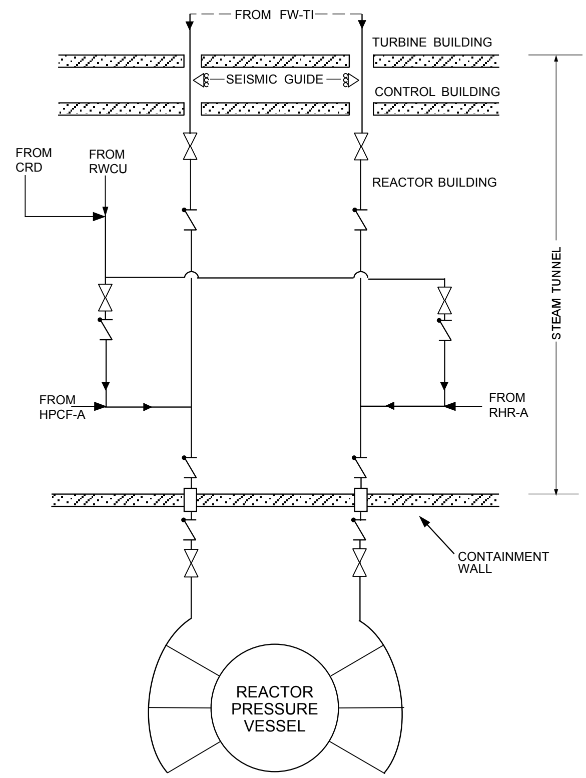


Figure 3-18. Feedwater System (Nuclear Island)

Chapter 4
Safety Systems

Overview

The ABWR Safety Systems design incorporates three redundant and independent divisions of Emergency Core Cooling Systems (ECCS) and containment heat removal (Figures 4-1 and 4-2). The RPV has no external recirculation loops or large pipe nozzles below the top of the core region. This allows for a reduced capacity ECCS while still keeping the fuel covered for the full spectrum of postulated LOCAs even assuming a single failure. Each of the three divisions within the ECCS network has one high pressure and one low pressure inventory makeup system. The high pressure configuration consists of two motor-driven high pressure core flooders (HPCF), each with its own independent sparger discharging inside the shroud, and the steam-driven Reactor Core Isolation Cooling System (RCIC), which discharges into the feedwater injection line. The HPCF pumps provide core makeup over the entire range of system operating pressures. The RCIC System, which has been upgraded to a safety system, has the dual function of providing high pressure ECCS flow following a postulated LOCA and reactor coolant inventory control for reactor isolation transients. The RCIC System, with its steam turbine-driven power, also provides a diverse makeup source during loss of all AC power events. The low pressure ECCS for the ABWR utilizes the three residual heat removal (RHR) pumps in the post-LOCA Low Pressure Flooding (LPFL) mode and are labeled LPFL. For small LOCAs that do not depressurize the reactor system, if the high pressure makeup is unavailable, an Automatic Depressurization System (ADS) actuates to vent steam from the reactor through the safety/relief valves (SRVs) to the suppression pool, and depressurizes the reactor vessel to allow the LPFL pumps to provide core coolant makeup flow.

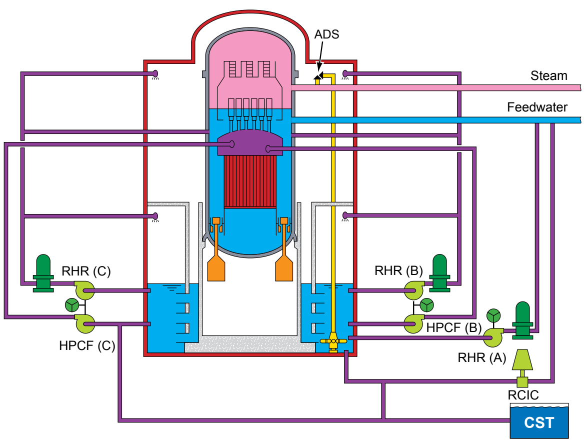


Figure 4-1. ECCS Divisional Configuration

The RHR System has a dual role of providing reactor cooling for normal shutdown and providing core and containment cooling following a postulated LOCA. The ABWR RHR System has been improved such that core and suppression pool cooling are achieved simultaneously since, in the core cooling mode, the flow from the suppression pool passes through the RHR heat exchanger, and the supporting heat removal systems. Reactor Building Cooling Water (RCW) and Service Water (RSW) are also initiated upon a LOCA signal.

As a result of these improvements in the ECCS network and the RHR System, there is an increase in the calculated safety performance margin of the ABWR over earlier BWRs. This has been confirmed by a Probabilistic Risk Assessment (PRA) for the ABWR, which shows that the ABWR is a calculated factor of at least 10 better than BWR/5 and BWR/6 in avoiding possible core damage from degraded events.

In addition to the ECCS, there are several other important safety systems in the ABWR. These in-

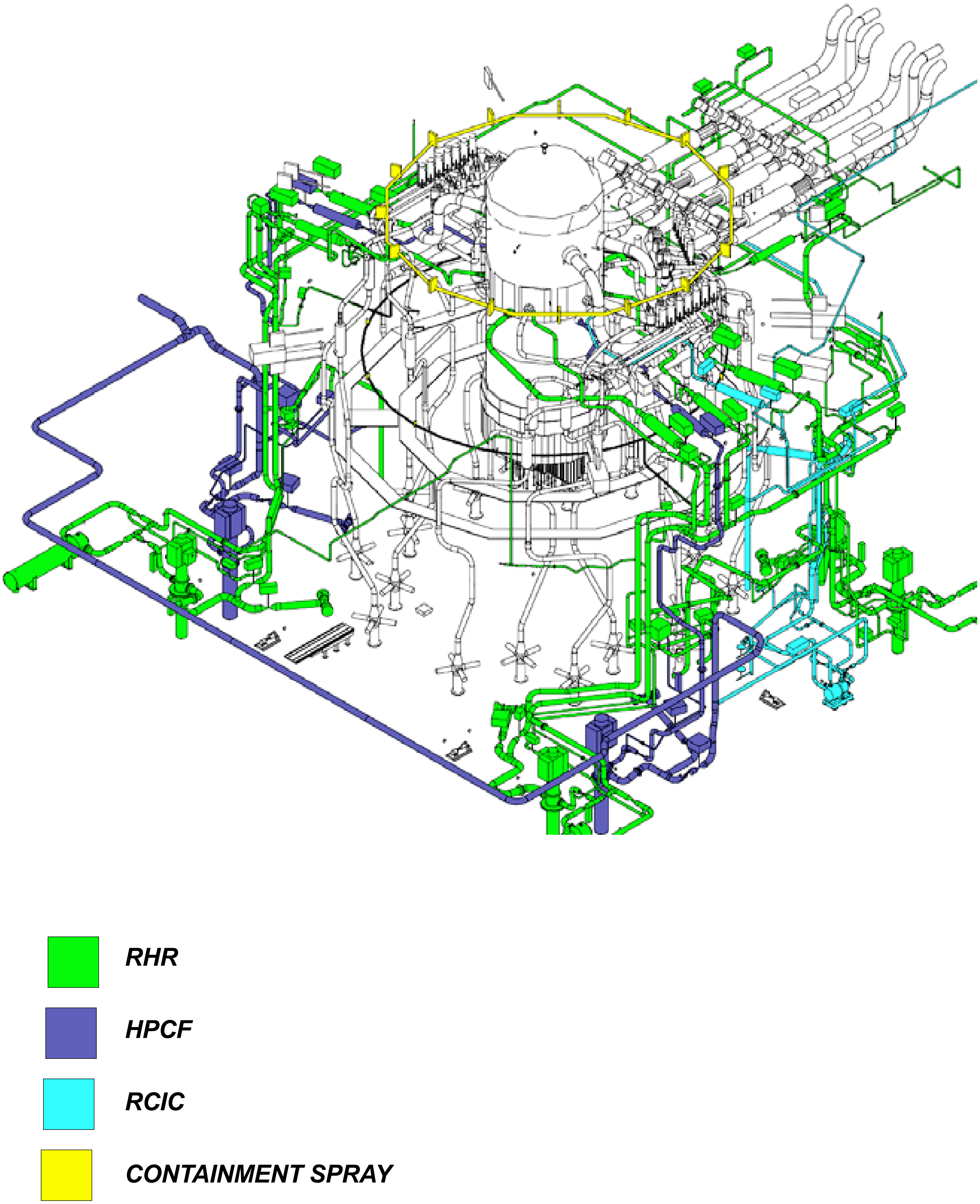


Figure 4-2. ECCS Piping Schematic

clude the Standby Gas Treatment System (SGTS), the Atmospheric Control System (ACS), the Flammability Control System (FCS), and the Standby Liquid Control System (SLCS). Finally, there is the Emergency Diesel Generator (EDG) System, which provides emergency AC power to operate the safety systems upon loss of offsite power.

Emergency Core Cooling Systems

High Pressure Core Flooder

The primary purpose of the HPCF System (Figure 4-3) is to maintain reactor vessel inventory after small breaks which do not depressurize the reactor vessel. HPCF systems, which are provided in two divisions, maintain an adequate coolant inventory inside the reactor vessel to limit fuel cladding temperatures in the event of breaks in the reactor coolant pressure boundary. Electrical and mechanical separation between the two divisions is assured, in addition to the physical separation, by placing each division in a different area of the Reactor Building. The HPCF systems are initiated by either high pressure in the drywell or low water level in the vessel. They operate independently of all other systems over the entire range of system operating pressures.

Both HPCF systems take their primary suction from the condensate storage pool (CSP) with the suppression pool (SP) as a secondary source. The suction source transfers automatically upon low level in the CSP or high level in the SP. The pumps are located below the condensate storage and suppression pools' normal water level to assure that net pump suction head is maintained. The HPCF System pump motors are powered by emergency diesel generators if auxiliary power is not available. The systems are also a backup to the RCIC System in response to transients.

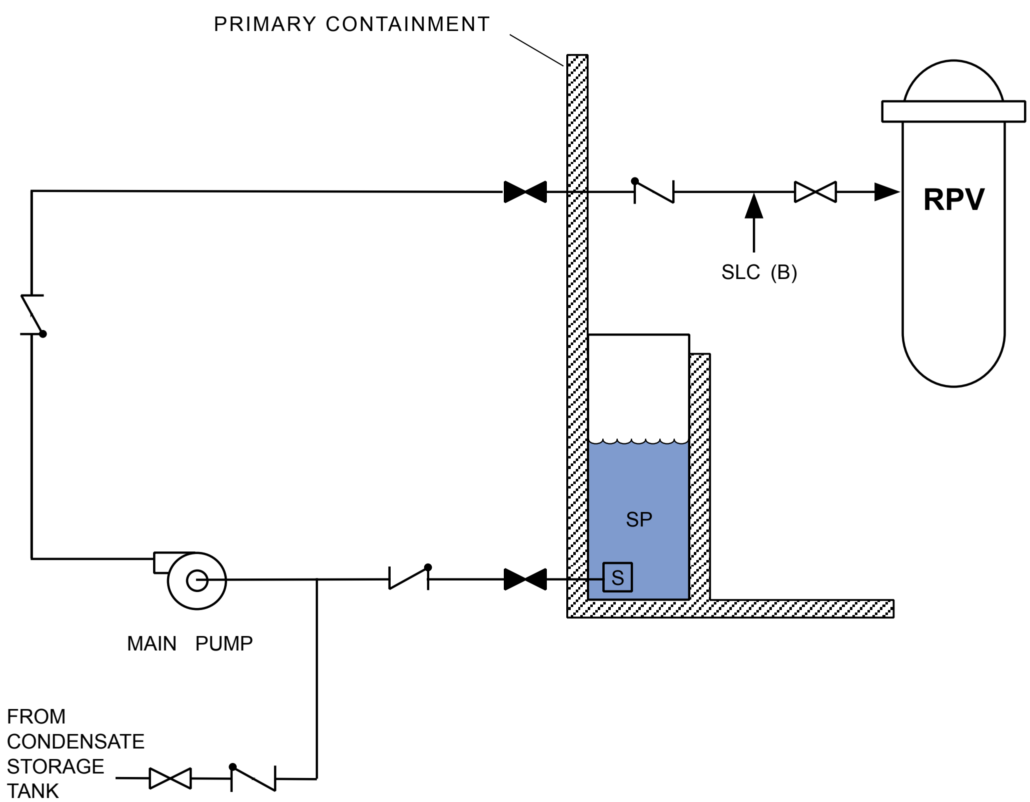


Figure 4-3. HPCF Schematic

Reactor Core Isolation Cooling

The primary purpose of the RCIC System (Figure 4-4) is to provide makeup water to the reactor vessel when the vessel is isolated. It is also part of the emergency core cooling network. The RCIC System uses a steam-driven turbine-pump unit and operates automatically in time and with sufficient coolant flow to maintain adequate water level in the reactor vessel for the following events:

- Vessel isolated and maintained at hot standby.
- Complete plant shutdown with loss of normal feedwater before the reactor is depressurized to a level where the shutdown cooling system can be placed in operation.
- Loss of AC power.

The RCIC System is contained within one division and consists of a steam-driven turbine which drives a pump assembly and the turbine and pump accessories. The system also includes piping, valves, and instrumentation necessary to implement several flow paths. The RCIC steam supply line branches off one of the main steamlines (leaving the RPV) and goes to the RCIC turbine with drainage provision to the main condenser. The turbine exhausts to the suppression pool with vacuum breaking protection. Makeup water is supplied from the condensate storage tank (CST) or the suppression pool with preferred source being the CST. RCIC flow is discharged to the feedwater injection line.

Following a reactor scram, steam generation in the reactor core continues at a reduced rate due to the core fission product decay heat. The turbine condenser and the feedwater system supply the makeup water required to maintain reactor vessel inventory. In the event the reactor vessel is isolated and the feedwater supply is unavailable, relief valves are provided to automatically maintain vessel pressure within desirable limits. The water level in the reactor vessel drops due to continued steam generation by decay heat. Upon reaching a predetermined low

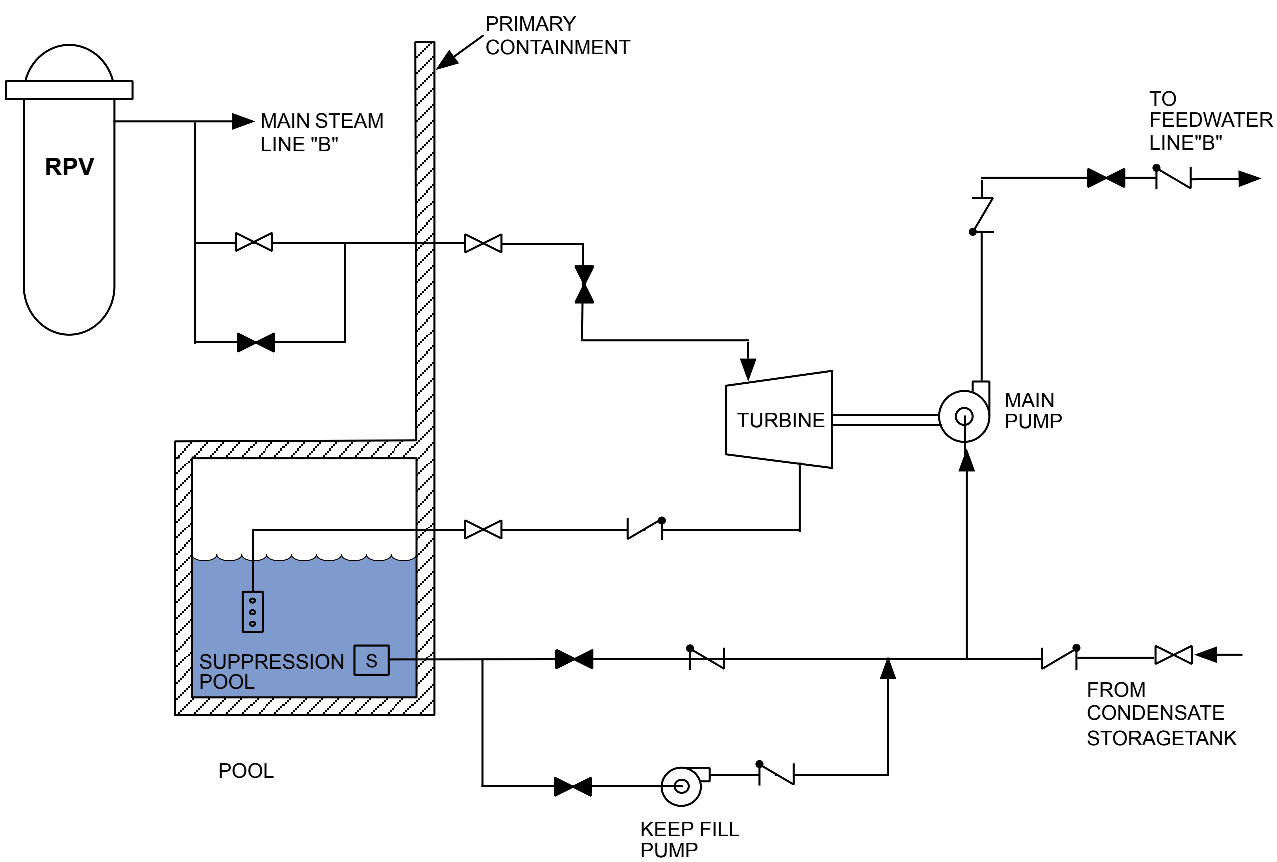


Figure 4-4. RCIC Schematic

level, the RCIC System is initiated automatically. The turbine-driven pump supplies water from the suppression pool or from the CST to the reactor vessel. The turbine is driven with a portion of the decay heat steam from the reactor vessel, and exhausts to the suppression pool.

In the event of a LOCA, the RCIC System is designed to pump water into the vessel from full operating pressure down to approximately 150 psig. During RCIC operation, the wetwell suppression pool acts as the heat sink for steam generated by reactor decay heat. This results in a rise in pool water temperature. Heat exchangers in the RHR System are used to maintain pool water temperature within acceptable limits by cooling the pool water directly.

Automatic Depressurization System

The ADS logic is automatically initiated after a short delay if an RPV low water level signal is present concurrently with a high drywell pressure signal. The ADS logic is also automatically initiated if only the RPV low water level signal is present. This initiation will occur after a longer delay to allow the high pressure ECCS a chance to restore the RPV water level to normal levels and thus avoid the ADS actuation. Both ADS initiation paths require an indication that at least one of the RHR or HPCF pumps is running before the initiation sequence is complete.

ADS initiation is accomplished by redundant trip channels arranged in two divisionally separated logics that control two separate solenoid-operated pneumatic pilots on each ADS SRV. Either pilot can operate the ADS valve. These pilots control the pneumatic pressure applied by the accumulators and the High Pressure Nitrogen Gas Supply (HPIN) System. The DC power for the logic is obtained from two separate divisions within the Safety System Logic and Control (SSLC). This arrangement makes the ADS initiation logic single-failure proof.

For ATWS mitigation, the ADS has an automatic and manual inhibit of the automatic ADS initiation to prevent ADS actuation during an ATWS. Automatic initiation of the ADS is inhibited unless there is a coincident low reactor water level signal and an average power range monitors (APRMs) downscale signal. There are also main control room switches for the manual inhibit of automatic initiation of the ADS.

The ADS can also be initiated manually. On a manual initiation signal, concurrent with positive indication of at least one of the RHR or HPCF pumps is running, the ADS function is initiated.

Residual Heat Removal

The RHR System removes residual heat during normal plant shutdown, reactor isolation and loss-of-coolant accident (LOCA). This system has six principal functions, each with a specific purpose (Figure 4-5):

Low Pressure Flooder (LPFL) Mode (3 loops)— The LPFL function provides a core cooling water supply to compensate for water loss beyond the normal control range from any cause up to and including the design basis (LOCA). During the LPFL mode, water is initially pumped from the suppression pool and diverted through the minimum flow lines until the injection valve in the discharge line is signaled to open on low reactor pressure. As the injection valve opens on low reactor pressure, flow to the RPV comes from the suppression pool, through the RHR heat exchanger, and the injection valve (blue flow path on the figure). This creates a flow signal that closes the minimum flow line. Loop flow is always through the RHR heat exchanger, which performs cooling. Safety requirements can be achieved even if any one loop is failed. Three individual loops are available. This mode is initiated automatically by a low water level in the reactor vessel or high pressure in the drywell (LOCA signal). Manual operation action may also initiate the LPFL mode. In addition, divisions B and C supply a small amount of water to the Flammability Control System (FCS) upon receipt of a LOCA signal.

Suppression Pool Water Cooling Mode (3 loops) This mode cools the suppression pool water during normal state below 49°C, which is the maximum allowable temperature in order to retain the temperature below 97°C just after reactor coolant blowdown in the worst LOCA event. It is automatically initiated on high suppression pool temperature, and can also be manually initiated. Suction is taken from the suppression pool, the flow goes through the

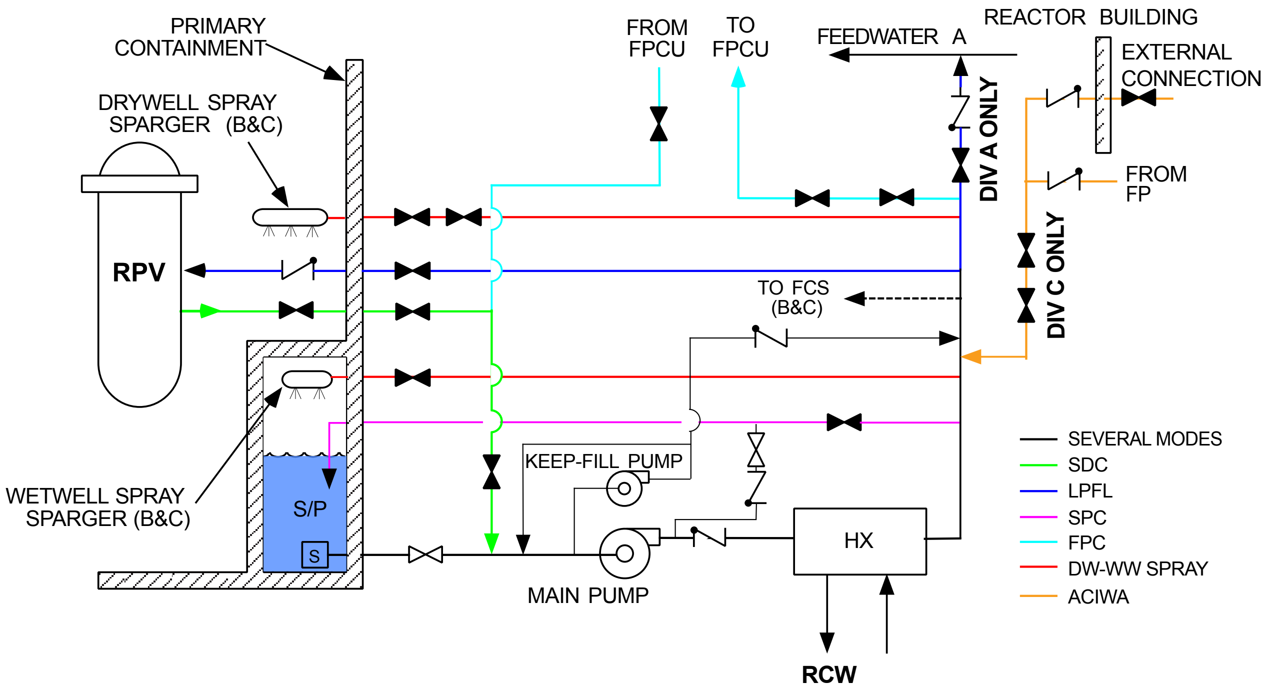


Figure 4-5. Residual Heat Removal System

RHR heat exchangers and is returned to the suppression pool (purple flow path).

Reactor Shutdown Cooling Mode (3 loops)— This mode removes decay heat and sensible heat of the RPV, piping and reactor coolant in cooperation with the turbine main condenser and feedwater system operation after reactor shutdown. It can also cool down the reactor water below 60°C within 20 hours after shutdown, and it makes refueling activity and equipment maintenance possible. After the RPV is at low pressure, suction is taken from the RPV (green flow path), the flow goes through the RHR heat exchangers and is returned to the RPV via the LPFL injection lines.

Primary Containment Vessel Spray Cooling Mode (2 loops)—This mode is manually initiated and sprays the water from the suppression chamber pool into the drywell and wetwell after the event of a LOCA. This sprayed water in the drywell returns to the suppression chamber through vent pipes after the drywell water level reaches the vent pipe inlet level. It is mixed with the sprayed water in the wetwell and cooled by the RHR System heat exchangers, and then it is sprayed again (red flow path).

This mode is available in divisions B and C. Each system can remove released coolant energy during an assumed feedwater line break, decay heat and generated heat by fuel cladding-H_2O reaction accompanied by overheated fuel in conjunction with LPFL mode. It can also prevent the primary containment from exceeding its maximum operating pressure and temperature. About 88% of this system flow rate is sprayed in the drywell and the remaining 12% is sprayed in the wetwell. It can also remove released iodine in the gas phase existing in the primary containment. The heat exchanger is cooled by the Reactor Building Closed Cooling System (RCW).

Supplemental Fuel Pool Cooling (3 loops)— The three RHR loops are capable of providing supplemental fuel pool cooling. This mode will be used only when the reactor is shut down and the Fuel Pool Cooling System (FPCU) is unable to maintain the fuel pool water temperature below the required design limits. This is a manually initiated operation. Refer to the cyan flow path.

AC-Independent Water Addition (1 loop)— The ACIWA mode of RHR Loop C provides a means for introducing water from Fire Protection (FP) through RHR Loop C piping and valves directly into either the RPV, drywell spray header, or wetwell spray header. The purpose is to prevent core damage or, if core damage has already occurred, to terminate melt progression when AC power is not available from either onsite or offsite sources. The ACIWA mode of RHR provides manual capability to prevent core damage when all ECCS are lost. Refer to the orange flow path.

Standby Gas Treatment System

The Standby Gas Treatment System (SGTS) consists of two 100% capacity divisions (B&C) which treat and then discharge either primary or secondary containment air to the plant stack (Figure 4-6).

SGTS automatically starts, takes suction from the secondary containment following a LOCA, and maintains a negative pressure of approximately 6 mm water in the Secondary Containment. In addition, SGTS will automatically process secondary containment atmosphere during refueling operations or primary containment atmosphere during purging operations if a high radiation signal is received. The system can also be manually initiated.

Each filter train consists of a moisture separator, main electric heater, prefilter, primary HEPA filter, charcoal adsorber and secondary HEPA filter. The charcoal adsorber bed is 15 cm thick and removes more than 99% of elemental iodine or methyl iodide.

Atmospheric Control System

The Atmospheric Control System (ACS) is de-

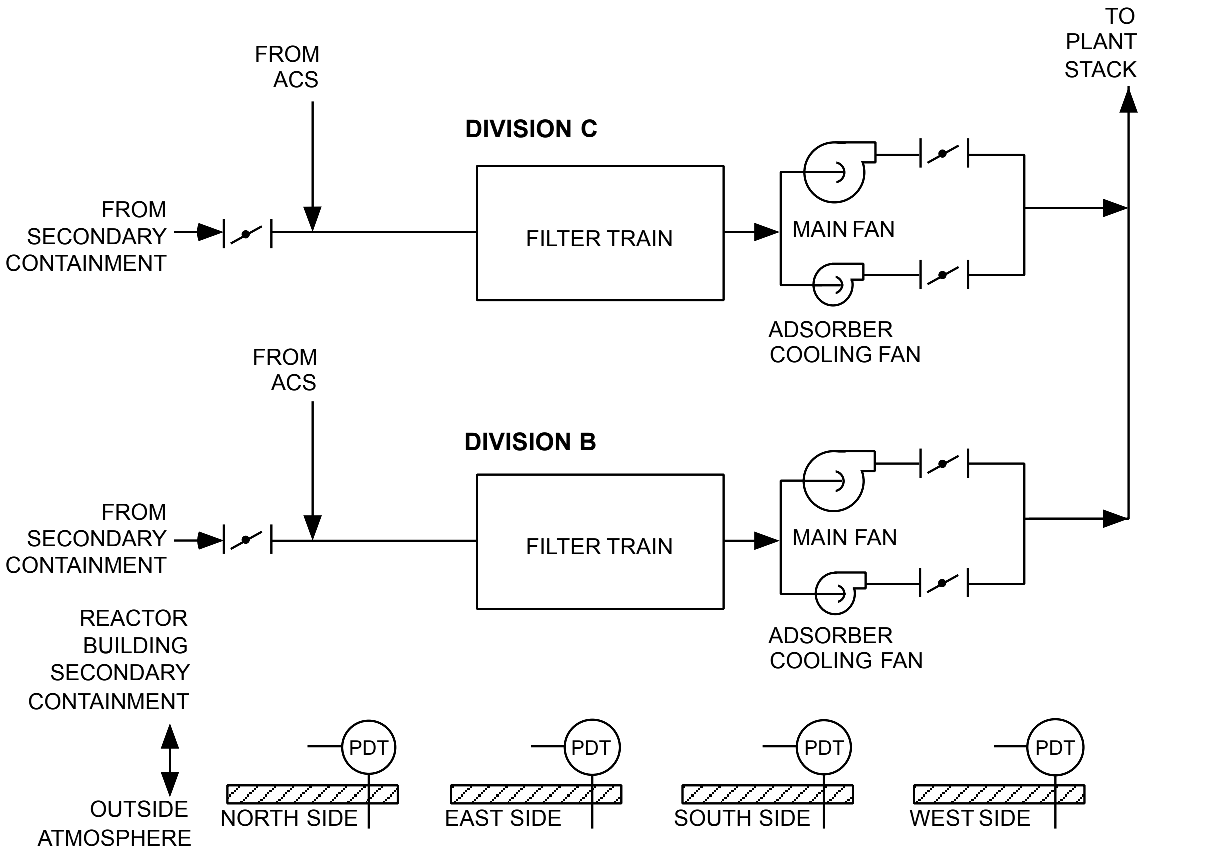


Figure 4-6. Standby Gas Treatment System

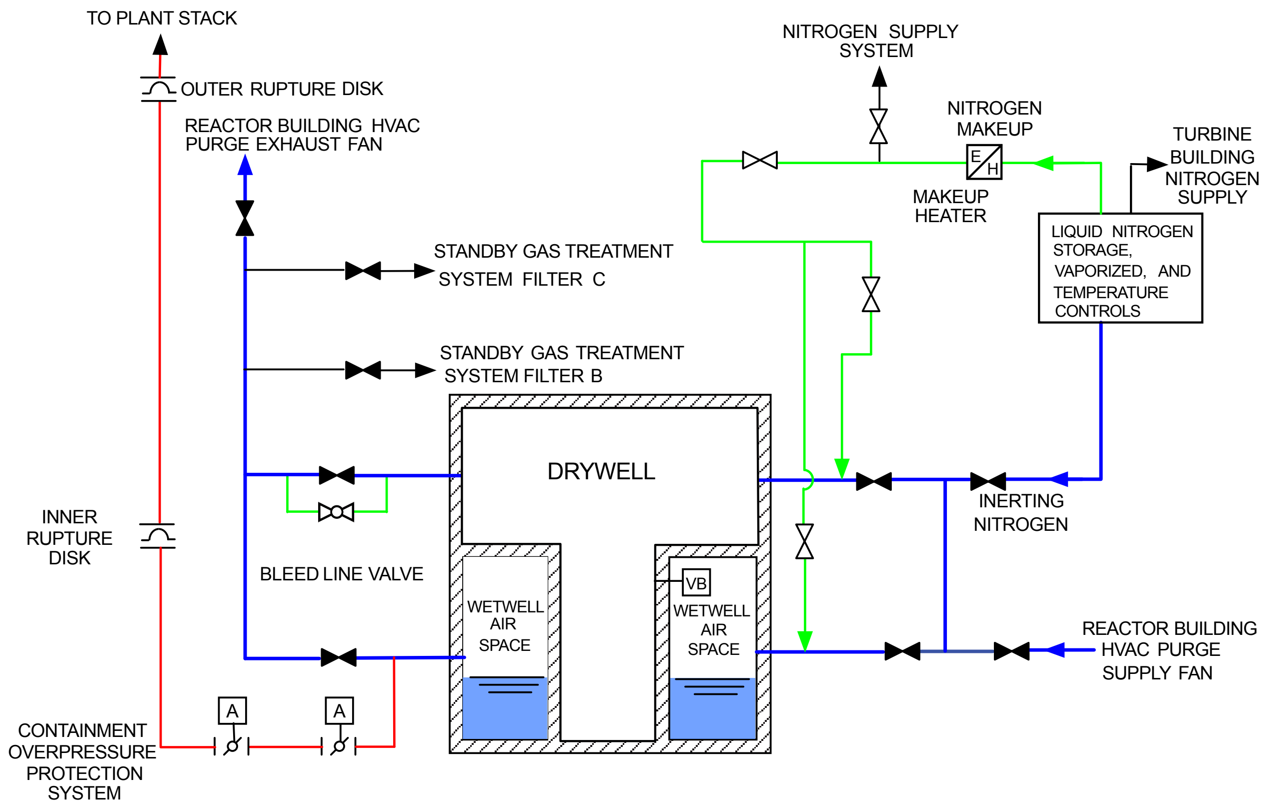


Figure 4-7. Atmospheric Control System

signed to establish and maintain an inert atmosphere (nitrogen) within the primary containment volume (PCV) (see Figure 4-7). An inert atmosphere is maintained in all operating modes except plant shutdown for refueling and/or maintenance. The system is sized to reduce containment oxygen concentrations from atmospheric to <3.5% by volume in less than 4 hours.

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 (DWC) System fans. Once inerting is complete, ACS 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 secondary containment.

The ACS also has a Containment Overpressure Protection System (COPS) which is designed to relieve containment pressure against rare severe accident sequences in which the structural integrity of the containment is challenged by overpressure. If the wetwell pressure reaches the setpoint of the inner rupture disk, the rupture disk opens and containment gases from the wetwell airspace, which have been scrubbed by the suppression pool, are vented to atmosphere through the plant stack. Once the containment pressure has been reduced to a safe level and normal containment heat removal has been regained, the two normally open air-operated isolation valves in the COPS relief path can be manually closed to reestablish the containment boundary. The pressure setpoint is established to assure the containment pressure does not exceed the Service Level C capability of the containment.

Flammability Control System

The Flammability Control System (FCS) controls the potential buildup of a combustible mixture of hydrogen and oxygen inside the PCV which could be produced from a design basis metal-water reaction and radiolysis of water during LOCA, or from beyond design basis events. It is comprised of two redundant thermal hydrogen and oxygen recombiner units located in two separate divisions (see Figure 4-8). Each unit is designed to maintain the concentration of oxygen below the flammability limit by recombining hydrogen and oxygen without relying on purging or release of radioactive material to the environment. These units are skid mounted with blowers, electric heaters, water spray coolers, piping, valves and instrumentation. The system is initiated manually by monitoring hydrogen and oxygen levels in the PCV.

Standby Liquid Control System

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. The system is sized to counteract the positive reactivity effect of shutting down from rated power to cold shutdown condition.

The SLCS is automatically initiated or it can be manually initiated from the main control room to pump the neutron absorbing solution into the reactor.

The system includes a boron solution tank, a test water tank, two positive displacement pumps, motor-operated injection and pump suction valves, and associated local piping, valves and controls in the Reactor Building outside the primary containment. The liquid is piped into the reactor vessel through the HPCF line downstream of the HPCF inboard check valve inside the primary containment. Figure 4-10 illustrates the system configuration.

The boron absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the fuel. The specified neutron absorber solution is sodium pentaborate. At all times, when it

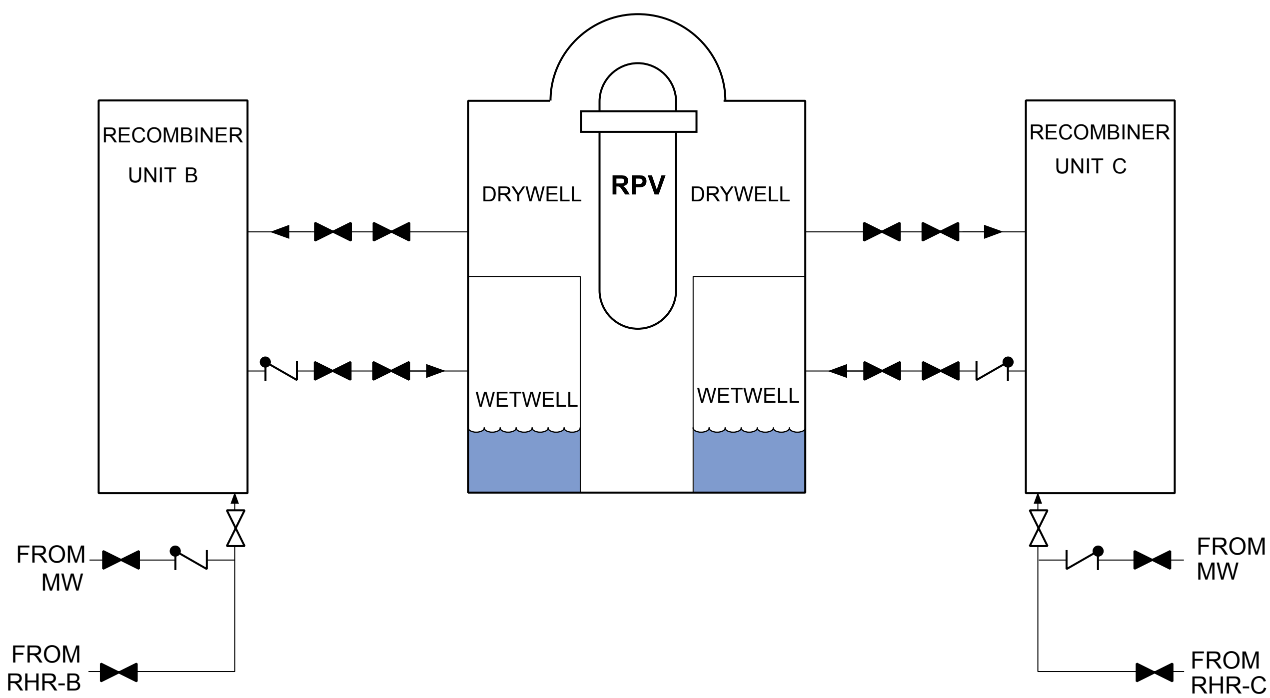


Figure 4-8. Flammability Control System

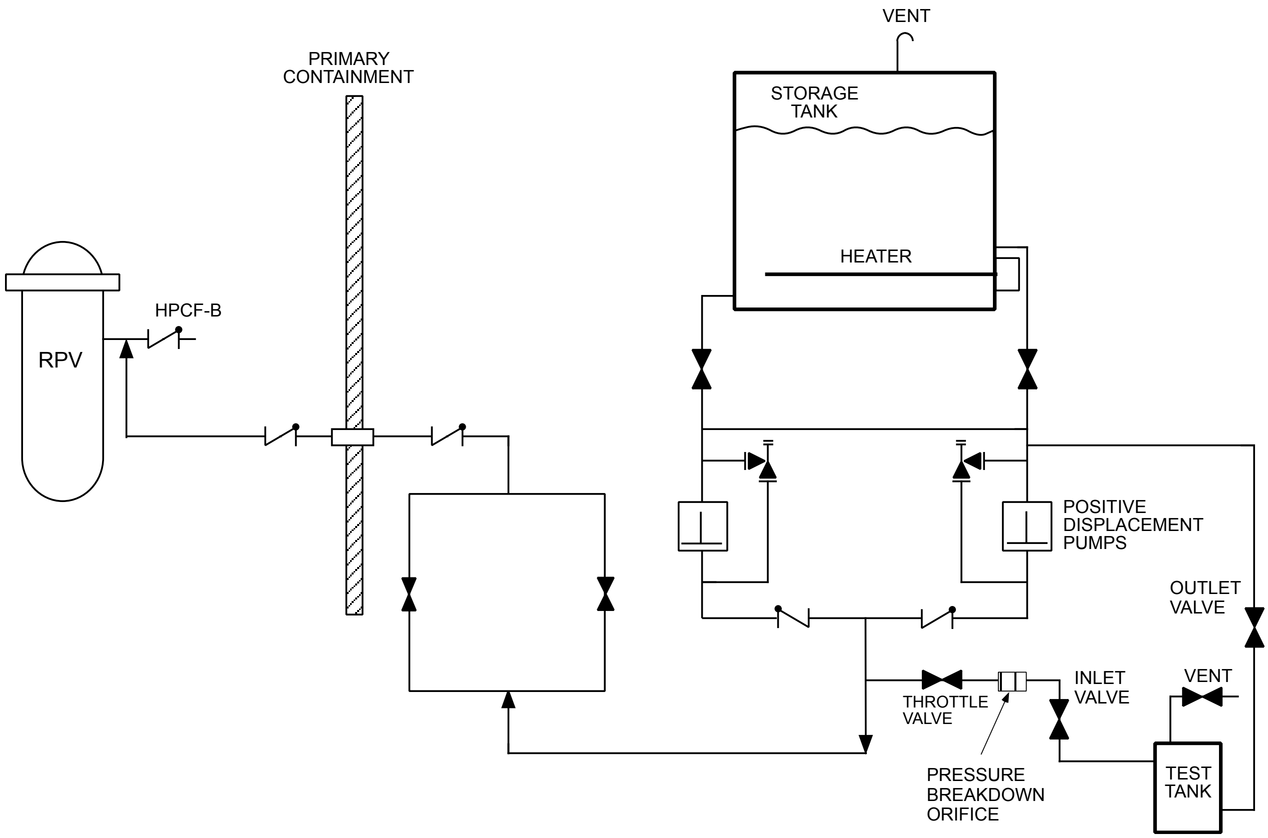


Figure 4-9. Standby Liquid Control System

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.

Emergency Diesel Generator System

The Emergency Diesel Generator (EDG) System consists of three diesel engines and their respective combustion air intake system, starting air system, fuel oil system (from the day tank to the engine), lubricating oil system, engine jacket cooling water system, engine exhaust system and silencer, governor system, and generator with its excitation and voltage regulation systems.

The three DGs are classified as Class 1E, safety-related and supply standby AC power to their respective Class 1E Electrical Power Distribution (EPD) System divisions (Divisions I, II, and III). The DG connections to the EPD System are shown on Figure 9-2.

The DGs are sized to supply their load demand following a LOCA. The DG air start receiver tanks are sized to provide five DG starts without recharging their tanks.

A LOPP signal (bus undervoltage) from an EPD System medium voltage divisional bus automatically starts its respective DG, and initiates automatic load shedding and connection of the DG to its divisional bus. A DG automatically connects to its respective bus when DG required voltage and frequency conditions are established and required motor loads are tripped. After a DG connects to its respective bus, the non-accident loads are automatically sequenced onto the bus.

LOCA signals from the RHR (Division I) and HPCF (Divisions II and III) Systems automatically start their respective divisional DG. After starting, the DGs remain in a standby mode (i.e., running at required voltage and frequency, but not connected to their buses), unless a LOPP signal exists. When LOCA and LOPP signals exist, load shedding occurs and required motor loads are tripped, and the DG automatically connects to its respective divisional bus. After a DG connects to its respective bus, the LOCA loads are automatically sequenced onto the bus.

A manual start signal from the main control room (MCR) or from the local control station in the DG area starts a DG. After starting, the DG remains in a standby mode, unless a LOPP signal exists.

DGs start, attain required voltage and frequency, and are ready to load in <20 seconds after receiving an automatic or manual start signal.

When a DG is operating in parallel (test mode) with offsite power, a loss of the offsite power source used for testing or a LOCA signal overrides the test mode by disconnecting the DG from its respective divisional bus.

The DG units are located in their respective divisional areas in the Reactor Building. The DG combustion air intakes are located above the maximum flood level. The DG combustion air intakes are separated from DG exhaust ducts. Class 1E DG unit auxiliary systems are supplied electrical power from the same Class 1E division as the DG unit. Independence is provided between Class 1E divisions and also between Class 1E divisions and non-Class 1E equipment. Each divisional DG with its auxiliary systems, is physically separated from the other divisions.

Chapter 5
Auxiliary Systems

Reactor Auxiliary Systems

The main auxiliary systems in the ABWR Nuclear Island consist of the Reactor Water Cleanup (RWCU) System, the Fuel Pool Cooling and Cleanup (FPCU) System, the Suppression Pool Cleanup (SPCU) System, the Reactor Building Cooling Water (RCW) System, the Reactor Building Service Water (RSW) System and the Drywell Cooling System (DWC). In addition, 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.

A short description of radioactive waste management systems is also provided in this chapter.

Reactor Water Cleanup System

The RWCU System (Figures 5-1 and 5-2) consists of piping, valves, pumps, heat exchangers and filter-demineralizers which are used to remove impurities from the reactor primary coolant water to maintain water quality within acceptable limits during the various plant operating modes. The system is comprised of 2 100% pumps and 2 100% filter demineralizers, only one of which is used during normal operation. Reactor water temperature is lowered to about 50°C through the tube side of the regenerative heat exchanger (RHX) and non-regenerative heat exchangers (NRHX), and then cleaned by a filter demineralizer before being reheated by the RHX prior to return to the RPV.

The pumps are canned rotor type that do not require seal maintenance, and the motors are cooled by continuous purge flow from the CRD System. Each pump has the capacity of 2% of rated feedwater flow.

The RWCU also assists during reactor heatup by removing excess water from the primary system to either low level radwaste or to the condenser hotwell, and assists during reactor cooldown by providing water for the head spray nozzle on the RPV to speed up the vessel cooling.

In plant emergency situations, the filter demineralizers and the regenerative heat exchanger can be bypassed, and the RWCU can provide an alternate decay heat removal path.

In addition to MSS and Feedwater, RWCU is the only other normally operating process system with primary system high pressure water located outside of primary 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 value located inside containment which can be used to effect isolation.

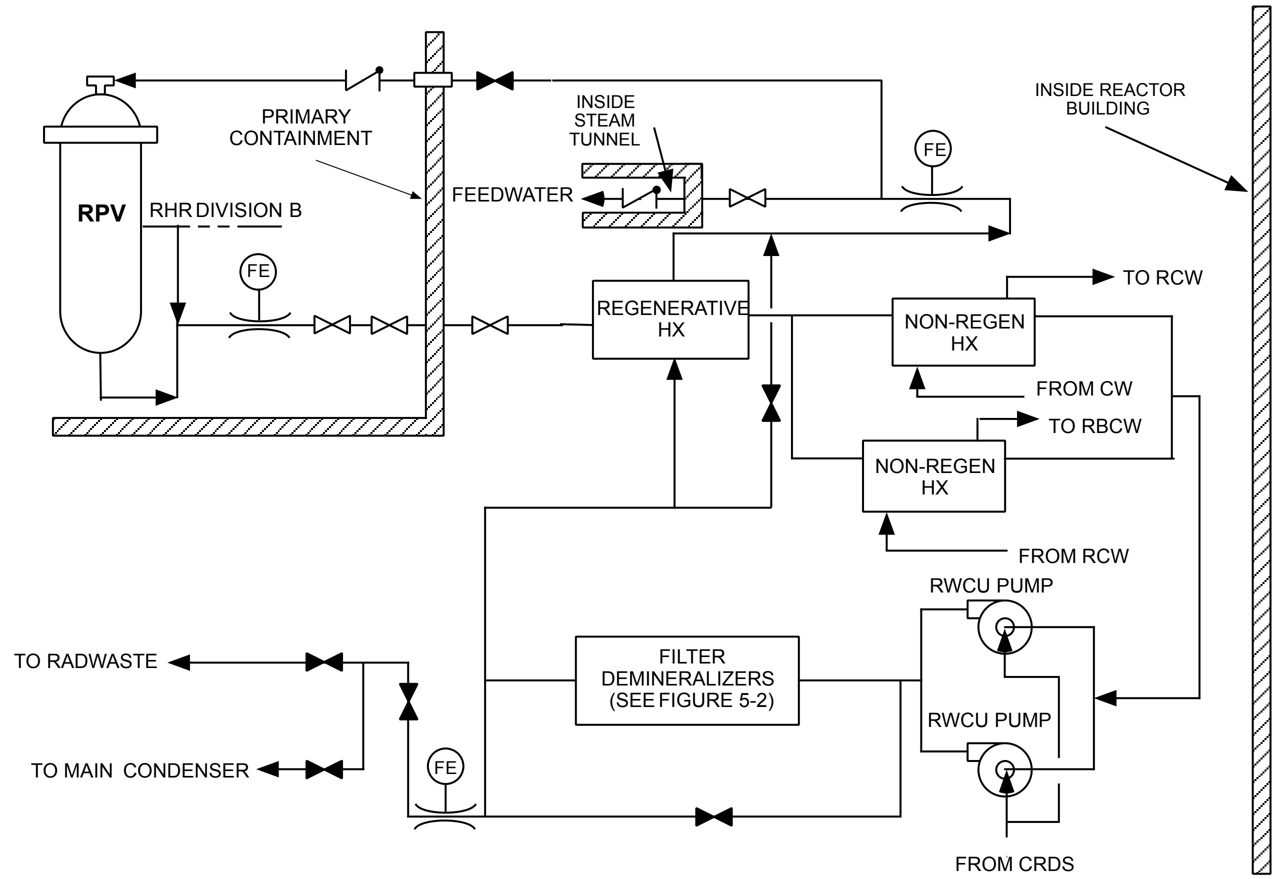


Figure 5-1. Reactor Water Cleanup System

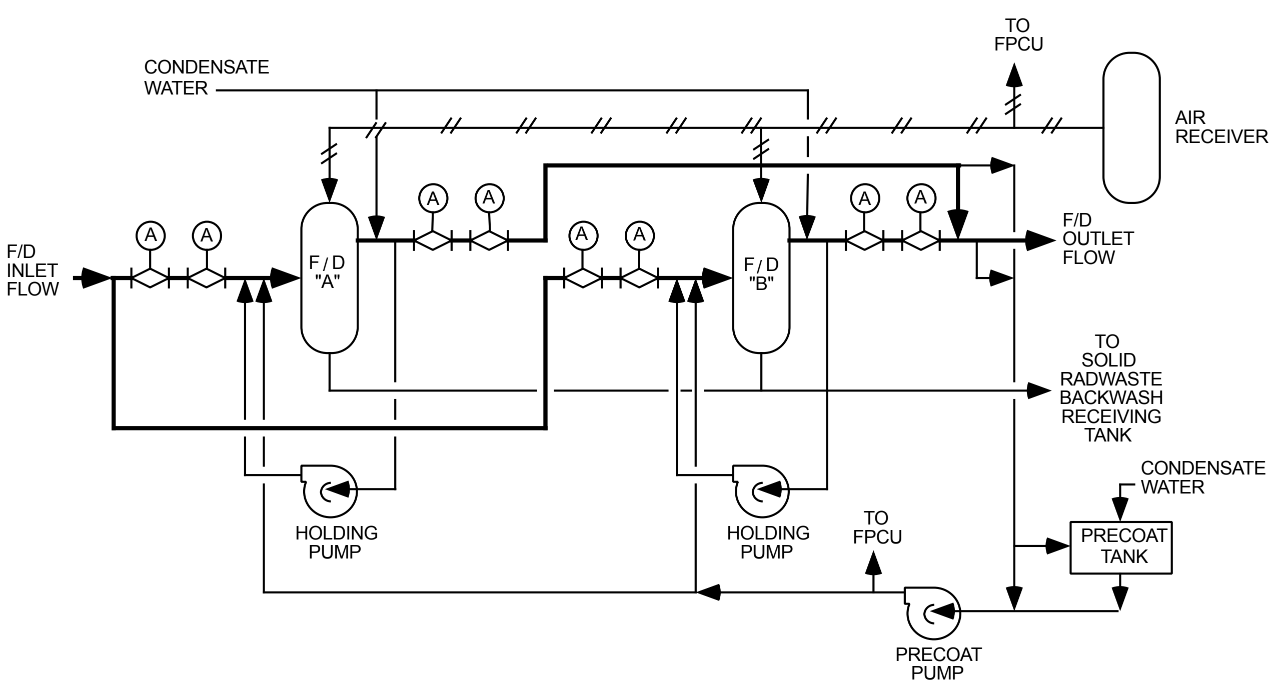


Figure 5-2. RWCU Filter Demineralizers

Fuel Pool Cooling and Cleanup and Suppression Pool Cleanup System

The FPCU and SPCU Systems (Figures 5-3 and 5-4) consist of piping, valves, heat exchangers and filter-demineralizers which are used to remove decay heat from the spent fuel storage pool and remove impurities from the water in the spent fuel pool and dryer/separator pool and suppression pool to maintain water quality within acceptable limits during various plant operating modes. The filter-demineralizers in the FPCU System are shared by the SPCU System for cleaning the suppression pool water. The filter demineralizers are similar in design to that of the RWCU. The two systems are used during refueling outages as follows:

- Prior to refueling, the SPCU System is used to cleanup and transfer water from the suppression pool and Condensate Storage Tank (CST) to fill the reactor well and dryer-separator pool. It is also used to drain the water from this area back to the suppression pool before reactor startup.
- During the fuel movement portion of the refueling outage, the FPCU System provides cooling to the spent fuel pool, reactor well and dryer-separator pool. RHR provides supplemental cooling in the event of a maximum heat load condition (e.g., full core off-load).

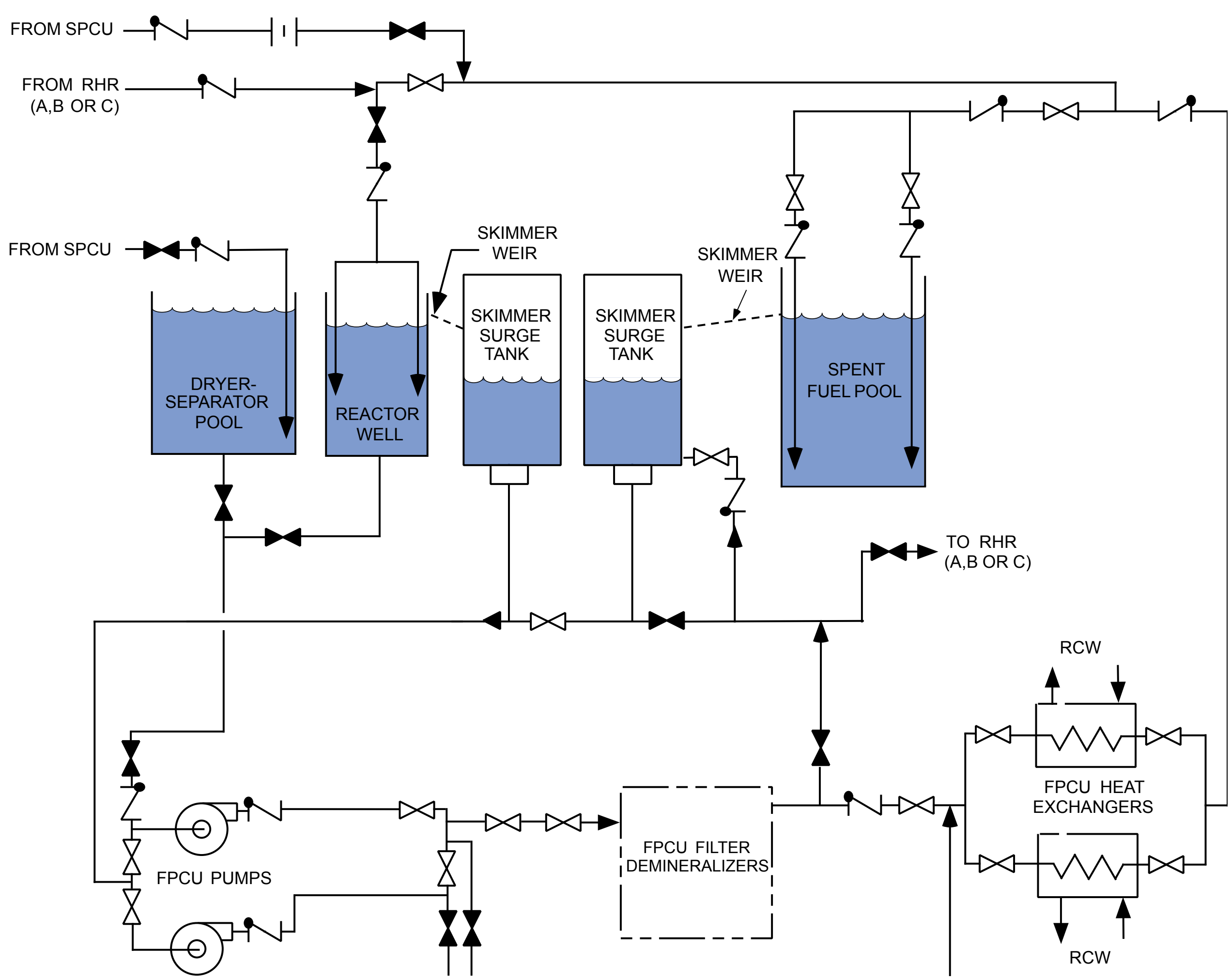


Figure 5-3. Fuel Pool Cooling and Cleanup System

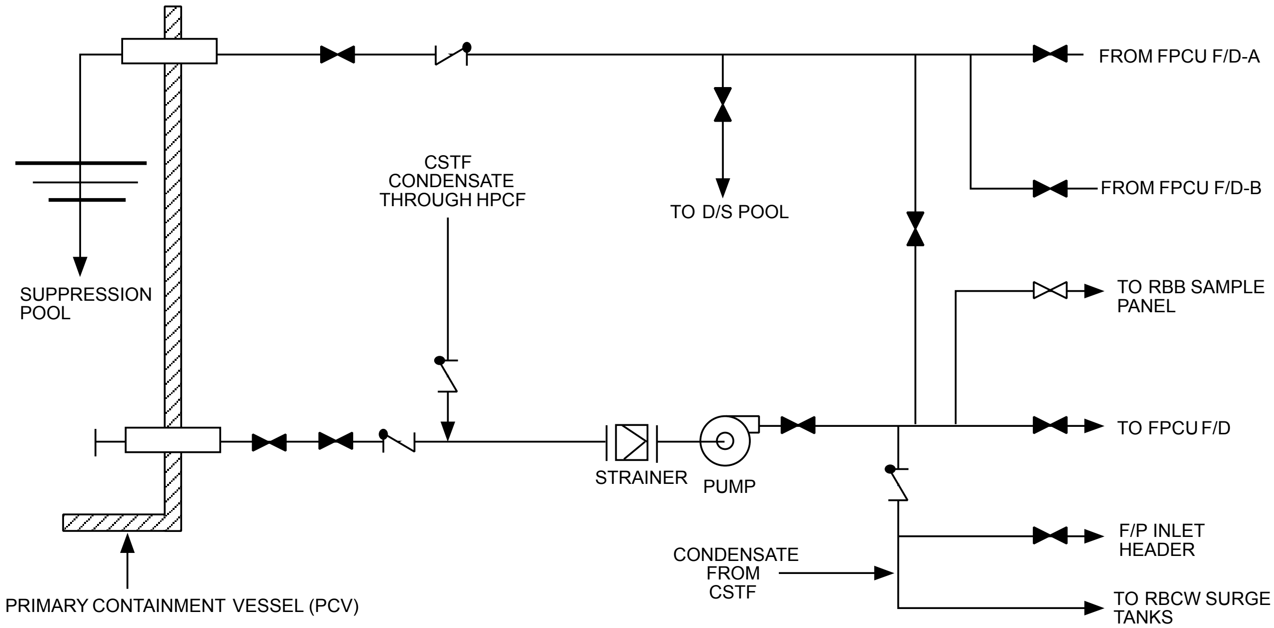


Figure 5-4. Supression Pool Cleanup System

- After refueling, the FPCU System is used to cleanup and transfer water from the reactor well and dryer-separator pool back to the suppression pool and drain the remaining water to radwaste.

Both systems can be operated at the same time, each using one of the two filter demineralizers. The FPCU also can receive cooling support from the RHR if necessary to keep the fuel pool temperature within limits.

Reactor Building Cooling Water System/ Reactor Building Service Water System

The Reactor Building Cooling Water (RCW) System (Figure 5-5) consists of piping, valves, pumps and heat exchangers which are used to provide cooling water to various systems in the Nuclear Island. The system is divided into three separate safety divisions, each with its own pumps and heat exchangers, to provide cooling water to equipment in the three ECCS and RHR Safety divisions. This includes cooling for EDGs and for the Emergency Chilled Water (ECW) system which supports HVAC in key areas after postulated accidents. The RCW System also provides cooling water to equipment in non-safety systems such as the RWCU, FPCU and other systems and equipment that require cooling water. Non-safety heat loads are distributed between the three RCW systems. Figure 5-5 shows a typical configuration for RCW-A. Normally only one of the two 50% RCW pumps is operating. If a LOCA occurs, the non-safety heat loads are isolated and the second RCW pump started in each division so that the RCW can automatically respond to the accident heat removal requirements.

The RSW System contains 2 100% pumps per division. To achieve low approach temperatures, provide easier maintenance and better performance, the ABWR uses flat plate heat exchangers; the pumps and heat exchangers are located in the Control Building. The RCW heat exchangers are cooled by water from the Reactor Building Service Water (RSW) System and ultimate heat sink, which can be a cooling pond, cooling tower or natural body of water, depending on unique site conditions.

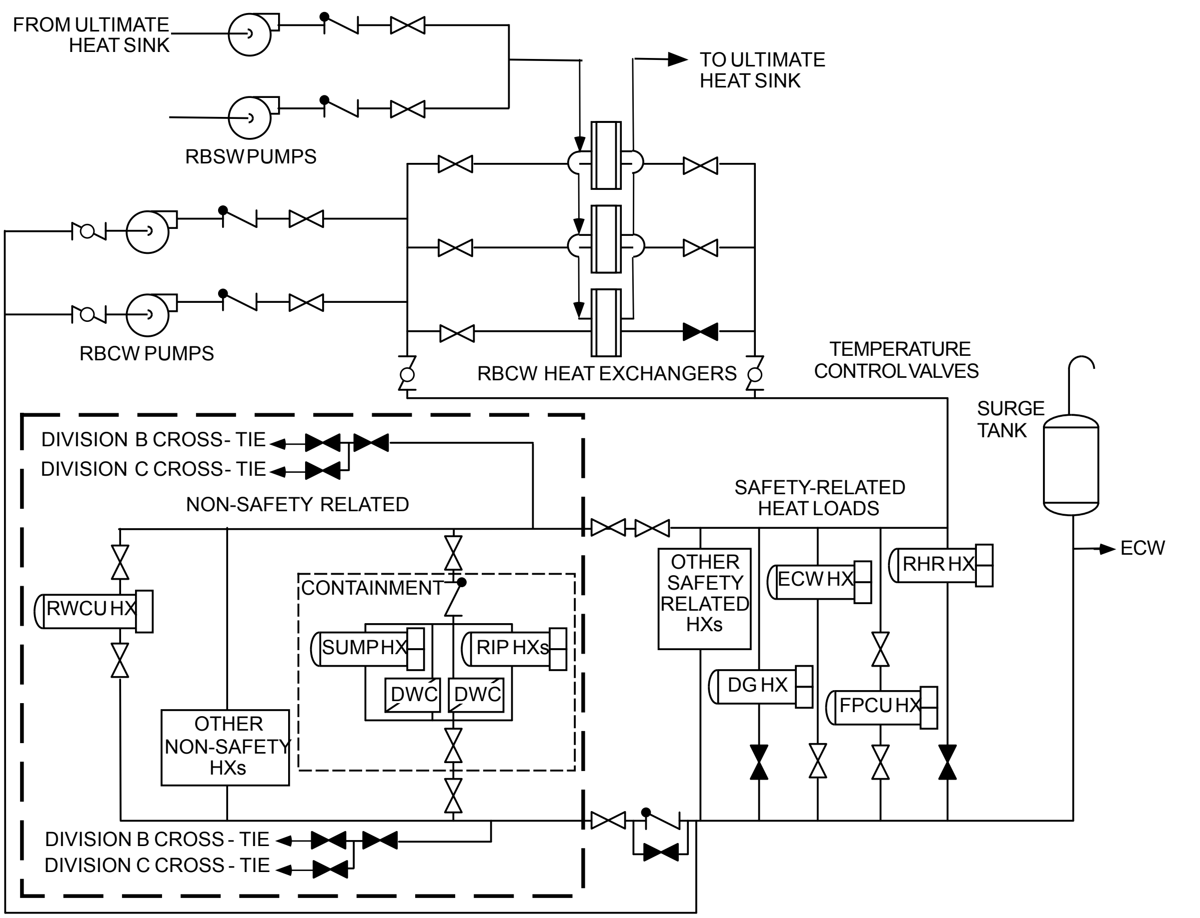


Figure 5-5. Reactor Building Cooling Water System and Reactor Building Service Water System

Drywell Cooling System

The purpose of the Drywell Cooling System (DWC) is to provide conditioned air/nitrogen to the drywell head area, upper and lower drywell, and shield wall annulus during normal plant operation, refueling outage and normal operation transients to cool equipment and maintain the drywell temperature within limits to ensure the integrity of the concrete structure.

The system is comprised of three fans, three first-stage cooling coils supplied with RCW cooling water, two second-stage cooling coils supplied with chilled water, a distribution header and duct work (see Figure 5-6). Hot air is drawn from the upper drywell space over the first-stage coils, cooled and discharged to the common distribution header. At the header, part of the cooled air is ducted directly to the lower area of the drywell, control rod drive area, shield wall annulus and reactor vessel support skirt with the remainder passes through the second-stage cooling coils and is ducted to the upper drywell and drywell head area.

Radwaste

The radwaste facility has been improved compared to past designs. The use of filters and deep bed demineralizers without resin regeneration for condensate treatment permits a reduction in liquid radwaste effluent. The use of a total organic carbon oxidizer, polishing demineralizers in a lead-lag arrangement, a roughing demineralizer and mobile process equipment have improved processed water quality for plant reuse or offsite discharge. Han-

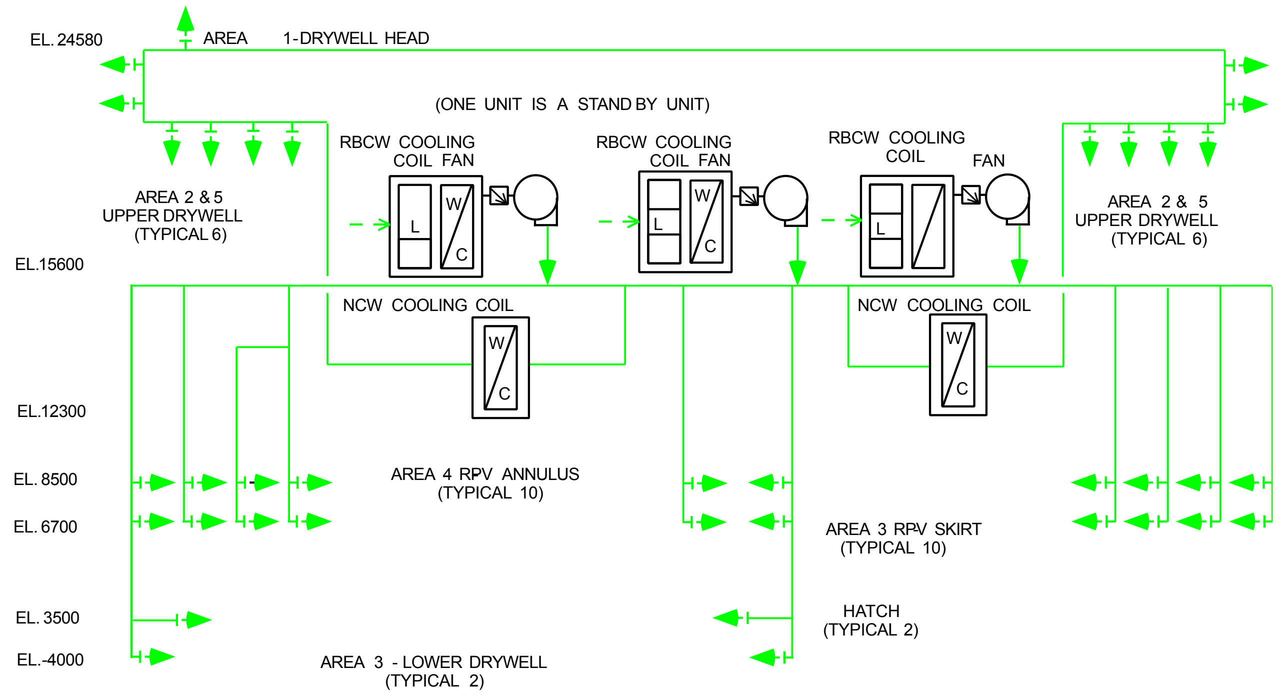


Figure 5-6. Drywell Cooling System

dling of dry solid waste is either by compaction or offsite volume reduction processing facilities. Spent resin, filter media and other types of wet waste material are dewatered and stored in high integrity containers. The impact of these improvements in the ABWR designs gives assurance that the total radwaste volume for the plant will be less than 21 cubic meters per year (or 100 drums/year), reducing the radwaste volume by a factor of three compared to current U.S. operating plants. Annual releases to the environment from the ABWR radwaste systems are "as low as reasonably achievable" in accordance with guidelines set forth in 10CFR50, Appendix I. These levels are several orders of magnitude below the NRC established limits in 10CFR20.

Radwaste systems include the Liquid Radwaste Management (LRM) System, the Offgas (OG) System, and the Solid Radwaste Management (DRM) System. Radwaste requirements and designs may vary depending on specific utility requirements; for example, whether supercompaction and/or incineration is permitted or not. What follows is a typical design description.

Liquid Radwaste Management System

The Liquid Radwaste Management System (LRW) collects, monitors, and treats liquid radioactive wastes for return to the primary system whenever practicable. The LRW System is designed to segregate, collect, store, and process potentially radioactive liquids generated during various modes of typical plant operation: Startup, Normal, Hot Standby, Shutdown, and Refueling. The LRW equipment is selected, arranged, and shielded to permit operation, inspection, and maintenance with minimum radiation exposure to personnel. Additionally, the system is designed such that it may be operated to maximize the recycling of water within the plant, which would minimize the releases of liquid to the environment. All potentially radioactive liquid wastes are collected in sumps or drain tanks at various locations in the plant and transferred to collection tanks in the radwaste facility.

The LRW block flow diagram is shown in Figure 5-7. Waste processing is done on a batch basis. Each batch is sampled as necessary in the collection tanks to determine concentrations of radioactivity and other contaminants. The LRW is composed of four subsystems which are designed to collect, treat, and recycle or discharge the different categories of waste water. The four subsystems are the Low Conductivity Subsystem, High Conductivity Sub-

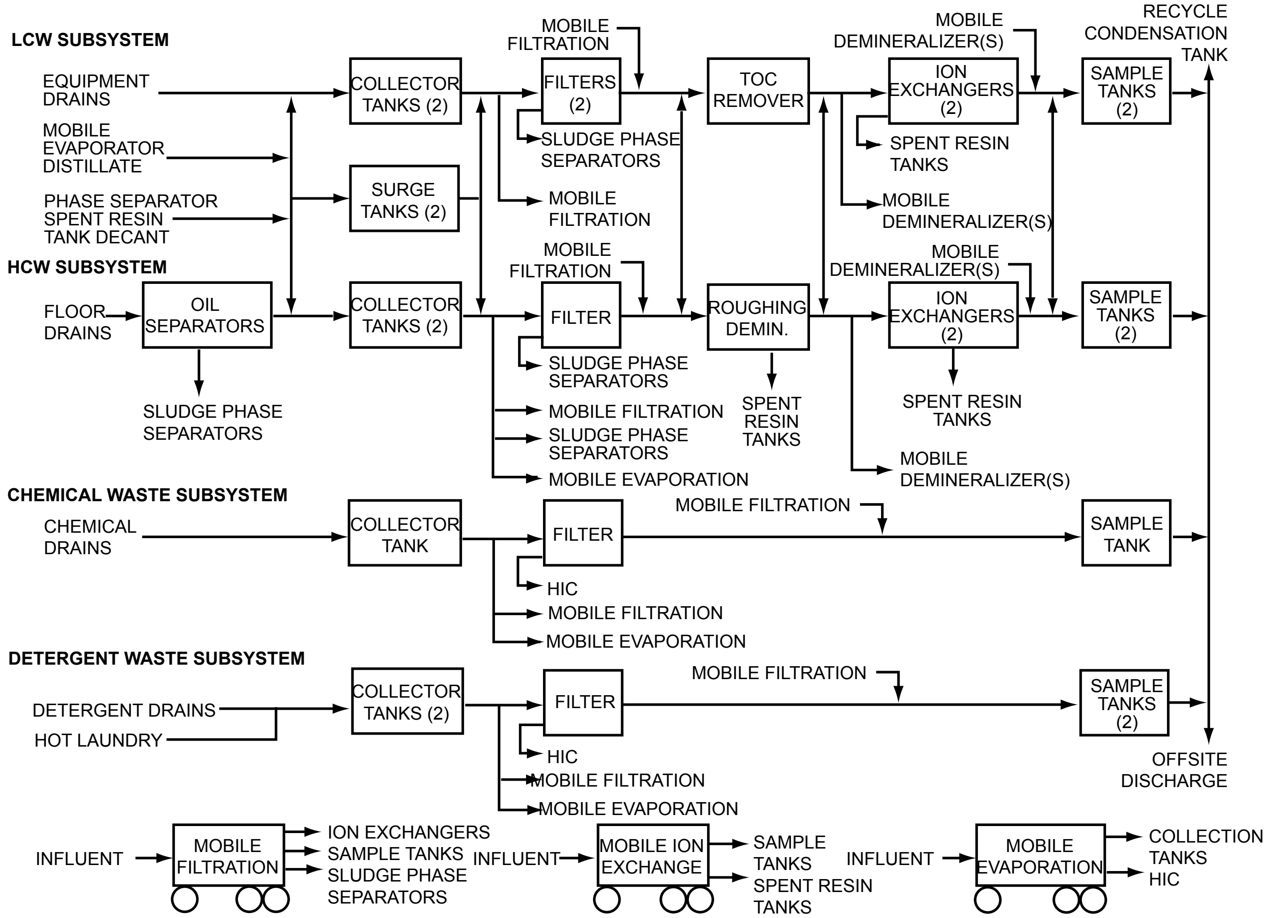


Figure 5-7. Liquid Radwaste System

system, Chemical Waste Subsystem and Detergent Waste Subsystem.

Low Conductivity Subsystem: This subsystem collects and processes water of relatively low conductivity. Equipment drains and other low-conductivity wastes (LCW) are collected, filtered for removal of insolubles, processed for removal of total organic carbon (TOC) contaminants, demineralized on a mixed resin, deep-bed demineralizers for removal of solubles, processed through a second polishing demineralizer, and then routed to condensate storage unless high conductivity or TOC requires recycling for further treatment. A second LCW filter, arranged in parallel with the first, is also provided.

High Conductivity Subsystem: This subsystem collects and processes dirty radwaste [i.e., water of relatively high conductivity and solids content (HCW)]. Floor drains are typical of wastes found in this subsystem. These wastes are collected, filtered for removal of insolubles, demineralized on a roughing demineralizer, processed through two polishing demineralizers, and then released via the offsite discharge pathway unless low condensate storage tank inventory or low dilution water flow requires plant reuse, or high radioactivity requires recycling for further treatment. Protection against inadvertent release of liquid radioactive waste is provided by design redundancy, instrumentation for the detection and alarm of abnormal conditions, automatic isolation, and administrative controls.

Chemical Waste Subsystem: This subsystem collects and processes chemical wastes from the radioactive laboratory and chemical decontamination operations. Normally, chemical drain wastes of low activity are treated by filtration, sampled and released via the liquid discharge pathway on a batch basis. However, mobile evaporation can be used as a process option for chemical drains.

Detergent Waste Subsystem: This subsystem collects and processes detergent wastes from personnel showers and laundry operations. Normally, laundry drain wastes and other detergent wastes of low activity are treated by filtration, sampled and released via the liquid discharge pathway on a batch basis. However, mobile filtration and/or evaporation can be used as a process option for detergent drains. During periods of high laundry use, such as during outages, excess laundry above the capacity of the plant laundry will be sent offsite for processing by a licensed vendor.

Offgas System

The Gaseous Waste Management or Offgas System (OG) 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 as low as reasonably achievable. The OG process flow diagram is shown in Figure 5-8.

The main condenser evacuation system removes the noncondensable gases from the main condenser and discharges them to the OG. The evacuation system consists of two 100% capacity, multiple-element, two-stage steam jet air ejectors (SJAEs) with intercondensers, for normal station operation, and mechanical vacuum pumps for use during startup.

The OG System receives air and noncondensable gases from the SJAEs and processes the effluent for the decay and/or removal of gaseous and particulate radioactive isotopes. The OG System also reduces the possibility of an explosion from

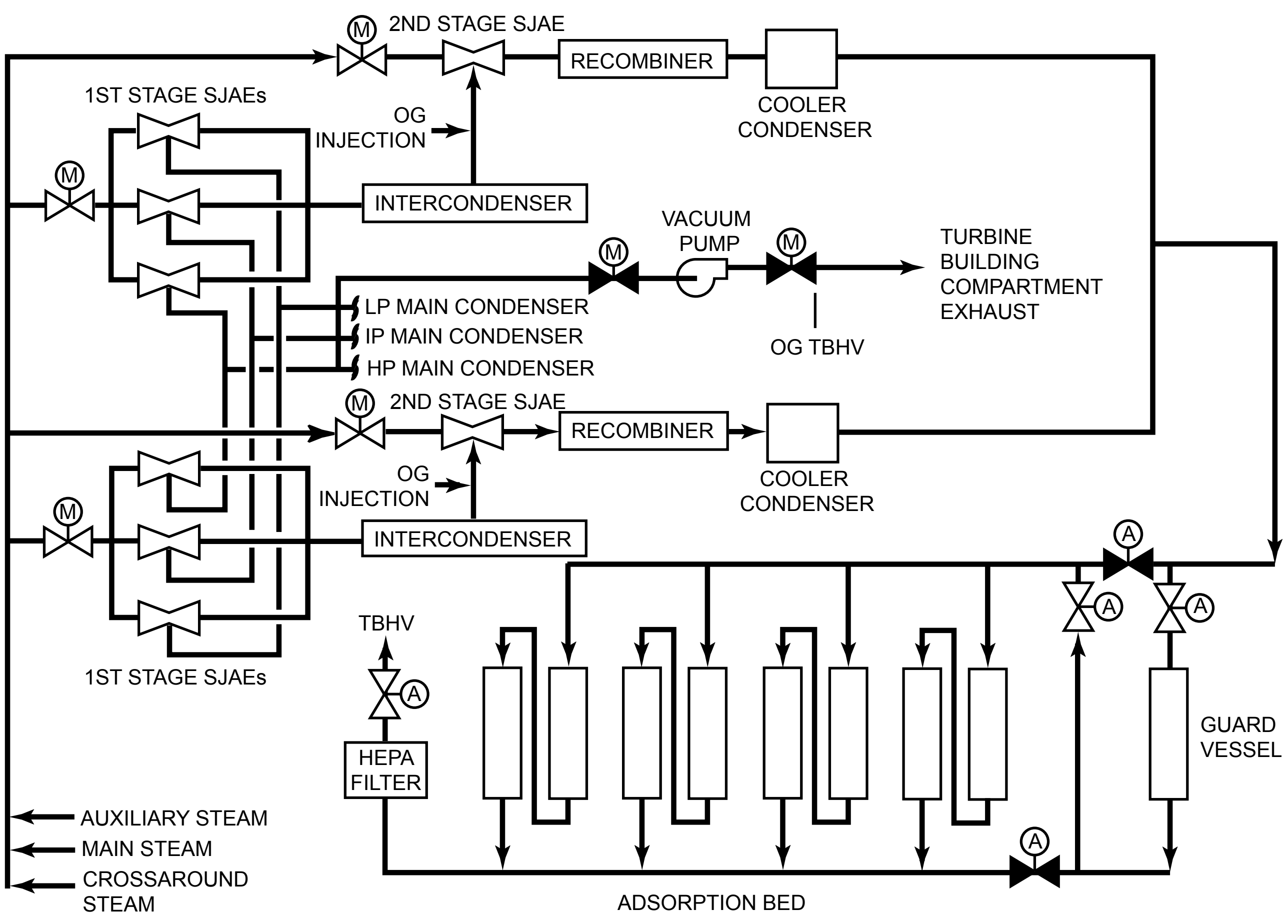


Figure 5-8. Offgas System

the buildup of radiolytic hydrogen. 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.

The remaining noncondensables (principally air with traces of krypton and xenon) are passed through activated charcoal beds which are operated at an ambient temperature (25°C) and provide a holdup volume to allow time for the krypton and xenon to decay. After processing, the gaseous effluent is monitored and released to the environs through the plant stack.

Solid Radwaste Management System

The Solid Radwaste Management System, also known as the Dry Radwaste Management (DRM) System, provides for the safe processing, handling, packaging, and short-term storage of radioactive wet and dry solid waste. The DRM is designed to provide dewatering/compaction and packaging for radioactive wastes produced during startup, normal operation, and shutdown and to store these wastes, as required, in the Radwaste Building. The DRM block flow diagram is shown in Figure 5-9.

Wet waste produced by this system is transferred to fill stations where it is loaded into high integrity containers (HICs) and dewatered. Dry active waste is surveyed and disposed of whenever possible via the provisions of 10CFR20.302(a). The remaining combustible and noncombustible waste is compacted and loaded into containers for offsite disposal or loaded into sea vans for offsite processing, which includes incineration and supercompaction, and disposal.

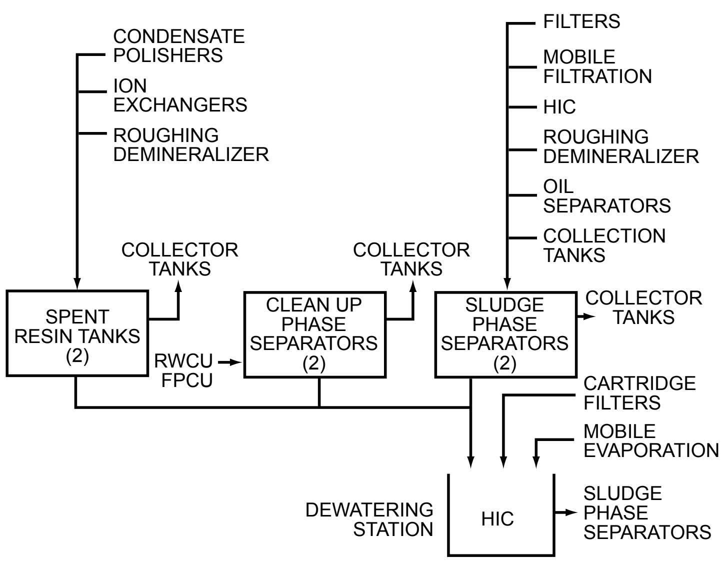


Figure 5-9. Solid Radwaste Flow

Chapter 6
Core and Fuel Design

Introduction and Summary

The design of the Advanced Boiling Water Reactor (ABWR) 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.

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. GE 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 ABWR 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 ABWR are basically the same as employed in previous GE designed plants operating around the world. Key features of the ABWR reactor core design are summarized in the following paragraphs:

- The ABWR 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 1000 psia (6900 kPa), reduce cladding temperatures and stress levels.

- The low coolant saturation temperature, high heat transfer coefficients, and neutral water chemistry of the ABWR 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 ABWR 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 ABWR.

- Because of the large negative moderator density (void) coefficient of reactivity, the ABWR has a number of inherent advantages, including (1) ease of control using coolant flow as opposed to control rods for load following, (2) inherent self-flattening of the radial power distribution, (3) spatial xenon stability, and (4) ability to override xenon in order to follow load. The inherent spatial xenon stability of the ABWR 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 power distributions used in sizing the ABWR core include margins providing for operational flexibility.
- The ABWR fuel assembly pitch is 0.1 inch

more than the conventional BWR fuel assembly pitch 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 ABWR is arranged as an upright cylinder containing a large number of fuel assemblies (872) 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 ABWR reactor core is comprised of fuel assemblies, control rods and nuclear instrumentation. The fuel assembly and control rod mechanical designs are basically the 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 ABWR.

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 GE14 product line is currently GE's most advanced fuel assembly design. The GE14 designs contain 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. Additionally, a new fuel assembly is presently under development. The new assembly is anticipated to be designed, licensed and available for ABWR initial cores. Thus, GE provides two 10x10 fuel designs for the ABWR.

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.

Eight 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.

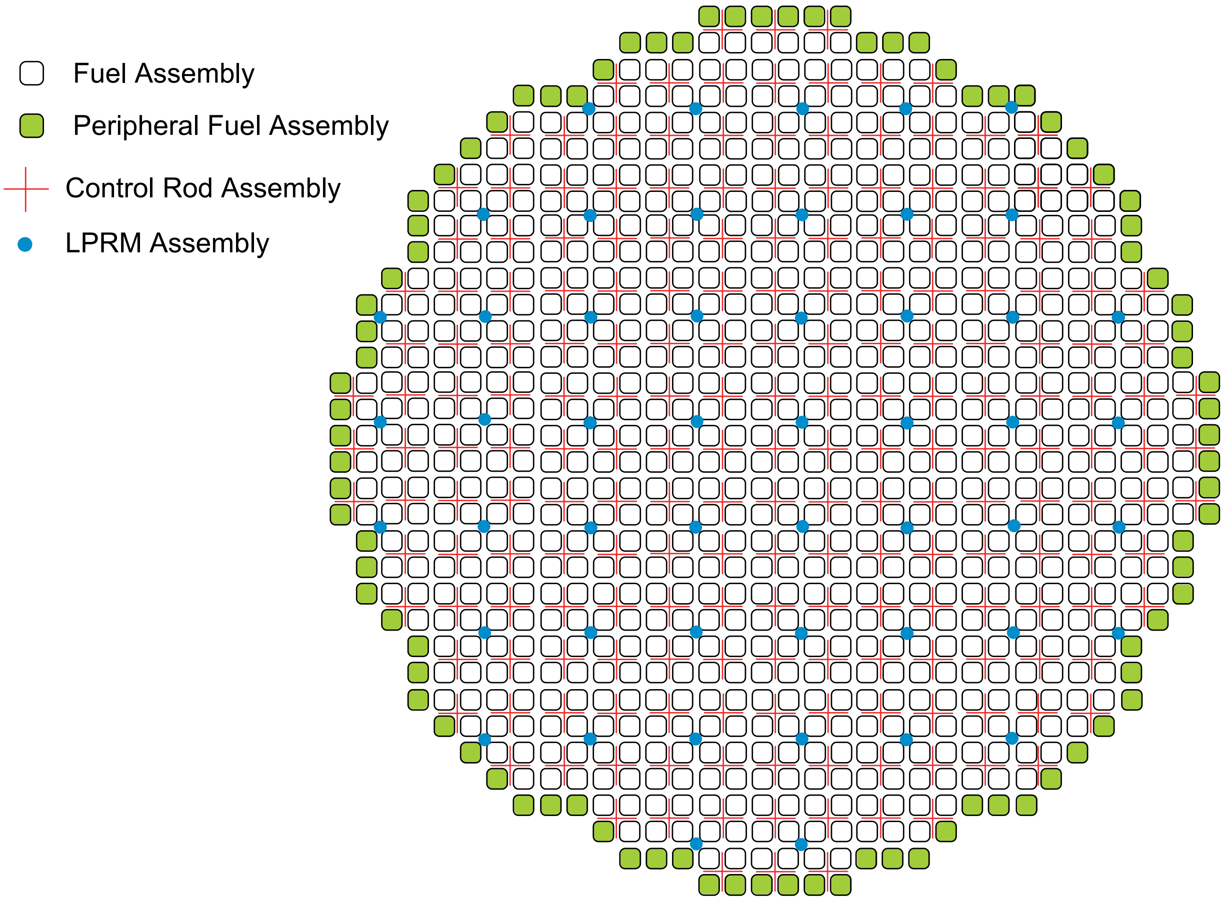


Figure 6-1. ABWR Core Configuration

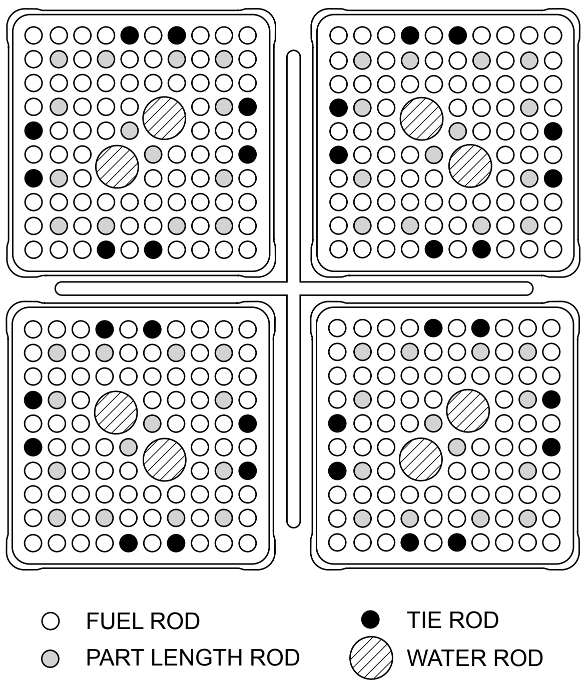


Figure 6-2. Four Bundle Fuel Module (Cell)

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 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

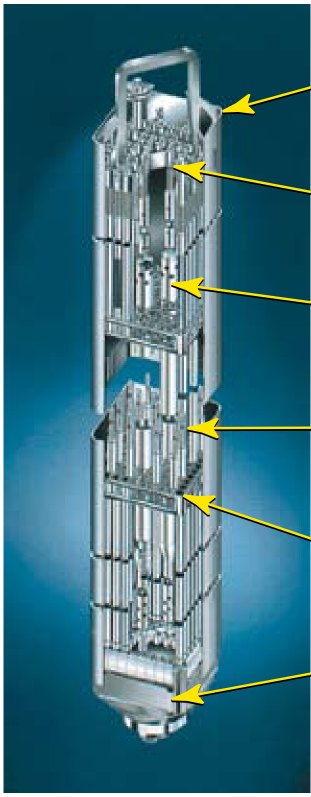


Figure 6-3. GE14 Fuel Assembly

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, high pressure drop lower tie plate with debris filter, large central water rods, and interactive channels. These key design features are individually discussed below.

Part Length Rods

Part length fuel rods (PLRs) were introduced with the GE11 fuel design and have been used in all subsequent GE designs. For GE14, the 14 PLRs terminate just above the fifth spacer 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.

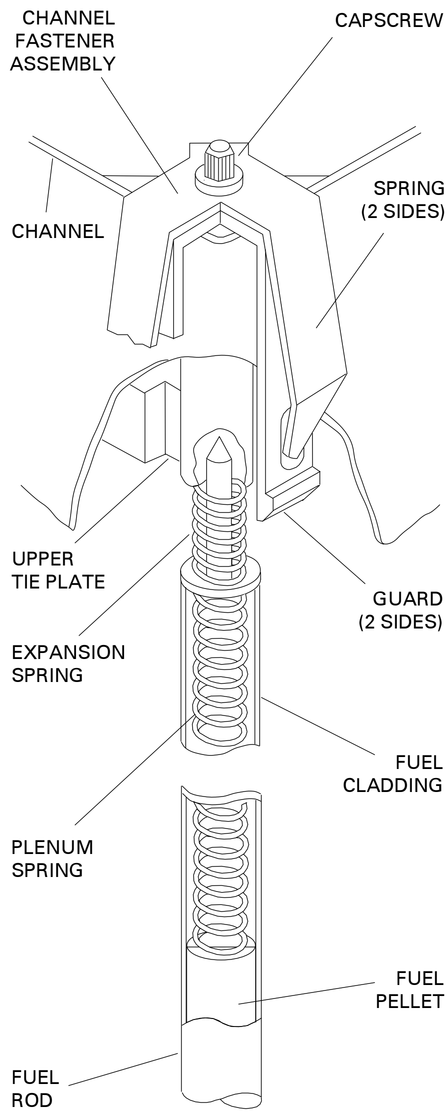


Figure 6-4. Channel Fastener Assembly

High Performance Spacers

The high performance Zircaloy ferrule spacer was developed to provide excellent critical power performance with acceptable pressure drop characteristics. This spacer concept is also used in a number of previous products. Eight spacers are used to maintain rod bow and flow-induced vibration margins for the reduced diameter 10x10 fuel rods of the GE14 design, while at the same time providing additional critical power capability.

Low Pressure Drop Upper Tie Plate

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 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, which serve to effectively filter debris. Figure 6-5 shows a top view of one of three possible debris filter LTPs. Debris filter LTP designs are standard with all GE fuel designs. Any of GE's debris filter LTPs can be applied to ABWR, as they all have the same hydraulic resistance.

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 6-6, 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.

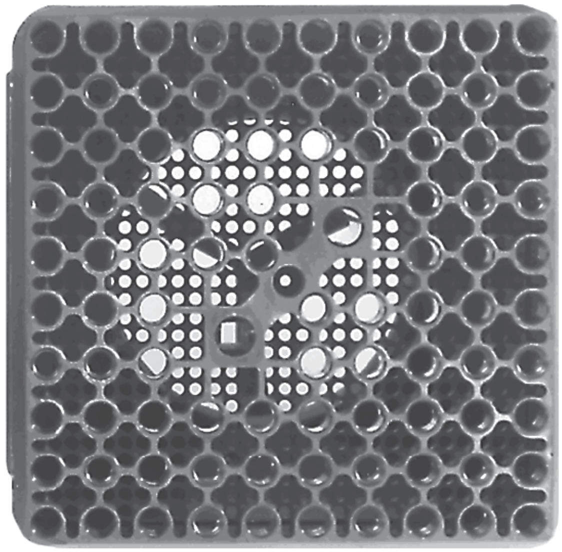


Figure 6-5. Top View of Debris Filter Lower Tie Plate

Control Rod Description

As shown in Figures 6-1 and 6-2, cruciform shaped 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

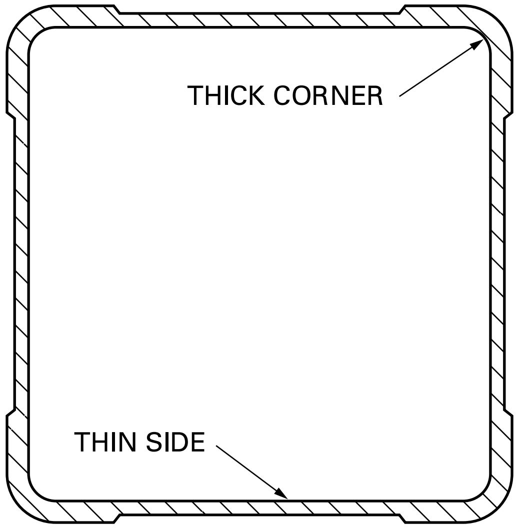


Figure 6-6. Cross Section of Interactive Channel

shutdown) or reactivity regulation. Because of this, the control elements are mechanically designed to withstand the dynamic forces resulting from a scram. In the ABWR, they are connected to bottom-mounted drive mechanisms which provide electric-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.

Typically, the cruciform control rods contain stainless steel tubes in each wing of the cruciform filled with boron carbide (B_4C) powder compacted to approximately 75% of theoretical density. The tubes are seal welded with end plugs on either end. Stainless steel balls are used to separate the tubes into individual longitudinal compartments. The stainless steel balls are held in position by a slight crimp in the tube. The individual tubes act as pressure vessels to contain the helium gas released by the boron-neutron capture reaction.

The tubes are held in cruciform array by a stainless steel sheath extending the full length of the tubes. A top casting and handle, shown in Figure 6-7, aligns the tubes and provides structural rigidity at the top of the control rod. Rollers, housed by the top casting, provide guidance for control rod insertion and withdrawal. A bottom casting is also used to provide structural rigidity and contains positioning rollers and a coupler for connection to the control rod drive mechanism. The castings are welded into a single structure by means of a small cruciform post located in the center of the control rod. Control rods are cooled by the core leakage (bypass) flow.

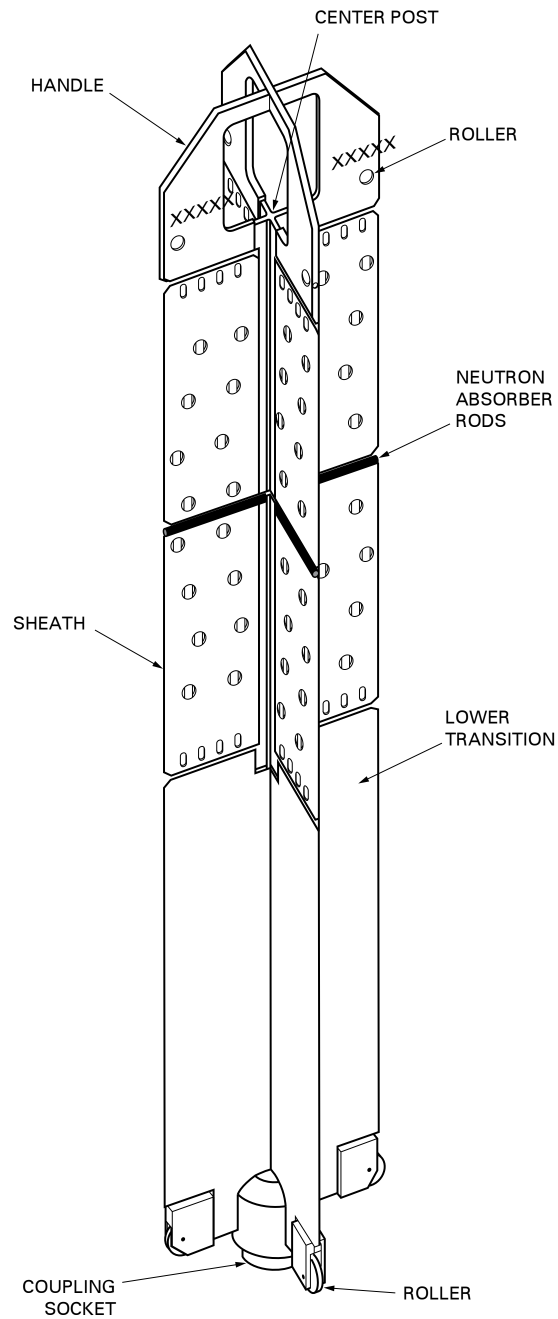


Figure 6-7. ABWR Control Rod

In addition to boron carbide, hafnium absorber may be placed in the highest burnup locations of select control rods, the full length outside edge of each wing and, optionally, the tip of each wing. Hafnium is a heavy metal with excellent neutron absorbing characteristics and does not swell at high burnups.

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 outer zone of fuel assemblies, located near the core periphery, has more restrictive orifices than the inner zone. 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 10 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 52 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 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 traversing in-core probe (ATIP) calibration tube. The ATIP sensor moves within the calibration tube to provide an axial scan of the neutron flux at that LPRM assembly location. A schematic of the ATIP, LPRM and SRNM assemblies is shown in Figure 6-8.

Neutron Sources

Several antimony-beryllium 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

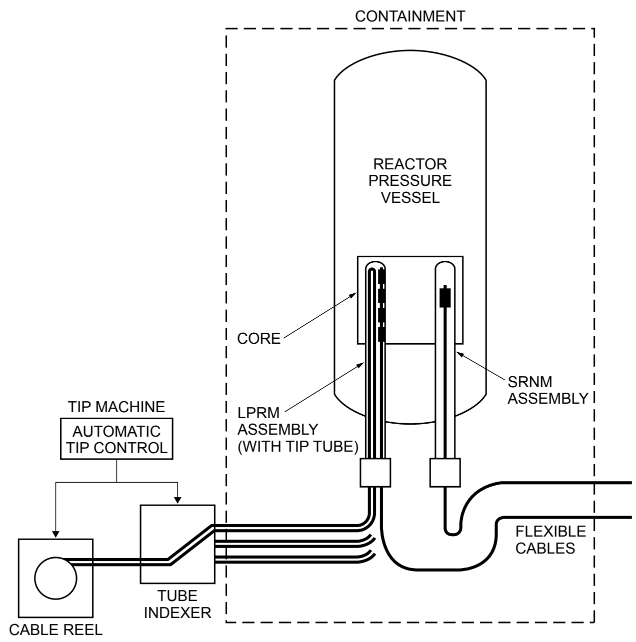


Figure 6-8. ABWR ATIP, LPRM and SRNM Schematic

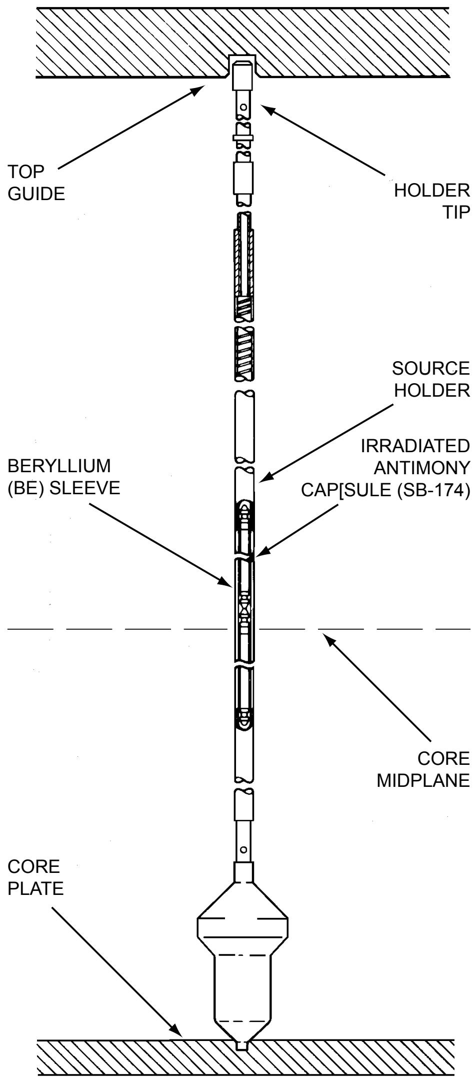


Figure 6-9. Neutron Source Schematic

disturbed during refueling.

The active portion of each source consists of a beryllium sleeve enclosing two antimony-gamma 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 beryllium sleeve and the antimony-gamma 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, GE 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 ABWR core map is illustrated in Figure 6-1. There are 872 fuel assemblies, 205 control rods and 52 LPRM assemblies. Also the core periphery zone with more restrictive inlet flow orifices is shown.

Additionally, typical control cell locations are shown in Figure 6-1. ABWR 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 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 UO_2 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 ABWR 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 ABWR to xenon instability permits significant flexibility in load-following capability.

  –**Load Changing by Flow Control:** Since the fuel Doppler reactivity opposes a change in load, the void effect must be (and is) larger than the fuel Doppler effect in order to provide load changing capability by flow (or moderator density) control. The ABWR is capable of daily load following between 100% and 50% power by adjusting core flow, with only minor adjustments to the control rod pattern at low power.

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 digadolinia trioxide (Gd_2O_3), 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 GE BWRs since the early 1970s, 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 fully withdrawn. 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 ABWR 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.

The average bundle enrichments and batch sizes are a function of the desired cycle length. The initial ABWR 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 ABWR 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 Automatic Traversing In-Core Probe (ATIP) Subsystem, and the Multi-Channel Rod Block Monitoring (MRBM) Subsystem.

The NMS design has been greatly simplified for ABWR application. Key simplification features include the SRNM, period-based trip logic, and automation of the Traversing In-core Probe (TIP) 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. TIP System operation for core flux mapping and calibrating the power range monitors has been fully automated in the ABWR design, thereby substantially enhancing operability.

Startup Range Neutron Monitoring (SRNM) Subsystem

The SRNM Subsystem monitors the neutron flux from the source range to approximately 15% of the rated power. 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 10 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 52 detector assemblies with 4 detectors per assembly. The 208 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 Traversing In-Core Probe (ATIP) Subsystem

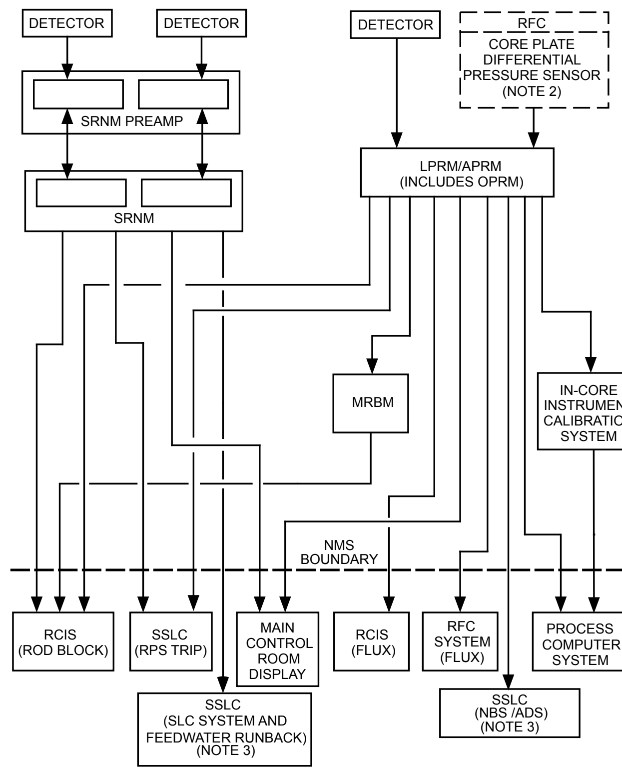
This is a single non-safety processor system included in the NMS used to provide steady state local power information for LPRM calibration and three dimensional reactor power determination.

The ATIP controller contains the purge control system, flux amplifiers and automatic/manual TIP sequencing controls for the three TIP machines located in the Reactor Building. The ATIP Subsystem performs an axial scan of the neutron flux in the core at the LPRM assembly locations. The subsystem can be controlled manually by the operator, or it can be under micro-processor-based automated control. The ATIP Subsystem typically consists of neutron-sensitive ion chambers, flexible drive cables, guide tubes, indexing machines, drive machines, and an automatic control system.

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. The rod block setpoint is a core flow biased variable setpoint.

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 6-10.



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. USED FOR RAPID CORE FLOW DECREASE TRIP.
3. SRNM AND APRM ATWS PERMISSIVE SIGNALS TO SSLC.
4. INTERCONNECTIONS MAY BE FIBER-OPTIC OR METALLIC.

Figure 6-10. Basic Configuration of a Typical Neutron Monitoring System Division

Chapter 7
Instrumentation and Control

Overview

The ABWR instrumentation and control (I&C) design features system redundancy, fault tolerant operation, and self-diagnostics while the system is in operation. This is made possible by the extensive use of advanced digital technologies. The ABWR I&C system represents the largest system change from previous BWR designs.

Previous BWRs used hard wired point-to-point control room to field monitoring and control systems; essentially there was one wire per function or ~30-50,000 wires coming from the field to the cable spreading room and then control room. The ABWR instead is designed with a three-layer I&C system that uses extensive multiplexing and fiber optics. The three layers are:

- Remote Multiplexer Units (RMUs) in the field. This equipment generally handles 300-400 signals per RMU and interfaces the I&C system with the normal field signals and actuators.
- A computer/controller layer. This layer has all of the dual and triple redundant controllers that operate the plant and a networked computer system - there is no single process computer.
- A display, control and alarm/annunciator layer. This layer is basically all the screens, peripherals and alarms in the control room and forms the I&C interface to the operator.

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

Design goals of the I&C system include:

- Minimize reactor trips/system unavailability due to human errors or 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.
- Achieve a one-in-fifty-year or less failure rate for I&C.
- Computerize operator aids and normal/emergency procedures to reduce "manual" data processing and centralize 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 and reduce the burden on the maintenance staff.
- Achieve a high degree of plant automation.
- Provide automatic load-following capability over the 50-100% power range.
-

Digital Measurement and Control

A standardized set of microprocessor-based instrument modules is used to implement most ABWR monitoring and control functions. The standardized Digital Measurement Controllers DMC) and Remote Multiplexer Units (RMU) exploit the many advantages of digital technology, including self-test, automatic calibration, user interactive front panels, standardization of the man-machine interface and, where possible, use of common circuit cards. These features reduce calibration, adjustment, diagnostic and repair time and reduce spare circuit card inventory requirements, as well as reduce control room instrument volume. As a result, system availability is improved due to the enhanced reliability and reduced mean time to repair.

The DMC chassis, RMU chassis and Man-Machine Interface (MMI) chassis are standard for all similar ABWR applications; only modular, plug-in interchangeable, circuit boards differ between systems. Various functional features provided in the I&C design include:

- Sensor signal processing.
- Redundant sensor power supplies to meet the requirements of all sensors.
- Functional microcomputers implementing data transfers, self-test functions and communications.
- High speed parallel data bus for communication between the functional microcomputer and other modules.
- Trip and analog outputs driving external relays, actuators, logic circuits, meters, and recorders.
- Redundant power supplies for the electronics.
- Fiber optic and other interfaces, allowing the DMC and MMI units to communicate directly with plant multiplexing networks.
- Menu-driven front panel for operator/technician interface.

Multiplexing

The Multiplexing System provides redundant and distributed control and instrumentation data communications networks to support the monitoring and control of interfacing plant systems. The system contains an Essential Multiplexing System (EMS) and a Non-Essential Multiplexing System (NEMS) for safety and non-safety/BOP systems, respectively. The system provides all electrical devices and circuitry (such as multiplexing units, bus controllers, formatters and data buses) between sensors, display devices, controllers and actuators which are defined and provided by other plant systems. The multiplexing system also includes the associated data acquisition and communication software required to support its function of plant-wide data and control distribution. As shown on Figure 7-1, digital technology and multiplexed fiber optic signal transmission technology have been combined in the ABWR design to integrate control and data acquisition for both the Reactor and Turbine Buildings.

Signals from various plant process sensors provide input to RMUs located near the sensors. The RMUs digitize input signals and multiplex the signals via fiber optic cables to the control room. 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 network and directed to the various actuators for control action.

The EMS has four control data networks (each of which is redundant), one per division with the NEMS being a control data network with dual redundancy. Whether EMS or NEMS, redundancy is such that a single cable can be lost or any RMU fail without affecting the operation of the remainder of the system.

Finally, 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 in 1E and non-1E areas to keep plant wiring as short as possible.

Digital Protection System Applications

Advanced Safety Systems Design

The Reactor Protection System (RPS), Neutron Monitoring System (NMS) and Leak Detection and Isolation System (LDI) are four-channel, while the ECCS is three-mechanical divisions actuated by two-out-of-four logic from four-channel sensor input. NMS is described in Chapter 6.

Safety System Logic and Control

Safety System Logic and Control (SSLC) and the associated EMS equipment are divided into four divisions. Each division is physically and electrically separated from the other divisions. Communications between divisions, as with communications with the NEMS, process computer, and control room instruments, is via fiber optic cable which provides complete electrical isolation and prevents spreading of electrical faults between safety system divisions and between safety and non-safety-related equipment. Communication between safety divisions and nonessential equipment is through "Data Gateways" which allow information to flow in only one direction.

Some control signals bypass the EMS when the signal design requirements are such that processing the signal through the EMS would cause the established design requirements (signal processing speed) to be exceeded.

The SSLC also controls the automatic actuation and operation of the following systems during emergency operation:

- High Pressure Core Flooder
- Reactor Core Isolation Cooling
- Residual Heat Removal
- Automatic Depressurization System
- Emergency Diesel Generators
- Reactor Building Service Water/Ultimate Heat Sink
- Reactor Building Cooling Water

Standby Liquid Control (SLCS) and Standby Gas Treatment System (SGTS) logic are separate from SSLC.

Reactor Protection System

The Reactor Protection System (RPS) is the overall complex of instrument channels, trip logic, trip actuators and scram logic circuitry that initiate

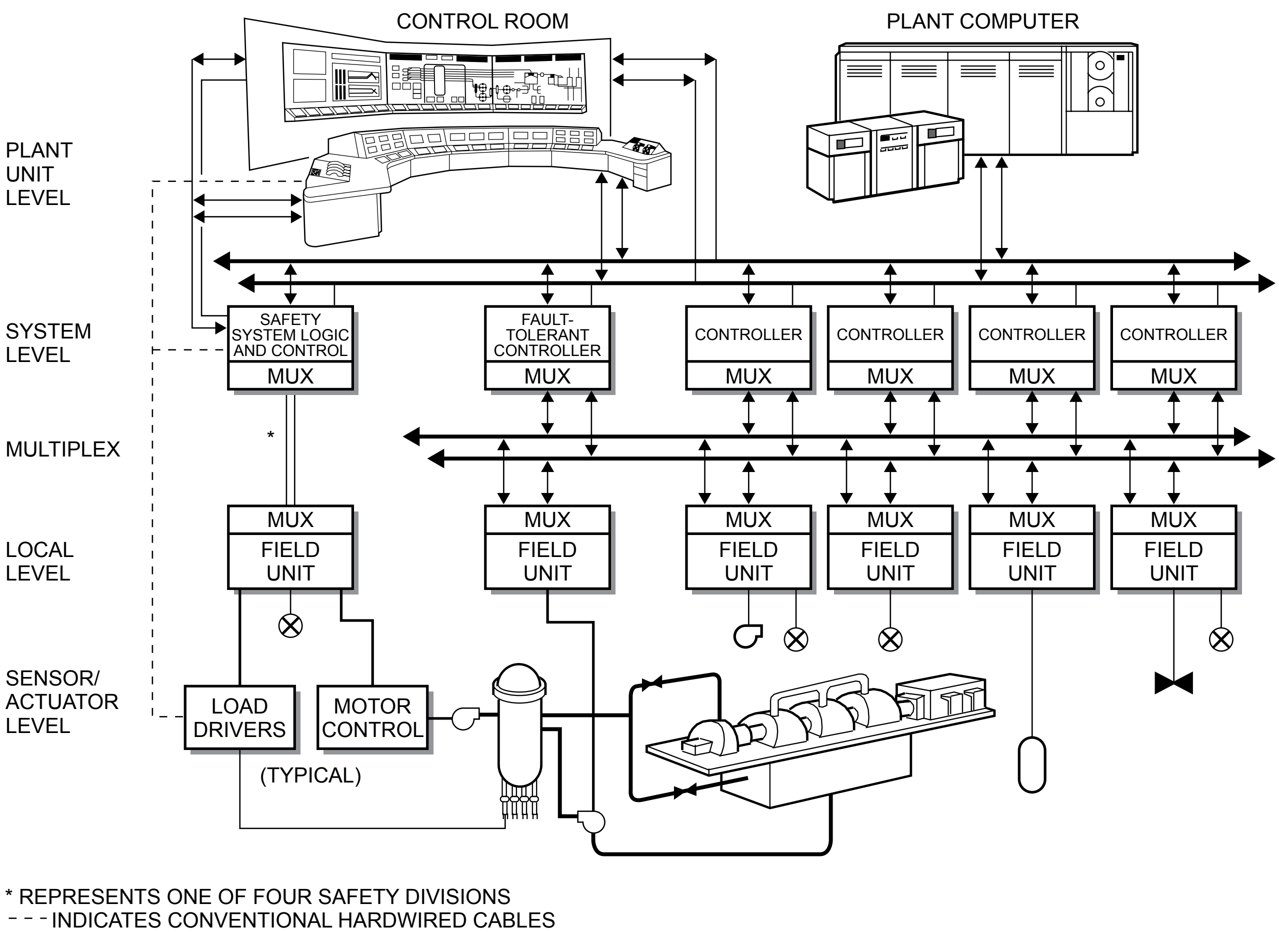


Figure 7-1. ABWR Integrated Multiplexing System Architecture

rapid insertion of control rods (scram) to shut down 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 annunciators. To accomplish its overall function, the RPS interfaces with the Essential Multiplexing System, Neutron Monitoring System, Process Radiation Monitoring System, Control Rod Drive System, Rod Control and Information System, Reactor Recirculation Control System, Process Computer System, Leak Detection and Isolation System, Nuclear Boiler System and associated plant systems and equipment.

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 redundant two-out-of-four logic arrangement that automatically reconfigures to a two-out-of-three logic if a channel fails or is bypassed. Manual RPS actions (scrams) are hard wired and always available to the operator.

Leak Detection and Isolation System

The Leak Detection and Isolation System (LDI) is a four-channel system that consists of temperature, pressure, flow and fission-product sensors with associated instrumentation, alarm, and isolation functions. This system detects and annunciates leakage and provides signals to close containment isolation valves, as required, in the following systems:

- Main Steamlines
- Reactor Water Cleanup System
- Residual Heat Removal System
- Reactor Core Isolation Cooling System
- Feedwater System
- Emergency Core Cooling Systems
- 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 MSIV is provided with a separate manual control switch in the control room which is independent of the automatic and manual leak detection isolation logic.

Fault-Tolerant Process Control Systems

The entire ABWR control system necessary for power generation is made up of a network of triple redundant and dual redundant Fault Tolerant Digital Controllers (FTDCs). Single controllers are used where the function is not important to power generation. In general, the key ABWR boiler control systems such as the feedwater control, recirculation flow control, turbine control, automatic power regulator and reactor pressure regulator systems are based on the triplicated, microprocessor-based FTDC. The remaining important BOP control systems are based on dual redundant FTDCs. Each FTDC includes two or three identical processing channels, which receive all the redundant process sensors inputs and perform the system control calculations in parallel.

For triple redundant process control, all FTDCs are active simultaneously and each provides an output to the NEMS network to the RMUs where the outputs are two-out-of-three voted (mid-value voting on continuous output signals (e.g., valve position demand) and two-out-of-three voting on discrete outputs (e.g., pump trip). Thus, the FTDC design eliminates plant trips due to single failures of control system components.

For dual redundant process control, one FTDC is active and the other is in “hot standby”; only one processor at a time provides an output to the NEMS network and to the RMUs but the other FTDC is "live" and can automatically and bumplessly assume command if the primary FTDC fails.

All important control signals are typically measured with three independent transducers (and occasionally measured with two); these input signals are delivered to all controllers by the NEMS and validated before control action is taken. This scheme and the controller redundancy eliminates plant trips due to single failures of control system components.

The FTDC hardware for all two or three process control systems is identical. Only the imbedded application firmware and the quantity and types of input and output modules deviate between the systems. The FTDC architecture includes:

- Two or three identical processing channels, each of which contains the hardware and firmware necessary to control the system.
- Dual multiplexing interface units per controller for communication to the redundant Non-Essential Multiplexing System.
- Interprocessor communication links between processing channels to exchange data in order to prevent divergence of outputs and to monitor processor failures.
- Redundant power supplies.
- Signal processing techniques applied to validate the redundant input signals for use in control computations.
- A portable Technician Interface Unit (TIU) to provide a menu-driven system which allows the technician to inject test signals, perform troubleshooting and calibrate process parameters.

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 dual and triplicated design also provides on-line repair capability to allow repair and/or replacement of a faulty component without disrupting any important plant process.

Automatic Power Regulator System

The primary objective of the Automatic Power Regulator System (APR) is to control reactor power during normal power generation by appropriate commands to change rod positions, or the change reactor recirculation flow. Either thermal power or gross generator electrical power can be controlled/demanded by the operator. Alternatively, the operator can engage a pre-programmed daily load-following schedule. The APR System always follows a predefined "trajectory" on the power/flow map for any mode of power operation.

The APR System also has the ability to pull the reactor critical and heat it to rated temperature and pressure from either a cold or hot standby condition. The APR System can also bring the reactor down to cold shutdown conditions. For either heatups or cooldowns, the reactor temperature rate is controlled to within Tech Spec limits by the APR commands to the Steam Bypass and Pressure Control System (SBPC) and RCIS.

The APR System consists of triply redundant process controllers; these receive information from the various plant sensors and issue commands to the RCIS to position control rods, to the RFC System to change reactor coolant recirculation flow, to the SBPC System to set pressure.

The APR System generally controls the nuclear control systems and works in parallel (but not synchronously) with the Power Generation Control System function of the plant Process Computer System; the latter system controls most other automation functions. The normal mode of operation of the APR System is automatic but if any abnormal plant condition is detected or if the downstream controllers receiving the APR commands fail or are switched to manual, the APR will automatically cease control operations, switch all downstream controllers to manual, and alarms will be activated to alert the operator. A failure of the APR System will not prevent manual controls of reactor power, nor will it prevent safe shutdown of the reactor.

Feedwater Control System

The Feedwater Control System (FWC) automatically controls the flow of feedwater into the reactor pressure vessel to maintain the water within the vessel at normal and predetermined levels for all modes of reactor operation, including heatup and shutdown. The operator can control reactor level between the requirements of the steam separators (this includes limiting carryover, which affects turbine performance, and carryunder, which affects RIP operation).

A fault-tolerant triplicated digital controller, using a conventional three-element control scheme, provides control signals to adjustable speed drives (ASDs) for the feedwater pump motors, to accomplish the control function.

The FWC System may operate in either single- or three-element control modes. At feedwater and steam flow rates below 25% of rated when the steam flow measurement is outside of the required accuracy or below scale, the FWC System utilizes only water level measurement in the single-element control mode.

When steam flow is negligible, as during heatup and cooldown, the FWC System automatically controls both the Reactor Water Cleanup (RWCU) System dump valve and the feedwater low flow control valve to control reactor level in the single element mode in order to counter the effects of density changes during heatup and purge flows into the reactor. At higher flow rates, the FWC System in three-element control mode uses water level, main steamline flow, main feedwater line flow, and feedpump suction flow measurements for water level control.

Steam Bypass and Pressure Control System

The Steam Bypass and Pressure Control System (SBPC) is a triply redundant process control system: in Manual, the operator can adjust bypass valve position and provide reactor pressure setpoint demands; in Automatic, these functions are provided by the APR. Only the operator can switch the SBPC System to Automatic but either the operator or the APR can switch the SBPC System to Manual.

Unlike previous BWRs, reactor pressure and not turbine inlet pressure is controlled by the SBPC System. In normal power generation, reactor pressure is controlled by automatically positioning the turbine control valves - the pressure control signal "passes through" the SBPC System to the turbine control system. During modes of operation where the turbine is off-line, flow limited, tripped or under control of its speed/acceleration control system during turbine roll or coastdown, reactor pressure is controlled by the bypass valves which pass steam directly to the main condenser under the control of the pressure regulator. Steam is also automatically bypassed to the condenser whenever the reactor steaming rate exceeds the flow permitted to pass to the turbine generator. With a full bypass design option, the turbine bypass system has the capability to shed up to 100% of the turbine-generator rated load without reactor trip or operation of SRVs. For all these modes of operation, the pressure regulation system provides main turbine control valve and bypass valve flow demands so as to maintain a nearly constant reactor pressure; it also indirectly (through the APR) provides demands to the recirculation system to optionally aid in maintenance of grid frequency.

Recirculation Flow Control System

The Recirculation Flow Control (RFC) System is a triply redundant process control system: in Manual, the operator can adjust individual or gang RIP speeds or demand a specific core flow; in Automatic, these functions are provided by the APR. Only the operator can switch the RFC System to Automatic but either the operator or the APR can switch the RFC System to Manual.

The RFC System consists of three redundant process controllers, adjustable speed drives (ASDs), switches, sensors, and alarm devices provided for operational manipulation of the ten RIPs and the surveillance of associated equipment. The solid-state ASDs provide variable voltage, variable frequency electrical power to the RIP induction motors. In response to either the plant operator or the APR or, optionally, grid frequency demands, the RFC System adjusts the ASD power supply output to vary RIP speed, core flow and reactor power. Extremely rapid reactor power changes can be achieved either by manual operation or by automatic operation from ~65-100% reactor power.

The objective of the RFC System is to control reactor power level, over a limited range, by controlling the flow rate of the coolant flow through the reactor. To change the coolant flow rate through the core, the speed of the RIPs are adjusted, either together in the gang mode or individually by commands from the RFC System to the ASDs controllers of the individual RIPs. The RIPs can be driven to operate anywhere between 30 to 100% of rated speed with the variable voltage, variable frequency power source supplied by the ASDs. Due to the low rotating inertia of the RIPs, which are coupled with the solid-state ASDs, the RIP can respond quickly to load transients and operator demands.

Turbine Control System

The Turbine Control System is a redundant process control system: in Manual, the operator can adjust the turbine load set; in Automatic, this function is provided by the APR. Only the operator can switch the turbine controller to Automatic but either the operator or the APR can switch the turbine controller to Manual.

The turbine generator uses a digital monitoring and control system which, in coordination with the turbine SBPC 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. 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.

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; the trip logic for all but bearing vibration is at least two-out-of-three logic.

Other Control Functions

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

Power Generation Control System

The Power Generation Control System (PGCS) is a subset of the process computer function implemented as a dual redundant controller. The APR System provides automation of the reactor control functions and the PGCS provides other nuclear island and BOP automation functions by providing the setpoints of lower level controllers and commands to various BOP equipment for normal plant startup, shutdown, and power range operations.

The PGCS works in parallel but is not synchronous with the APR System; one of the design features of the PGCS is that it contains no control algorithms but instead issues only supervisorial commands to the BOP controllers and systems which otherwise remain responsible for their own availability and operation. PGCS contains the algorithms for the automated control sequences associated with plant startup, shutdown, and power range operations.

The plant operator interfaces with the PGCS through a series of breakpoint controls to initiate automated sequences from the operator control console. In general, plant automation is broken down into logical steps and sequences like heatup or turbine roll which the operator can initiate and which then proceed to completion and halt until the operator initiates the next sequence. For selected operations that are not automated or that are contained within 1E systems, the system prompts the operator to perform such operations. A semiautomatic mode is also provided where the PGCS provides only guidance messages to the operator but does not actually operate plant equipment.

Rod Control and Information System

The Rod Control and Information System (RCIS) is a dual redundant process control system: in Manual, the operator can select and position the control rods manually, either one at a time or in a gang mode. If the RCIS is in Semi-Automatic mode, the operator needs to only give permission to start and stop control rod motion and the RCIS will insert or withdraw the control rods following a predefined control rod sequence. If the RCIS is in Automatic mode, it responds to commands for rod insertion or withdrawal from the APR; this will also follow a predefined control rod sequence. Only the operator can switch the RCIS controllers to Automatic, but either the operator or the APR can switch the RCIS to Manual.

The RCIS provides the means by which control rods are positioned from the control room for power control. The RCIS controls changes in the core reactivity, power, and power shape via the FMCRD mechanisms which 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, 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. The system includes the logic that restricts control rod movement (rod block) under certain conditions as a backup to procedural controls.

The RCIS contains as a subsystem, the ATLM (automatic thermal limit monitor), which provides an on-line measurement of plant thermal limits from the LPRMs and periodic process computer updates. The ATLM will automatically block rod motion if it detects operation near Tech Spec thermal limits.

Another RCIS subsystem is the Rod Worth Minimizer (RWM) Subsystem, which forces compliance to the defined control rod sequencing rules by independently issuing rod blocks should a high worth rod pattern develop.

The RCIS and the scram timing panel also support automatic measurement of control rod Tech Spec scram speeds for either planned or unplanned scrams.

Process Radiation Monitoring System

The Process Radiation Monitoring System (PRM) monitors and controls radioactivity in process and effluent streams and activates appropriate alarms, isolations, and controls. The PRM System 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. These include the following:

- Main steamline tunnel area
- Reactor Building ventilation exhaust (including fuel handling area)
- Control Building air intake supply
- Drywell sumps liquid discharge
- Radwaste liquid discharge
- Offgas discharge (pretreated and post-treated)
- Gland steam condenser offgas discharge
- Plant stack discharge
- Turbine Building vent exhaust
- Radwaste Building ventilation exhaust.

Area Radiation Monitoring System

The Area Radiation Monitoring (ARM) System 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 where nuclear fuel is stored or handled.

The ARM System 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 process computer. 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 Atmospheric Monitoring System

The Containment Atmospheric Monitoring (CAM) System measures alarms and records radiation levels and the hydrogen and oxygen concentration in the primary containment under post-accident conditions. It is automatically put in service upon detection of LOCA conditions.

The CAM System provides normal plant shutdown and post-accident monitoring for gross gamma radiation and hydrogen/oxygen concentration levels in both drywell and suppression chamber. The CAM System consists of two divisions which are redundantly designed so that failure of any single element will not interfere with the system operation. 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 (high drywell pressure or low reactor water level). The CAM System does not actuate nor interface with any other safety-related systems.

Process Computer

On-line networked computers are provided to monitor and log process variables and make certain analytical computations. The process computer cabinets are really several redundant computer functions that may, in fact, be several physical computers. These functions include:

- Most non-1E display support
- Core three-dimensional power monitoring (3D Monicore)
- Balance-of-plant (BOP) performance calculations
- Sequence of events
- Manual and automatic logging

Remote Shutdown System

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. Manual transfer devices are provided which override control outputs from the main control room and transfer controls to remote shutdown control. Control signals (but not process signals, since they interface with field RMUs and not the control room) are interrupted by the transfer switches. All necessary power supply circuits are also transferred to other sources. Operation of the transfer switches causes an alarm in the main control room; outside the main control room, access to the remote shutdown control panels is administratively and procedurally controlled. The Remote Shutdown System (RSS) functions ahead of the plant multiplexing system: all controls and indications are hard wired and will function regardless of the status of the multiplexing system.

Instrumentation and controls on the remote shutdown panels include the following:

- Controls and indications for operation of one HPCF loop to control reactor water level.
- Controls and indications for operation of two RHR loops to support shutdown cooling once reactor pressure has been reduced, and suppression pool cooling to control suppression pool temperature which may rise due to SRV operation.
- Controls to operate three SRVs for maintaining and reducing reactor pressure.
- Indications of reactor vessel water level and pressure, and suppression pool temperature and level.
- Controls and indications for operation of the RCW and RSW Systems.
- Controls and indications for electrical power distribution.
- Controls for manually starting and stopping two of the emergency diesel generators.

Main Control Room

The key elements of the ABWR main control room (MCR) design (Figure 7-2) are (1) the compact main control console (MCC) for primary operator control and monitoring functions, and (2) the integrated wide display panel, which presents an overview of the plant status that is clearly visible to the entire operating crew. Each of the units incorporates advanced man-machine interface technologies to achieve enhanced operability and improved reliability. Human factors engineering principles have been incorporated into the design of the MCR panels and into the overall MCR arrangement.

Total plant control is achieved from the main control console for all phases of operation. The console design incorporates touch-screen cathode ray tubes (CRTs), flat panel display devices, and a limited number of hard switches as the primary operator interface devices. The CRTs and flat panel displays are driven by the Plant Computer System (PCS). The main control console has a low profile so that the operators can perform their duties from

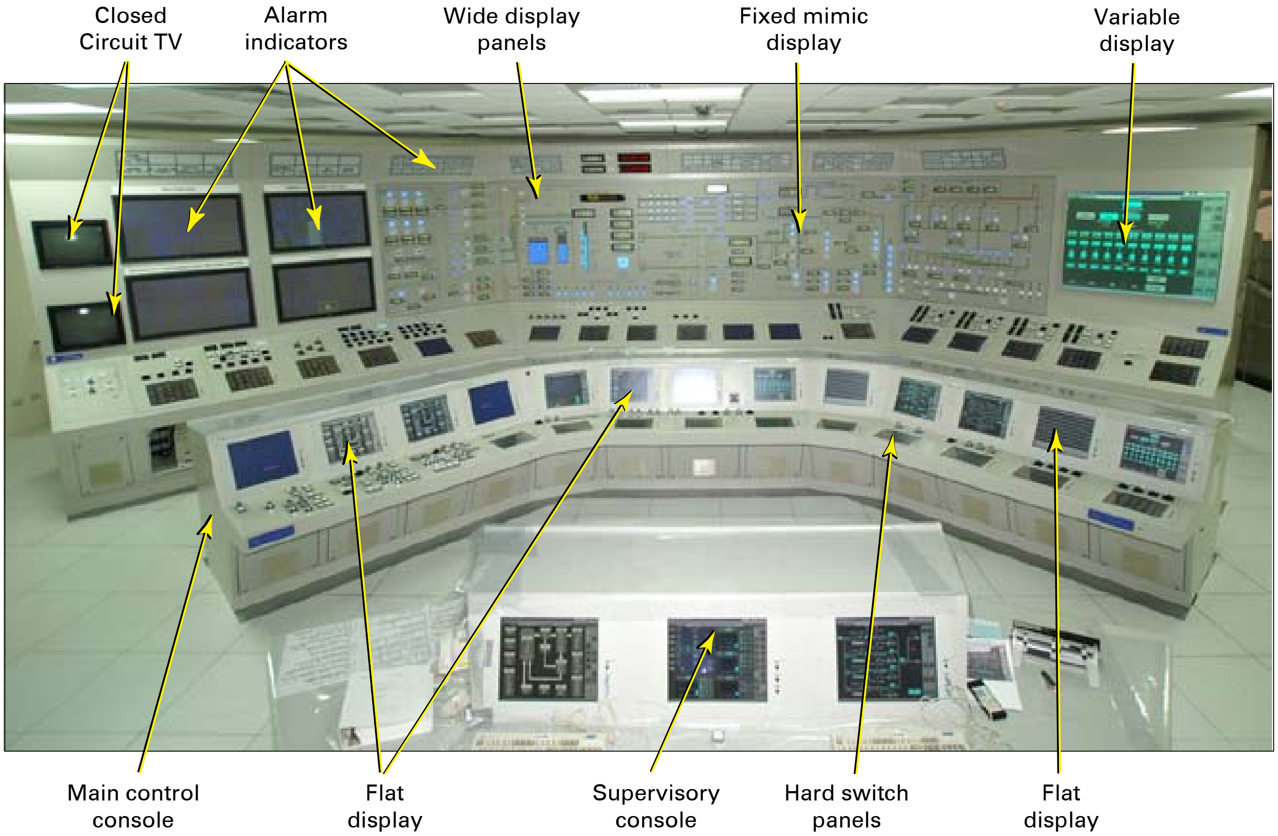


Figure 7-2. ABWR (Lungmen) Main Control Room

a seated position.

The Wide Display Panel provides summary information on plant status parameters and key alarms to the operators, supervisors and other technical support personnel in the MCR. The Wide Display Panel is located immediately in front of the operators when they are at their normal work station seated at the main control console. This Wide Display Panel includes a fixed mimic display, an approximately 100-inch large variable display, top-level plant alarms, detailed system level alarms, and touch-control flat panel displays. The Wide Display Panel incorporates the Safety Parameter Display System (SPDS) as part of the plant status summary information.

The MCR also includes a supervisors' console which has CRTs for monitoring of plant status. The supervisors' console is set back directly behind the operators in a position which ensures that a clear view of all operating activities is available to the supervisors.

Main Control Console

The 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 Wide Display Panel.

The MCC comprises the work stations for the two control room plant operators, and is configured such that the operators are provided with controls and monitoring information necessary to perform assigned tasks and allows the operators to view all of the Wide Display Panel from their seated position at the MCC. The console is configured in a truncated "V" shape. The normal plant control and monitoring functions are performed in the central area of the console, while the safety-related Nuclear Steam Supply (NSS) functions are located on the left-hand side and the balance-of-plant (BOP) functions are located on the right-hand side.

A primary means for operator control and monitoring is provided by the color-graphic, touch-screen CRTs mounted on the MCC. The CRT displays are driven by the Process Computer. There are many types of display formats which can be shown on the CRTs, including summary plant status displays, trend plots, system status formats, alarm summaries, plant operating procedure guidance displays, and plant automation guidance displays. Although each CRT is assigned a default display for a given operating condition, the operators have the flexibility to select any display on each of the seven touch-screen CRTs. This multi-redundant display capability ensures continued normal plant operation in the event of a failure of one or more of the CRTs.

The system status displays provide information on individual plant systems. The touch screens on the CRTs provide direct control for non safety-related systems at the system component level. The application of this touch screen capability for control of non safety 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 on the MCC CRTs support the operators' decision-making process. The presentation of alarms employs optimization techniques designed to prioritize alarms and filter or suppress nuisance alarms which require no specific operator action. An example of this alarm processing would be the suppression of the audible alarms associated with the Reactor Protection System during the period of a reactor scram.

The ABWR MCC also provides flat panel displays (e.g., electroluminescent, plasma, or liquid crystal displays) for extended monitoring and control capability. These touch-control flat panel displays are driven by microprocessor-based controllers which are completely diverse from the controllers. This diversity of displays and controls in the console design enables continued plant operation even in the unlikely event of a total loss of all CRTs.

The flat panel display devices are used to support both safety and non safety system monitoring and control functions. The flat panel displays which are used as safety-system interfaces are fully qualified to Class 1E standards. The safety-related flat displays are located on the left side of the MCC. For control and monitoring of the three redundant and independent divisions of the Emergency Core Cooling System (ECCS) and reactor primary containment heat removal, two flat panel display devices are provided in each of those divisions. One flat panel display is typically used for monitoring and the other is used for control. Flat panel displays for monitoring and control of major non safety systems are also located on the MCC.

In addition to the touch-screen CRTs and flat panel display devices described above, the MCC is equipped with dedicated, "hard" switches located on the horizontal desk surfaces of the console. Some of these hard switches are the sequence master control push-button switches used for initiating automation sequences for normal plant operations and for changing system operating modes. Other hard switches are hard-wired directly to the actuated equipment (for absolute assurance of function) and provide backup capability for initiating safety system functions and key plant protection features, such as manual scram, SLCS initiation and turbine trip functions.

A limited number of dedicated operator interfaces are provided in the center of the MCC for key systems such as the Rod Control and Information System. These dedicated interfaces contain hard switches and indicators to provide quick and convenient access to key system interfaces under all plant conditions.

Wide Display Panel

The Wide Display Panel (WDP) is a large vertical board which provides information on overall plant status with real-time data during all phases of plant operation. The information presented on the wide display panel is clearly visible from the main control console, the supervisors' console, and other positions in the control room where support personnel may be stationed. The Wide Display Panel provides a fixed mimic display, a large (~100-inch diagonal) variable display. Spatially dedicated alarm windows for critical, plant-level alarms are also provided on the left-hand side Wide Display Panel. Spatially-dedicated detailed system level alarms are located above their respective systems on the fixed-mimic display. At the base of the Wide Display Panel, there are multiple flat display devices for individual system surveillance, monitoring and control.

The fixed mimic display is arranged on two, adjacent, upright panels which comprise the center and right-hand sections of the Wide Display Panel. The two panels are driven by independent microprocessor-based controllers. The center panel is seismically qualified and is driven by safety-related, Class 1E microprocessors. Information on this panel includes the critical plant parameters required for a safety parameter display system and Type A post-accident monitoring indications. 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. The right panel of the fixed 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 fixed 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.

The large variable display is located on the right upright panel of the Wide Display Panel. The basic purpose of the large variable display is to provide information on important plant process parameters which supplements the overview information on the fixed mimic display. The information presented on the large variable display can be changed, depending on the plant operating conditions and the needs of the operating crew. Any display format available on the MCC CRTs 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 CRT display formats which would be selected under plant emergency conditions.

Closed circuit TVs are provided which allow 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. 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.

The touch-control flat displays located at the base of the Wide Display Panel 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. These flat displays are driven by microprocessor-based controllers which are separate from the plant Process Computer System.

Plant Automation

The ABWR 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 ABWR provide for enhanced operability and improved capacity factor relative to conventional BWR designs. However, the extent of automation implemented in the ABWR 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 ABWR automation design provides for three distinct automation modes: Automatic, Semi-Automatic, and Manual. In Automatic mode, the operator initiates automated sequences of operation from the MCC. Periodic breakpoints are inserted in the automated sequence which 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 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 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.

Operation

The ABWR 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. One-man operation is possible due to implementation of several key design features: (1) the Wide Display Panel for overall plant monitoring, (2) plant-level automation, (3) system-level automation, (4) the compact MCC design, and (5) implementation of operator guidance functions which display appropriate operating sequences on the main control panel CRTs. 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 ABWR control room design is based upon having two operators normally stationed at the 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 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 touch-screen CRTs and flat displays.

Chapter 8
Plant Layout and Arrangement

Plant Layout

The ABWR Plant includes all buildings which are dedicated to housing systems and the equipment related to the nuclear system or controls access to this equipment and systems. There are five buildings within the scope:

- Reactor Building – includes the reactor pressure vessel, containment, and major portions of the nuclear steam supply system, refueling area, diesel generators, essential power, nonessential power, emergency core cooling systems, HVAC and supporting systems.
- Service Building – personnel facilities, security offices, and health physics station.
- Control Building – includes the control room, the computer facility, Reactor Building component cooling water system and the control room HVAC system.
- Turbine Building – houses all equipment associated with the main turbine generator. Other auxiliary equipment is also located in this building.
- Radwaste Building – houses all equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.

Development of the ABWR plant and building arrangements is 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 open top 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.

The site plan of the ABWR includes the Reactor, Turbine, Control, Radwaste, Service and supporting buildings. Provision is made within the Reactor Building for at least ten years spent fuel storage. Separate buildings can be provided for additional onsite waste storage. Figure 8-1 illustrates the site plan of the ABWR for a single unit arrangement.

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. The artist's rendering in Figure 8-2 provides a view of a typical two-unit site (Lungmen).

The ABWR 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.

No.	FACILITY
1	REACTOR CONTAINMENT
2	REACTOR BUILDING
3	CONTROL BUILDING
4	MAIN STEAM/FEEDWATER TUNNEL
5	TURBINE BUILDING
6	SERVICE BUILDING
7	RADWASTE BUILDING
8	HOUSE BOILER
9	CONDENSATE STORAGE TANK
10	UNIT AUXILIARY TRANSFORMERS
11	NORMAL SWITCHGEAR
12	DIESEL OIL STORAGE TANK (3)
13	STACK
14	EQUIPMENT ENTRY LOCK
15	FIRE PROTECTION WATER STORAGE TANK (2)
16	FIRE PROTECTION PUMPHOUSE
17	BUNKER FUEL TANK
18	COMBUSTION TURBINE GENERATOR
19	RADWASTE TUNNELS RB, CB, TB
20	DG OIL TRANSFER TUNNEL (3)

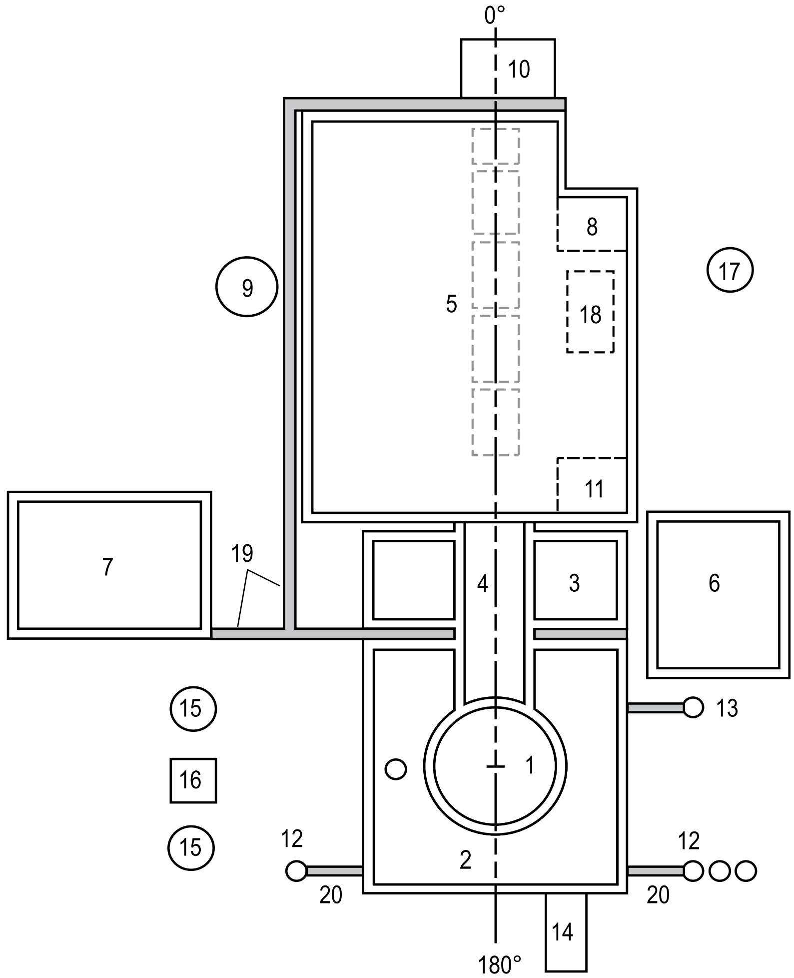


Figure 8-1. ABWR Site Plan

A three-dimensional perspective of the Reactor, Control and Turbine Buildings is shown in Figure 8-3.

The layout of the Reactor, Control and Turbine Buildings was based on the following considerations (for United States standardized design).

- 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 cross-over to the other area without returning to the

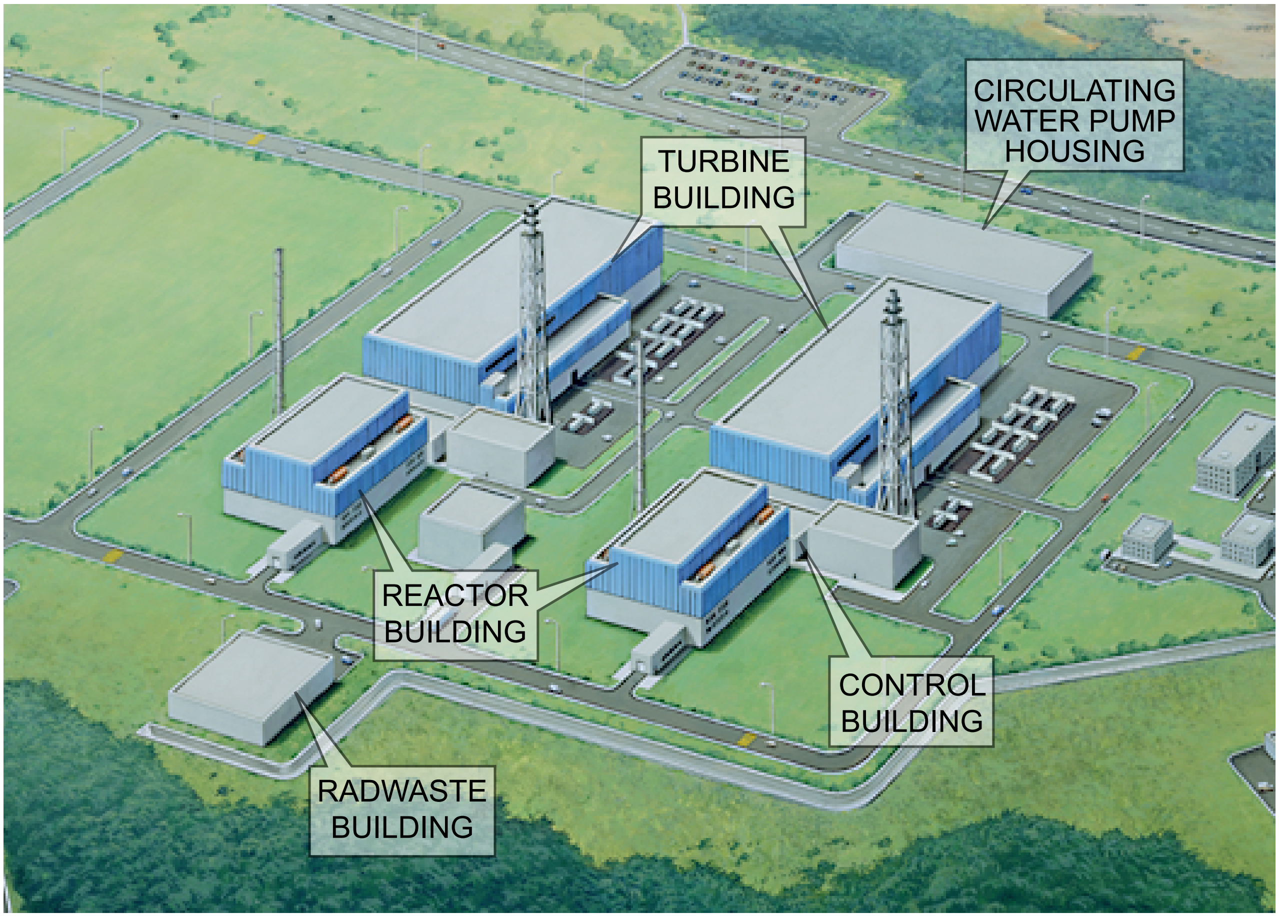


Figure 8-2. Conceptual picture of Lungmen Site in Taiwan

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 can be aligned with its axis in-line with the Reactor Building. This can be done to minimize the possibility of turbine missile impact on the containment vessel.
- 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.

Reactor Building

The Reactor Building houses the containment, drywell, and major portions of the nuclear steam supply system, steam tunnel, refueling area, diesel generators, essential power, nonessential power, Emergency Core Cooling Systems, HVAC system, and other supporting systems.

The ABWR Reactor Building is a reinforced concrete structure. The integrated Reactor Building and containment structure has been analyzed and licensed for a safe shutdown earthquake (SSE) of 0.3g for an “all-soils” site envelope. The Lungmen design is for 0.4g SSE.

A secondary containment surrounds the primary containment and provides a secondary containment function, including a Standby Gas Treatment System (SGTS).

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 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.

Removal of the post-LOCA decay heat is achieved by the Containment Heat Removal System, consisting of the suppression pool cooling mode, wetwell, and drywell spray features. 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 accidents, based on an assumed containment leak rate of 0.5% per day, show that the offsite doses after an accident is less than 1 rem. This favorable dose rate is made possible by trapping fission products within the secondary containment with a slight negative pressure and processing the air through the SGTS.

Key distinguishing features of the ABWR Reactor Building design include:

- Elimination of external recirculation loops 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 volume of the ABWR Reactor Building is reduced to approximately 167,000 cubic meters. 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.

The major equipment access to the Reactor Building is via a double door vestibule at grade level. This entry area is connected to the refueling floor by a large hatch serviced by the Reactor Building crane. 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.

Figure 8-4 shows an elevation view of the Reactor building. The major features and significant equipment are shown in this view. Grade elevation is at about the same elevation as the floor slab where the Main Steam and Feedwater lines are shown. Access to the lower drywell is by means of the locks shown over the suppression pool region.

Figure 8-5 shows a plan view of the reactor building at the refueling floor elevation. Shown here is the laydown arrangement of the large parts that are removed the RPV and its surrounding areas during a maintenance or refueling outage. This laydown assumes concrete plugs are used over the drywell head in the reactor cavity; even more space would be available if water shielding were used in the reactor cavity.

Figure 8-6 shows a plan view at the basemat elevation. Four safety division quadrants are seen in this figure. On the top right quadrant is Division A; Division C is in the lower right quadrant and Division B is in the lower left quadrant. The fourth

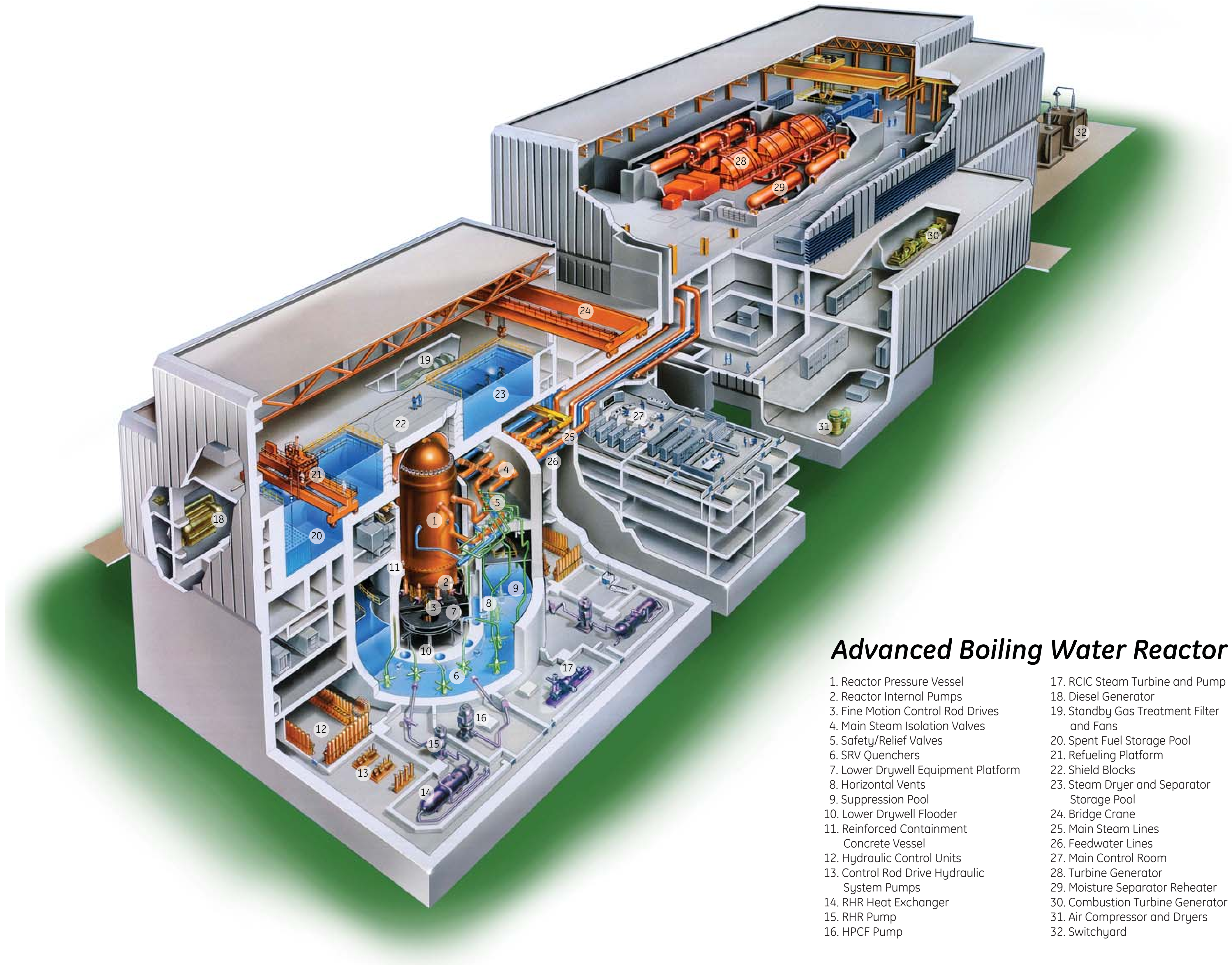


Figure 8-3. ABWR Cutaway View of the Reactor, Control and Turbine Buildings

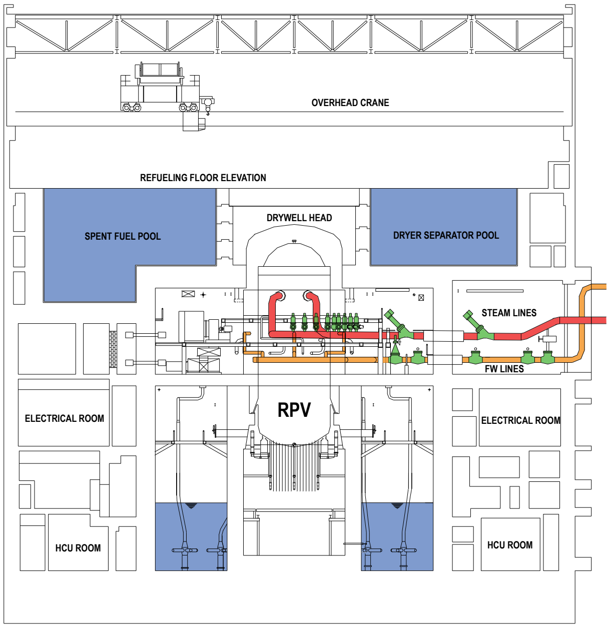


Figure 8-4. ABWR Reactor Building 0-180° Elevation

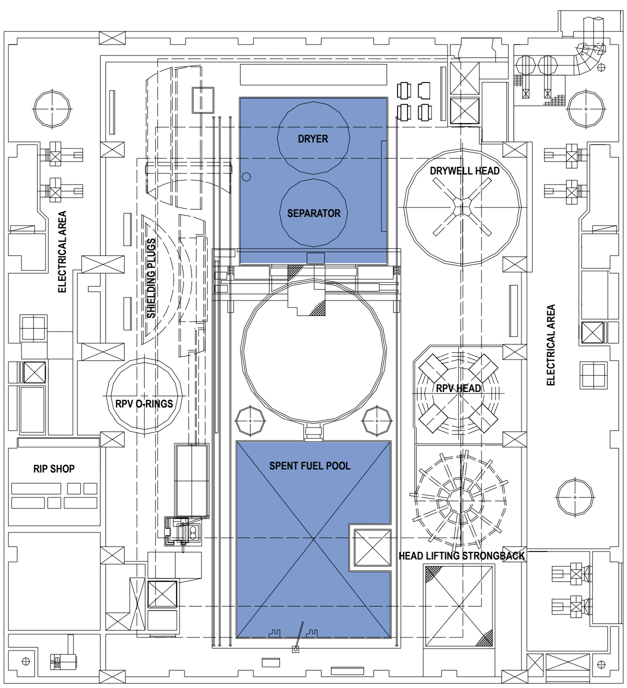


Figure 8-5. ABWR Reactor Building Refueling Floor Elevation

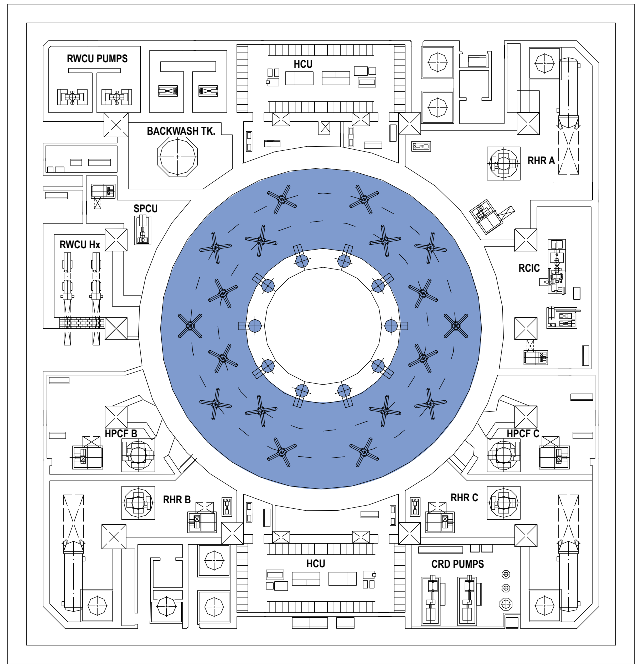


Figure 8-6. ABWR Reactor Building Basemat Elevation

division is electrical only and is located in the upper left quadrant. In the center of the picture is a plan view of the drywell and wetwell showing the pressure suppression vent system and the submerged quenchers that channel steam passed though the SRVs to the suppression pool

Primary Containment System

The ABWR pressure suppression primary containment system, which comprises the drywell (DW), wetwell (WW), and supporting systems, is designed to have the following functional capabilities (see Figure 8-7):

- 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

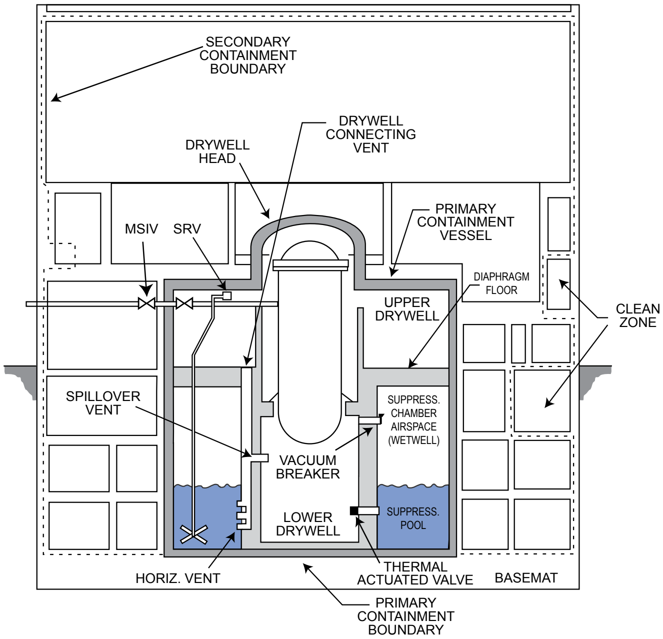


Figure 8-7. ABWR Containment Features

full range of loading conditions consistent with normal plant operating and accident conditions, including LOCA related loads in and above the suppression pool (SP). The containment structure is designed to accommodate the negative pressure difference between the DW and WW and the Reactor Building.

- 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 DBA to values less than leakage rates that could result in offsite radiation doses greater than those set forth in 10CFR100.
- 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 DW is comprised of two volumes:

- An upper drywell (UD) volume surrounding the reactor pressure vessel (RPV) and housing the steam and feedwater lines and other connections of the reactor primary coolant system, safety/relief valves (SRVs) and the drywell HVAC coolers.
- A lower drywell (LD) volume housing the reactor internal pumps, fine motion control rod drives (FMCRDs) and under vessel components and servicing equipment. The UD is a cylindrical, reinforced concrete structure with a removable steel head and a reinforced concrete diaphragm floor. The cylindrical RPV pedestal, which is connected rigidly to the diaphragm floor, separates the LD from the wetwell. It is a prefabricated steel structure filled with concrete after erection. Ten drywell connecting vents (DCVs) are built into the RPV pedestal and connect the UD and LD. The DCVs are extended downward via steel pipes, each of which has three horizontal vent outlets into the suppression pool.

Wetwell Structure

The WW is comprised of an air volume and a 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 wetwell boundary is a cylindrical reinforced concrete wall which is continuous with the UD boundary. A reinforced concrete mat foundation supports the entire containment system and enclosed structures, systems and components, and extends to support the Reactor Building surrounding the containment.

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 0.5% per day from all sources, excluding main steam isolation valve

(MSIV) leakage.

Containment System

The drywell is designed to withstand the pressure and temperature transients associated with the rupture of any primary system pipe inside the drywell and also the rapid reversal in pressure when the steam in the drywell is condensed by the Emergency Core Cooling System (ECCS) flow following post-LOCA flooding of the RPV.

A redundant vacuum breaker system has been provided between the drywell and wetwell. The purpose of the wetwell-to-drywell vacuum relief system is to prevent backflooding of the suppression pool water into the lower drywell and to protect the integrity of the diaphragm floor slab between the drywell and wetwell, and the drywell structure and liner.

In the event of a pipe break within the drywell, the increased pressure inside the drywell forces a mixture of air, steam and water through the drywell connecting vents (DCVs), and horizontal vents into the suppression pool, where the steam is rapidly condensed. The noncondensable gases transported with the steam escape to and are contained in the free air volume of the wetwell. There is sufficient water volume in the suppression pool to provide submergence of the upper row of horizontal vents when water is removed from the pool during post-LOCA drawdown by the ECCS. This drawdown floods the RPV to the steamlines, floods the lower drywell to its drain to the DCV, and provides for water in transit from the break on its gravity drain back to the suppression pool. The design pressure of the containment is 45 psig (3.16 kg/cm^{2}g).

During isolation transients, when the MSIVs close, the SRVs discharge steam from the relief valves through their exhaust piping and quenchers into the suppression pool.

The suppression pool is sized to accommodate the stored energy within the RPV during a LOCA without exceeding its design temperature. During isolation transients there are many hours of decay heat absorption storage capability.

For beyond design basis events, piping with temperature actuated valves connect the DCV with the lower drywell. This provides a passive flooding capability.

Reinforced Concrete Containment Vessel Description

For the ABWR plant, a circular cylindrical reinforced concrete containment vessel (RCCV) with a top slab and a monolithically connected diaphragm slab has been selected. The RCCV is integrated with the Reactor Building and forms a major structural part of this building. The flat annular top slab of the drywell is integrated with the fuel pool girders and the dryer/separator cavity floors which are framed into the Reactor Building structural walls and floors. The cylindrical design with its simple geometry allows easy and fast construction. The pressure-retaining concrete walls of the RCCV are lined with leak tight steel plates and form the boundary of the primary containment system.

The RCCV supports the internal drywell structures. Major components framed to the RCCV are the equipment hatch and personnel locks. It also serves as an anchor point for the Main Steam and Feedwater lines.

The structural framing scheme mentioned earlier takes advantage of both the RCCV and Reactor Building to carry dynamic and shear loads and, hence, reduce overall size and thickness of the supporting walls. The Reactor Building has been separated into four quadrants to provide separation for the three-division configuration of the safety systems plus one quadrant for non safety systems.

The RPV is supported by the reactor pedestal and is concentric within the RCCV (see Figure 8-8). The pedestal has multiple functions. It also supports the drywell diaphragm floor, the lower drywell access tunnels, the horizontal vent system and communicating vents between the lower and upper drywells. The composite steel-concrete design allows fabrication in the shop applying modular con-

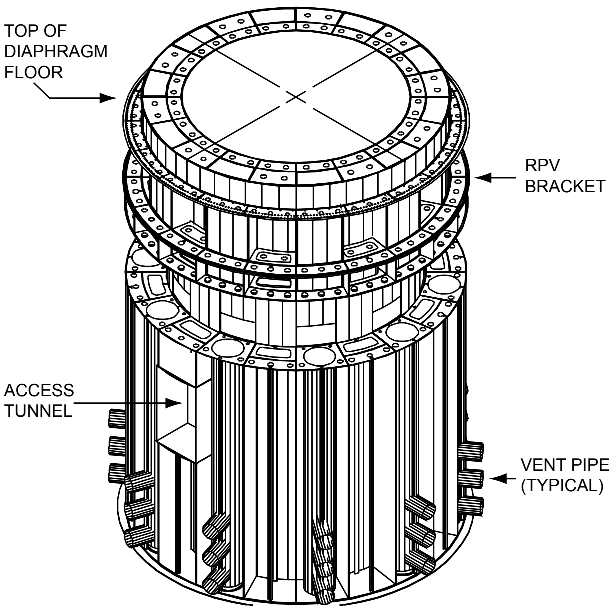


Figure 8--8. ABWR Reactor Pedestal

struction techniques, and erecting it in one piece.

Fire Protection

The basic layout of the plant and the choice of systems to mitigate the effects of fire enhance the resistance of the ABWR plant to fire. The safety-related systems are designed such that there are three independent divisions, any one of which is capable of providing safe shutdown of the reactor. In addition, there are non-safety-related systems such as the Condensate and Feedwater Systems which can be used to achieve safe shutdown. The plant arrangement is such that points of possible common cause failure between non-safety-related systems and safety-related systems have been eliminated.

Plant Arrangement

The plant is laid out with the Control Building between the Reactor and Turbine Buildings so that power and control signals from the Reactor and Turbine Buildings enter the Control Building on opposite sides of the Control Building. This arrangement ensures that a potentially damaging fire in the Turbine Building will not disable non-safety-related systems capable of providing safe shutdown in both the Turbine and Reactor Buildings.

Preferred power is supplied through the Turbine Building to the Reactor Building for the safety-related loads. These non-safety-related power sources are backed up by the safety-related diesel generators, which are not affected by events in the Turbine Building.

Divisional Separation

There are three complete divisions of cooling systems, any one of which is capable of safe or emergency shutdown of the plant. 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 two other divisions available. Each division is served by its own HVAC equipment from within that division.

The remote shutdown panel provides 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 also have a rating of three hours. Electrical and piping penetrations through a fire barrier have seals with a three-hour rating.

There are two firewater pumps in the plant, one motor driven and one diesel driven. Each of these meets requirements for flow and pressure demand at the most hydraulically remote hose connection in the plant. Sprinkler systems in the Reactor Building and sprinkler and wet standpipe systems in the Control Building are designed to remain functional following a Safe Shutdown Earthquake (SSE).

Flood Protection

The ABWR 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 no penetration of tunnels below grade through exterior walls.

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.

Other Buildings

Turbine Building

The Turbine Building 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. The small size of the ABWR Turbine Building makes a significant contribution to capital cost savings and a shorter construction schedule.

Control Building

The Control Building includes the control room, the computer facility, the cable tunnels, some of the plant essential switchgear, some of the essential power, the Reactor Building water system, and the essential HVAC system.

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. For system descriptions, see Chapter 5.

Service Building

In the Service Building are the following facilities: Administration; Access Control and Security to the Reactor building; Health Physics Laboratory; and Change Rooms.

The principal function besides housing the administrative operations of the plant is to control traffic into and out of the Reactor Building complex.

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 ultimate heat sink location and temperature 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 1350 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 system, main condenser, condenser evacuation system, turbine gland seal system, turbine bypass system, extraction steam system, condensate cleanup 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 pumped back to the reactor feedwater pump suction by the heater drain pumps. 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, SJAE condensers, offgas recombiner condensers to the condensate boost pumps. The condensate boost pumps deliver the feedwater through the low-pressure feedwater heaters to the reactor feed pumps. The reactor feed pumps 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 ABWR plant features and parameters are reported herein based on typical central U.S. site conditions.

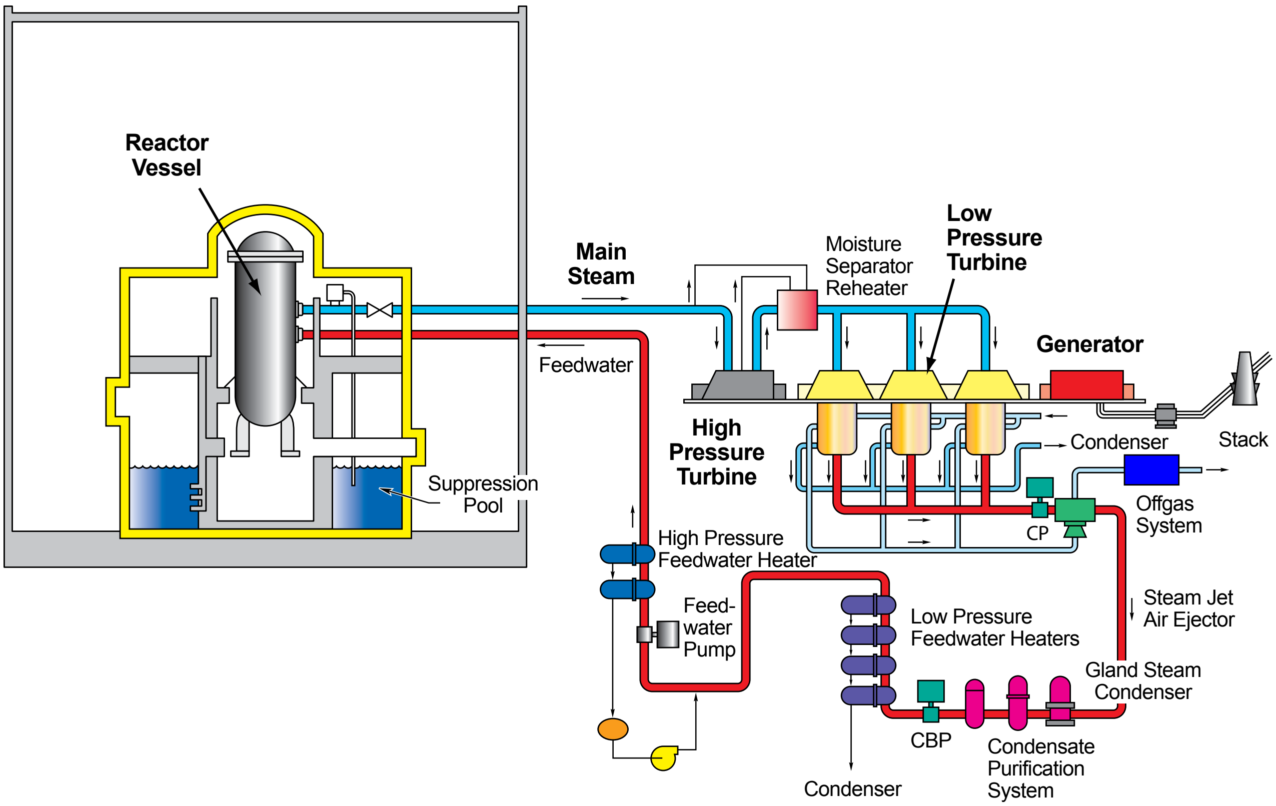


Figure 9-1. Reference Steam and Power Conversion System

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 33% of the rated steam flow is provided to discharge excess steam directly to the condenser. Although the ABWR Standard Plant design is for 33% bypass, this capability could be increased to a full load reject capability without affecting the Nuclear Island (as has been done for Lungmen).

Turbine Main Steam Systems

The Turbine Main Steam System delivers steam from the reactor to the turbine generator, the reheaters, the turbine bypass system, and the steam jet air ejectors (SJAEs) from warm-up 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.

Main Turbine/Generator

The main turbine is an 1800 rpm, tandem compound six flow, reheat steam turbine. 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 1350 MWe. For utilities generating 50 Hz power, the turbine shaft speed is 1500 rpm.

Main Condenser

The Main Condenser is a multi-pressure three-shell type deaerating type condenser. 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 noncondensable gases from the power cycle. The MCES removes the hydrogen and oxygen produced by radiolysis of water in the reactor, and other power cycle noncondensable 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.

The MCES consists of two 100% capacity, double stage, SJAE units (complete with intercondenser) for power plant operation where one SJAE unit is normally in operation and the other is on standby, as well as a mechanical vacuum pump for use during startup. The last stage of the SJAE is a noncondensing stage.

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 pump establishes a vacuum in the Main Condenser and other parts of the power cycle. The discharge from the vacuum pump is then routed to the Turbine Building compartment exhaust system, since there is then little or no effluent radioactivity present. Radiation detectors in the Turbine Building compartment exhaust system and plant vent alarm in the Main Control Room (MCR) if abnormal radioactivity is detected. Radiation monitors are provided on the main steamlines 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 pressure of about 0.034 to 0.051 MPa absolute is established in the Main Condenser by the mechanical vacuum pump and when sufficient nuclear steam pressure is available.

During normal power operations, the SJAEs are normally driven by cross-around steam, with the main steam supply 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, should the mechanical vacuum pump prove to be unavailable.

Turbine Gland Steam System

The Turbine Gland Steam System provides steam to the turbine glands and the turbine valve stems. The Turbine Gland Steam System 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 a motor-driven blower.

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 a three-valve chest that is connected to the main steamlines upstream of the turbine stop valves, and of three dump lines that separately connect each bypass valve outlet to one condenser shell. The system is designed to bypass at least 33% of the rated main steam flow directly to the condenser. The TBS, in combination with the reactor systems, provides the capability to shed 40% of the T-G rated load without reactor trip and without the operation of SRVs. A load rejection in excess of 40% is expected to result in reactor trip but without operation of any steam safety valve. Optionally, some utilities desire 100% bypass (full load rejection) capability. In this case there are more bypass valves required.

The turbine bypass valves are opened by 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 servo valves to allow rapid opening of bypass valves upon turbine trip or generator load rejection.

The bypass valves automatically trip closed whenever the vacuum in the main condenser falls below a preset value. The bypass valves are also closed on loss of electrical power or hydraulic system pressure. The bypass valve hydraulic accumulators have the capability to stroke the valves at least three times should the hydraulic power unit fail.

When the reactor is operating in the automatic load-following mode, a 10% load reduction can be accommodated without opening the bypass valves, and a 25% load reduction can be accommodated with momentary opening of the bypass valves. These load changes are accomplished by change in reactor recirculating flow without any control rod motion.

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 last stage of feedwater heating and extraction steam from the low pressure turbines supplies the first four stages. An additional low pressure extraction drained directly to the condenser protects the last-stage buckets from erosion induced by water droplets.

Condensate and Feedwater System

The Condensate and Feedwater System (CFS) provides a dependable supply of high-quality feedwater to the reactor at the required flow, pressure, and temperature. The condensate pumps take the deaerated condensate from the main condenser hotwell and deliver it through the SJAE condenser, the gland steam condenser, the offgas condenser, the condensate demineralizer to the inlet of the condensate boost pumps. The condensate boost pumps deliver the feedwater through three parallel strings of four low pressure feedwater heaters to the reactor main feedwater pumps (FWP) section. The FWPs each discharge through two stages of high pressure heaters (two parallel strings) to the reactor. Each reactor FWP is driven by an adjustable speed synchronous motor. The drains from the high pressure heaters are pumped into the suction of the FWPs.

For a description of the feedwater system on the Nuclear Island side, see Chapter 3.

Moisture Separator Reheater System

Four horizontal cylindrical-shell, combined moisture separator/reheaters are installed in the steam path between the high and low pressure turbines. The moisture separator/reheaters 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 heater drain tank. The dry steam passes upward across the heater which is supplied with main steam. Finally, the reheated steam is routed to the combined intermediate valves which are located upstream of the low pressure turbines' inlet nozzles.

Circulating Water System

The Circulating Water System (CWS), 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 power cycle heat sink. The CWS consists of the following components: (1) screen house and intake screens, pumps; (2) condenser water boxes and piping and valves; (3) tube side of the main condenser; (4) water box fill and drain subsystem; and (5) related support facilities such as for system water treatment, inventory blowdown and general maintenance.

To circulate the cooling water, there are at least three fixed speed motor-driven pumps that are arranged in parallel and discharge into a common header. This arrangement permits isolation and maintenance of any one pump while the others remain in operation.

Other Turbine Auxiliary Systems

Turbine Building Service Water System

The Turbine Building Service Water (TSW) System provides cooling water from the power cycle heat sink to the cold water side of the TCW heat exchangers. The TSW System consists of pumps, motor-operated valves, strainers, piping and instrumentation.

Turbine Building Cooling Water System

The Turbine Building Cooling Water (TCW) System is a closed-loop cooling water system that supplies cooling water through the TCW heat exchangers to Turbine Island equipment coolers and rejects heat to the TSW System.

Station Electrical Power

AC Power Distribution

The AC electrical power distribution consists of three independent load groups (Figure 9-2). Each group receives preferred power from its own unit auxiliary transformer (UAT) which, in turn, derives its power from the main generator during plant operation. The main generator has its own circuit breaker which allows power to be back-fed through the main transformer to supply the auxiliaries when the plant is off-line. This also facilitates the startup of the plant without the need for startup transformers.

Each load group supplies power to plant power generation (PG) loads, plant investment protection (PIP) loads, and safety-related (Class 1E) loads through separate buses and associated distribution systems. Alternate preferred power can be supplied through two reserve auxiliary transformers (RATs).

Three dedicated Class 1E emergency diesel generators (DGs) supply automatic backup power to each of the three safety-related divisions. Each DG is capable of providing the required power to safely shut down the reactor after loss of preferred power and/or a loss-of-coolant accident (LOCA), to maintain the safe shutdown condition, and to operate the Class 1E auxiliaries necessary for plant safety after shutdown. The diesel generators are now located in the Reactor Building.

In a similar manner, a non-Class 1E combustion turbine generator (CTG) supplies automatic backup power to selected PIP loads. In the event of station blackout (i.e., loss of all AC power including DGs), the CTG can be manually connected to any of the Class 1E buses.

Each of the three UATs has dual secondary windings. One winding provides 13.8 kV to the PG buses and the other winding provides 4.16 kV to the PIP and Class 1E buses. Each 4.16 kV Class 1E bus feeds its associated Class 1E 480 V unit substations through 4.16 kV/480 V/277 V power center transformers. Power for the non-Class 1E 480 V auxiliaries is supplied from power centers consisting of either 13.8 kV/480 V/277 V or 4.16 kV/480 V/277 V transformers and associated metal-clad switchgear. To the extent practical, power center transformers are uniformly sized (i.e., the same kVA rating) to improve voltage regulation, reduce costs, and achieve standardization.

One of the two RATs, which provide alternate preferred power to the electrical distribution system, has a 13.8 kV secondary winding which feeds the six PG buses. The other RAT has two 4.16 kV secondary windings with one winding providing power to the three PIP buses, and the other providing power directly to a stub bus which, in turn, provides power through individual circuit breakers to the three Class 1E buses.

The Division I DG provides its output at 4.16 kV to the Class 1E Division I bus. Likewise, the Division II DG provides its output at 4.16 kV to the Class 1E Division II bus, and the Division III DG provides its output at 4.16 kV to the Class 1E Division III bus. The Class 1E divisions are independent and separated, and there are no automatic inter-connections between them.

The CTG provides its output at 13.8 kV to the CTG bus which:

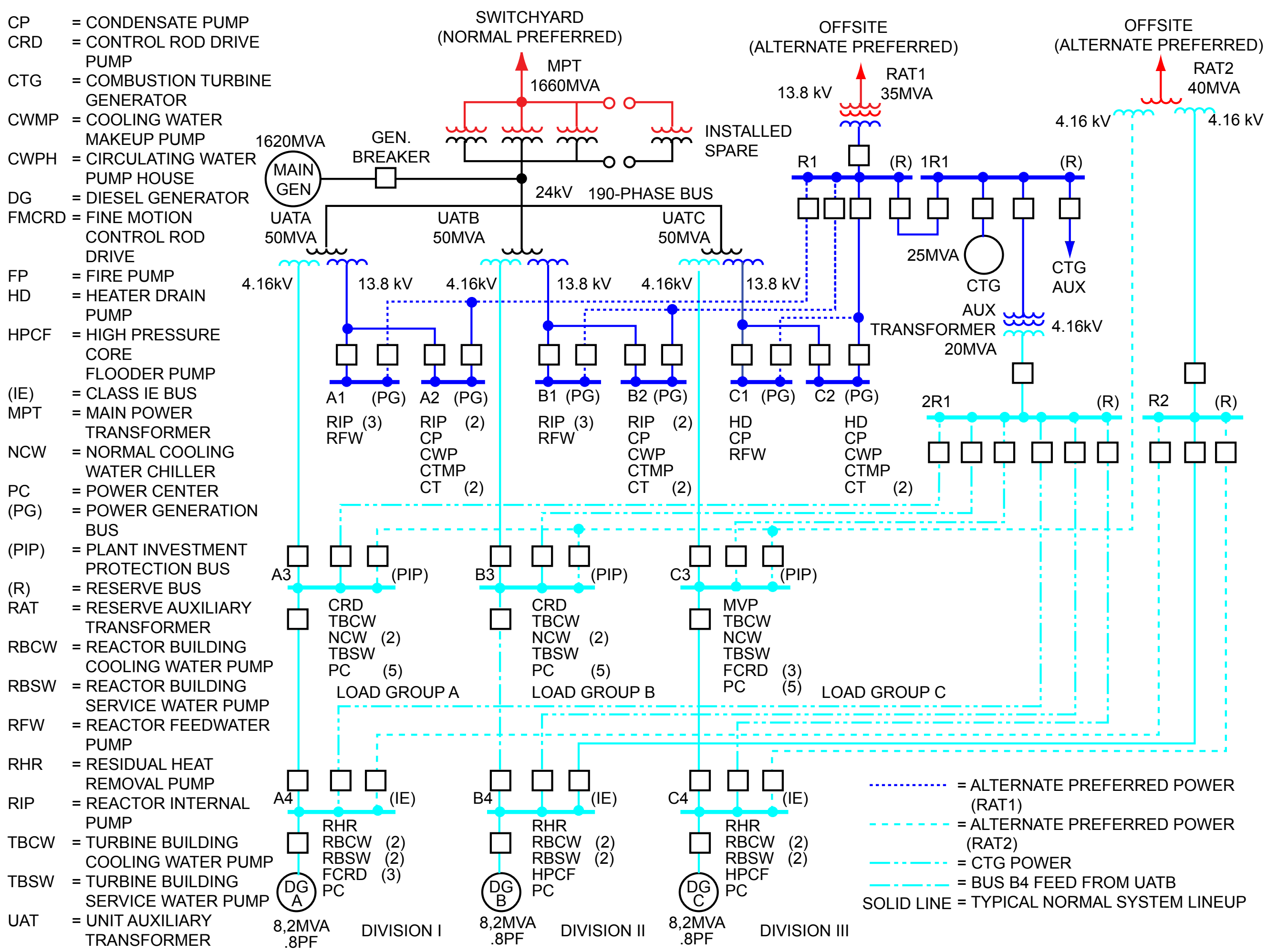


Figure 9-2. Electrical Power Distribution System

- Provides power to a 4.16 kV CTG bus through a transformer. The 4.16 kV CTG bus then provides power directly to the three PIP and three Class 1E buses, as needed.
- Provides power to the 480 V CTG auxiliary bus through a transformer.
- • Contains a bus tie (through a circuit breaker) to the 13.8 kV RAT output bus in order to permit the CTG to start and run a condensate pump, cooling tower fans, or other selected PG loads, if needed.

Should a loss of preferred power occur, each Class 1E bus automatically transfers to its own DG, and selected PIP loads automatically transfer to the CTG. However, the interconnection capability of the ABWR is such that any plant loads can be manually connected to receive power from any of the six sources (i.e., the two switching stations, the CTG, and the three DGs).

DC Power Distribution

The DC Power Supply (DCPS) consists of three separate subsystems:

- Safety-related 125 VDC (four independent and separated Class 1E divisions)
- Non-safety-related 125 VDC (three individual load groups)
- Non-safety-related 250 VDC (two individual load groups)

Each DC subsystem consists of a battery and associated charger for each division or load group, power distribution panels, and all the associated control, monitoring and protective equipment and interconnecting cabling. In addition, the DCPS employs standby chargers that are shared between the batteries to enable the individual battery testing and off-line equalization.

The DCPS operates with its battery and battery chargers (except standby chargers) continuously connected to the DC system. During normal operation, the DC loads are powered from the chargers with the batteries receiving a continuous charging current (i.e., floating) on the system. The chargers are powered from the 480 VAC supply of the same division or load group, with exception of the Division IV charger, which is powered from the Division II 480 VAC supply. In case of loss of AC power to the charger or its failure, the DC loads are automatically assumed by the battery of each respective division or load group.

The DCPS also facilitates uninterruptable regulated AC power through inverters in the Vital AC (VAC) Power Supply. The VAC supplies constant uninterrupted AC power to those loads which require continuity of AC power during a loss of preferred power event. Each Class 1E division and non-Class 1E load group has its own individual uninterruptable power supply (UPS) unit.

Chapter 10
Safety Evaluations

Licensing Framework

The ABWR licensing basis generally follows the USNRC regulations and regulatory guides. This framework is the model which has been adopted in a number of countries, including Japan and Taiwan (which have also approved ABWR). In particular, Regulatory Guide 1.70, Rev 3, is used as the basis for documenting the safety analyses and the Standard Review Plan (SRP) is used by the USNRC in reviewing the design.

The events to be analyzed within the design basis are only loosely coupled to a probabilistic basis, and can be considered to be deterministic, both in the event description and in the acceptance criteria. Key points of this policy are:

- The assumption that Anticipated Operational Occurrences (transients) are caused by a single equipment failure or single operator error
- Use of the single active failure (N-1) criterion for postulated accidents
- Certain prescribed hardware and analyses to mitigate special events, such as Anticipated Transients Without Scram (ATWS)

There are no specific US regulations governing severe accidents, but a special set of review guidelines for severe accidents was promulgated by the USNRC in the form of several documents (SECY 90-016, SECY 93-087). These formed the basis for the safety review of the severe accident features provided in ABWR. The ABWR is the first plant to have undergone a specific review for severe accident mitigation and received approval from the USNRC.

Safety Design Approach

The ABWR safety design approach was to provide safety features which meet the regulations, but also to use PRA techniques to guide the feature selection. Insights gained from the PRAs were also used to improve plant technical specifications, emergency procedure guidelines, the control room interface, and 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.

The ABWR safety design approach looked at four key areas for safety improvement:

- Operational transients
- Design basis accidents
- Special regulatory mandated events
- Beyond design basis accidents, including core melt, commonly called severe accidents

Since improved operations lead directly to improved safety through reduced challenges to safety systems, the ABWR introduced a number of improvements in this area:

- Increased redundancy and diagnostics for I&C control and protection systems so that single failures do not lead to scrams
- The use of 99% trip avoidance statistical design for separating nominal and analytical instrument set points
- Increased redundancy in key hardware, such as

recirculation and feedwater pumps

PRA techniques were used to guide the ECCS architecture. For example, significant improvement in core damage frequency (CDF) was obtained in going from 2 divisions to 3 divisions. However studies showed little value in going to 4 divisions, due to common mode failure considerations. Therefore, for design basis accident mitigation the ECCS architecture chose a 3-division approach with a high pressure and a low pressure flooder in each division and heat removal in each division. Each division also contains all supporting services, such as emergency diesel generators. Although it is conceptually possible to meet a deterministic N-2 failure criterion with 3 divisions, PRA studies indicated station blackout to be the dominant safety threat. Therefore, in one division the high pressure flooder was made steam driven. In practice, the ABWR is nearly N-2, with only one out of hundreds of possible N-2 combinations not able to provide core safety.

Containment design for the ABWR uses a double barrier approach, with a low leakage inerted pressure suppression primary containment pressure boundary surrounded by a slightly negative pressure processed atmosphere secondary containment. The US NRC granted regulatory credit for holdup of containment leakage past the MSIVs in the steamlines and main condenser in offsite dose evaluations. This change to past design practice was made to gain consistency in approach between ABWR and current practices of US operating BWRs.

For external threats, such as seismic events, the ABWR is designed to a site envelope (see Appendix A) conservative enough to cover at least 90% of the potential sites in the USA. In this way, a utility can compare its specific site to the envelope, and if the site is within the analyzed conditions, no further safety analysis will be required.

For the key new features, particular attention was paid to insure that their introduction did not lead to new safety issues. For the recirculation system design, redundant power supplies are used and M-G sets with flywheels are included to insure a slow flow coastdown in case of loss of power. The all recirculation pumps trip event is reclassified as an infrequent operational occurrence, with time-temperature acceptance criteria instead of MCPR. The justification for this change is the low probability of simultaneous loss of power at all 10 RIPs. The pump attachments to the reactor vessel were designed so as to make impossible a pump shaft blowout in case of a pressure boundary failure in a pump housing.

For the FMCRD, the drives are supported from the core plate to avoid a drive ejection on pressure boundary failure. In addition, an electromechanical brake which is engaged at all times except during rod movement was added to avoid rod runout on pressure boundary failure. Safety grade limit switches are included in the design to detect rod separation from the drive - this eliminates the possibility of a reactivity event from a rod drop accident. Finally, diverse C&I systems to monitor rod movement eliminated rod withdrawal error transients.

For all design basis events the response was designed to be fully automatic, with no operator action needed to accomplish any of the safety functions - reactor trip, core cooling, containment isolation, heat removal and radiation protection of the public.

Certain special regulatory mandated events were also addressed directly in the design. ATWS mitigation is completely automated and includes the following features:

- Trip of 4 out of 10 recirculation pumps on high reactor pressure
- Diverse C&I scram (Alternate Rod Insertion) signal for rapid hydraulic-driven rod insertion
- FMCRD electric run-in following a scram signal
- Recirculation flow runback to reduce power
- Feedwater pump trip after two minutes, if necessary, to further reduce power
- Liquid poison injection via the SLCS after 3 minutes, if power is still not down

Station Blackout mitigation includes (in addition to the steam driven RCIC) an alternate onsite diverse AC power source - a Combustion Turbine Generator (CTG) for the standard design. The CTG is a standby onsite non-safety power source designed to feed permanent non-safety loads during loss of offsite power (LOOP) events. Implementation at multiple unit sites has taken alternate approaches: swing EDG at Lungmen, emergency power sharing at Kashiwazaki-Kariwa.

The CTG can supply power to nuclear safety-related equipment if there is complete failure of the emergency diesel generators and all offsite power. Under this condition, the CTG can provide emergency backup power through manually-actuated Class-1E breakers in the same manner as the offsite power sources. This provides a diverse source of onsite AC power.

Since active systems form the basis for safety protection for transients, design basis accidents and special events (prevention), the ABWR has provided primarily passive severe accident mitigation features to protect the containment from over-pressurization and to limit the consequences to the public even if the pressure were to exceed the design pressure. The philosophy behind this approach is that the only way a severe accident could occur is by complete failure of the active systems.

Design Basis Transient and Accident 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., 8x8, 9x9 or 10x10); however, the ABWR reactor was designed to assure flexibility of use of advancing fuel technologies while maintaining the capability of at least 15% operating margins to fuel limits if desired by the utility, as required by the US utility sponsored Utility Requirements Document. A wide variety of transient analyses have been performed with different fuel designs to demonstrate that the above requirement can be met, even with longer fuel operating cycles up to 2 years in length.

With the elimination of large pipes attached to the RPV below core elevation, the ABWR has no core uncovery even for the most limiting design basis loss of cooling accident (DBA LOCA). Thus there is no concern over peak fuel clad temperature (PCT) after DBAs. The peak pressure in the containment after DBA has been shown to have 15% margin relative to the design pressure of 0.31 MPa (45 psig). Thus the ABWR represents a very robust design for the traditional design basis accidents.

Severe Accident Mitigation

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 ABWR capability to prevent severe reactor accidents from occurring and the 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 process. The final evaluation indicates that events resulting in damage to the reactor core are extremely unlikely, but that even if such events were to occur, passive accident mitigation features would limit the offsite dose such that the effect on the public and surrounding land would be insignificant.

ABWR Probabilistic Risk Assessment (PRA)

A comparison of the internal events PRA for the ABWR to PRAs performed for other reactors clearly demonstrates the overall improvement in safety (see Figure 10-1). The US NRC risk goal for the frequency of core damage events in new reactors is one in 100 thousand years. The CDF for the ABWR was found to be less than 2 per ten million years. This represents more than a factor of 10 improvement as compared to the previous BWR designs, and a factor 100 improvement as compared to most light water reactors (see Table 10-1).

Probabilistic methods were also applied to events initiated externally: tornado, flood, fire and

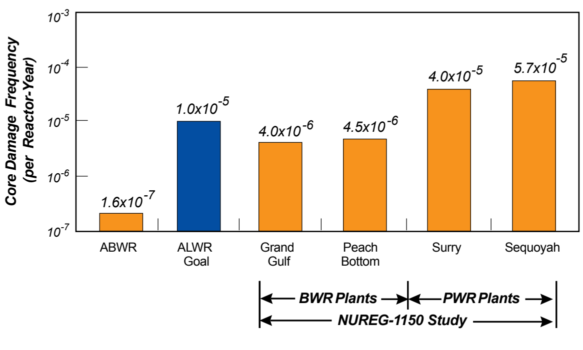


Figure 10-1. Comparison of CDF to Other Plants

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.

The frequency of core damage due to a tornado was found to be extremely low because all safety components are located in the concrete Reactor Building, and the three divisions of emergency diesel generators reduce the probability of a Station Blackout (SBO) due to failure of AC power.

The Reactor Building is designed to physically separate the safety divisions. In addition, drains in the upper floors of the Reactor Building were included in the design to direct water away from electrical equipment towards the lower floor of the building, where water can accumulate without damaging safety equipment. As a result, a flood cannot damage more than one safety division. Therefore, the core damage frequency associated with flooding was found to be a very small fraction of the NRC Safety Goal.

The evaluation of fires was based on the Fire-Induced Vulnerability Evaluation methodology developed by the Electric Power Research Institute

Table 10-1. ABWR Internal Events CDF

Event	Frequency/yr	%
Station Blackout	1.1×10^{-7}	71
Transients	4.5×10^{-8}	29
LOCA	6.9×10^{-9}	<1
ATWS	2.7×10^{-10}	<1
Total	1.6×10^{-7}	

(EPRI). This conservative methodology provides procedures for performing quantitative screening analyses of fire risk. No risk significant fires were identified for the ABWR design due in part to the complete divisional separation of all safety systems and the ability to initiate and control safety systems from the remote shutdown panel.

The risk of seismic events was evaluated using seismic margins method. The ABWR was designed for a Safe Shutdown Earthquake (SSE) of 0.3g for an envelope of possible soils. 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 ABWR was shown to have a factor of more than two margin to the design SSE. This ensures that there is very little possibility of a core damage event as a result of an earthquake.

The risk of core damage occurring during refueling and maintenance operations was also evaluated probabalistically and found to be a small fraction of the overall core damage risk. This is primarily due to the large body of water overlying the core during refueling operations.

ABWR Features to Mitigate Severe Accidents

As a result of PRA evaluations and other implied requirements in the USNRC severe accident review guidelines, several new features were added to the ABWR design. Five of the primary features are described; the first four of these are passive.

Inerted Containment

The ABWR containment is normally inerted with nitrogen containing < 3.5% oxygen (see discussion of the Atmospheric Control System in Chapter 4). Therefore any potential for hydrogen burning or detonation after a severe accident is avoided.

Lower Drywell Flooder

The Lower Drywell Flooder floods the lower drywell with water from the suppression pool during severe accidents where core melting and subsequent vessel failure occur. Several pipes run from the vertical pedestal vents into the lower drywell (see Figure 8-6). Each pipe contains a fusible plug valve connected by a flange to the end of the pipe that extends into the lower drywell. In the unlikely event that molten corium flows to the lower drywell floor and is not covered with water, the lower drywell atmosphere will rapidly heatup. The fusible plug valves open when the drywell atmosphere (and subsequently the fusible plug valve) temperature reaches 260°C. The fusible plug valve is mounted in the vertical position, with the fusible metal facing downward, to facilitate the opening of the valve when the fusible metal melting temperature is reached. When the fusible plug valves open, suppression pool water will be supplied through the system to the lower drywell to quench the corium, cover the corium and remove corium decay heat. The result will be a reduced drywell temperature and less pressure from non-condensable gas generation. There will be less chance of overpressurizing the containment and increasing leakage. The Lower Drywell Flooder is a passive injection system. No operator action is required.

Corium Shield

The bottom of the lower drywell contains two features to mitigate against continued core-concrete reactions after quenching by the passive lower drywell flooders. The first of these is the use of non-limestone aggregrate concrete (so-called basaltic concrete). This minimizes any further production of carbon-based noncondensables, such as CO and CO_2. In addition the drywell sumps are covered with refractory oxide bricks to prevent intrusion of molten corium into the sumps.

Containment Overpressure Protection System

If an accident occurs which 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 relieving the wetwell atmosphere to the plant stack. Providing a relief path from the wetwell airspace precludes an uncontrolled containment failure. Directing the flow to the stack provides a monitored, elevated release. The relief line, designed for ~ 1 MPag, contains two rupture disks, in series, which open at a pressure above the design pressure but below the Service Level C capability of the containment (see Figure 4-7). If overpressure occurs, the rupture disks will open; and pressure is relieved in a manner that forces escaping fission products to pass through the suppression pool. Relieving pressure from the wetwell, as opposed to the drywell, takes advantage of the decontamination factor provided by the suppression pool. After the containment pressure has been reduced and normal containment heat removal capability has been regained, the operator can close two normally open air-operated valves in the relief path to reestablish containment integrity. Initiation of the pressure relief system is totally passive. No power is required for initiation or operation of the pressure relief function for an indefinite period.

AC-Independent Water Addition

Two fire protection system pumps are provided on the ABWR: one pump is powered by AC power, the other is driven directly by a diesel engine. A fire truck can provide a backup water source. One of the fire protection standpipes is cross-connected to the RHR injection line to the reactor vessel through normally closed, manually operated valves. From this line, fire protection water can be directed to the reactor vessel after the reactor vessel has been depressurized. Fire protection water can also be directed to the drywell spray header to reduce upper drywell pressure and temperature. Figure 4-5 shows the piping arrangement.

An integrated view of how the passive severe accident mitigation features work together is shown in Figure 10-2. Analyses of the dominant severe accident sequence, which is a low pressure core melt following a Station Blackout, shows that the COPS pressure will not be reached for more than 24 hours. In addition, the conditional containment failure probability, defined as the loss of containment as a fission product barrier, was calculated to be 0.2%, far less than the goal of 10% set by the U.S. Utility Requirements Document and USNRC guidelines.

Summary

For a nominal U.S. site, the offsite dose as a function of probability is given in Figure 10-3. It can be seen that large releases do not occur even at the one in a billion probability level.

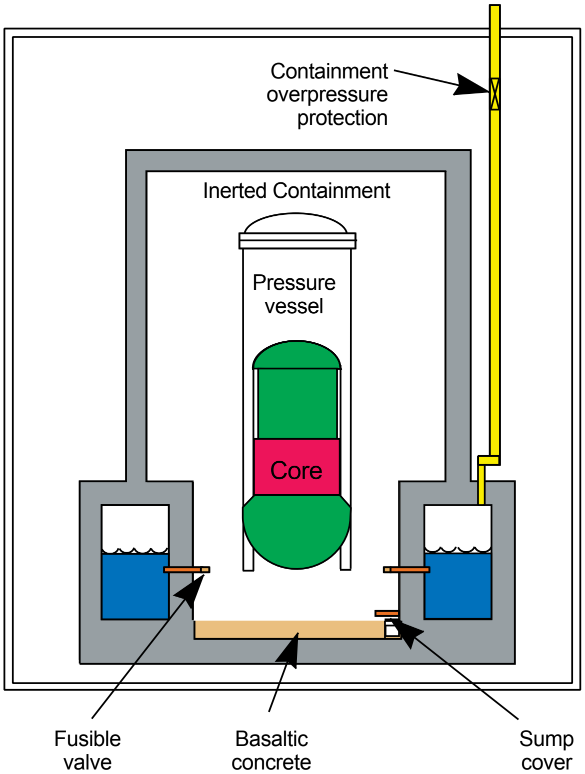


Figure 10-2 ABWR Passive Severe Accident Mitigation Features

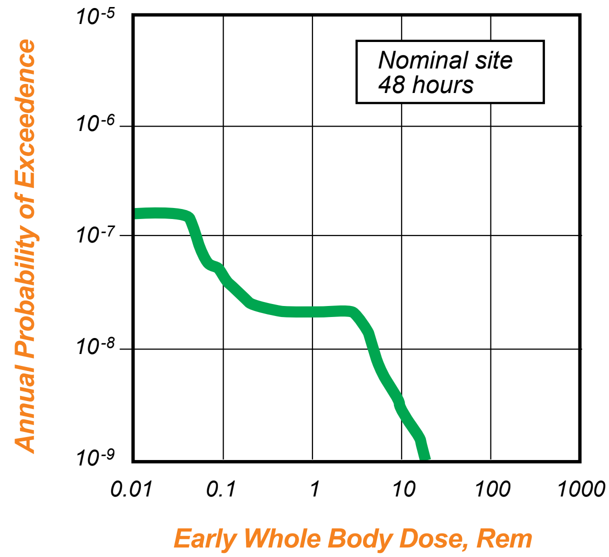


Figure 10-3. ABWR Whole Body Dose at 800 m

In summary, the detailed PRA of the ABWR design demonstrates that both the core damage frequency and the offsite risk goals established by the US NRC are met with substantial margin. The ABWR represents a substantial improvement in safety as compared to earlier plant designs. The high degree of safety is attributable to the many improved features in the design and to the use of PRA in the design process.

Chapter 11 Plant Operations

Basic BWR Operation

Control of the BWR during normal operation is simple, largely because the direct cycle creates a strong interrelationship between the three primary operating conditions: reactor thermal power, pressure, and water level. Because they are strongly interrelated, these conditions can be automatically controlled. Thermal power, and hence steam flow rate, is changed in a BWR by changing either the re-circulation flow rate or control rod position. Steam dome pressure in the reactor pressure vessel (RPV) is automatically maintained at a constant value by controlling the opening of the main turbine control valves or, if the pressure increases significantly, by opening the turbine bypass valves to allow some of the steam to bypass the turbine and go directly to the main condenser. Reactor water level is automatically maintained at a constant level by controlling the speed of the reactor water feed pumps and, hence, the feedwater flow rate. By coordinating the control of pressure, water level, and reactor power, the ABWR can be operated in a load-following mode for automatic load-following operations. Above 65% power, automatic load following is achieved by only changing re-circulation flow. Below this level, power is controlled by using control rods.

Plant startup and shutdown operations are also simple in the ABWR. During plant startup, control rods are withdrawn to bring the reactor to criticality. RPV heatup, pressurization and initial power ascension are achieved by continuing to withdraw the control rods. Reactor power is increased in this way until, in order of events, the main turbine is synchronized, the first load is applied to the generator and the automatic re-circulation flow control range is reached. When this last condition is reached, reactor power is increased by increasing the recirculation flow rate exclusively.

During full power operation, the turbine accepts the steam generated in the reactor and operates in a "turbine-follow- reactor" mode.

For a plant shutdown, the reverse sequence occurs. After the control rods are fully inserted, reactor cooldown and decay heat removal are accomplished by bypassing steam to the main condenser and by operation of the Residual Heat Removal (RHR) System.

Operating Map

The ABWR operating map is a steady-state representation of reactor power vs. recirculation flow (Figure 11-1). The nearly horizontal lines are control rod lines representing prescribed control rod patterns. The nearly vertical lines are lines of constant recirculation pump speeds. Any operational path that changes the power and flow from one condition to another condition through control rod maneuver and/or re-circulation flow change can be traced on this figure.

Operation in certain areas of the power-flow map is prohibited in order to (1) maintain core thermal limits, (2) avoid operation above the licensed power level, (3) avoid conditions where core instability may occur, (4) avoid regions not analyzed, or (5) prevent operation where excess moisture in the steam may be carried to the main turbine. Although the power-flow map indicates possible operating states, normal plant operation is along the 100% control rod line

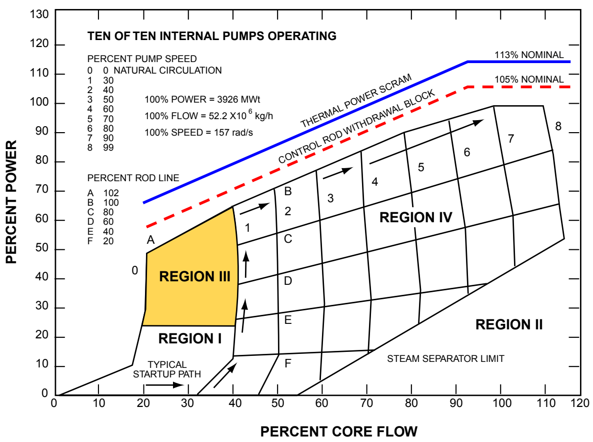


Figure 11-1 ABWR Power-Flow Map

(i.e., Curve B in Figure 11-1). In this figure, normal plant startup and shutdown trajectory is indicated. Automatic protective interlocks prevent operation outside prescribed limits or initiate automatic reactor shutdown if required.

Plant Startup and Shutdown

To ensure consistent and error-free operation, plant startup and shutdown operations are automated. The startup and shutdown sequences are divided into a number of small, automated sequences. For example, during startup, establishment of condenser vacuum, criticality, heatup, pressurization, turbine synchronization, power ascension by control rods, and power ascension by re-circulation flow are all automated sequences. The operator, however, initiates automatic sequences after checking that all prerequisites of a given sequence are satisfied. In cases where a mode change involves safety-related equipment (e.g., repositioning of the reactor mode switch during a startup or shutdown), the computer system will prompt the operator to take a manual action. There is no automation associated with the initiation or shutdown of safety-related equipment. Manual operation of any equipment from the main control console is always available on demand.

The ABWR can reach 100% power from a cold shutdown condition in less than 25 hours, and in less than 5 hours from a hot condition (all control rods fully inserted). Figure 11-2 shows the key steps in the plant startup sequence, and Figure 11-3 shows the key steps in the plant shutdown sequence. Normally, during shutdown, the reactor internal pumps (RIPs) continue to run at minimum speed until the reactor vessel is flooded in preparation for refueling. However, if the water level within the vessel is raised above the separator spillover, natural circulation alone is sufficient to transfer the core decay heat to the downcomer region of the RPV, where it is removed by the RHR System. Automatic Response to Transients.

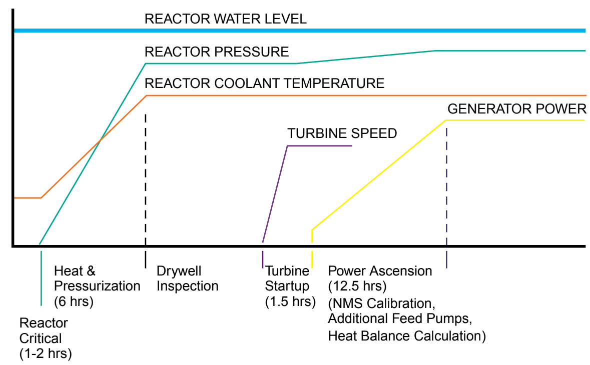


Figure 11-2 ABWR Startup Sequence

For certain transient events, such as the trip of one reactor feedpump, condensate pump, control rod drive (CRD) hydraulic pump, or reactor internal pump, the ABWR is designed to remain on-line with no change in power. This is facilitated by automating the responses to such transients.

Other transients will require a power reduction. This latter class of transients includes the loss of feedwater heating, tripping of two or three reactor internal pumps, partial generator load rejection, full load rejection (for plants electing to have a large turbine bypass system), tripping of one main condenser circulating water pump, or high main condenser pressure. Response to this type of transient is also automated and involves reducing reactor power to within the capability of the remaining operable equipment. When the failed equipment is returned to service, full power operation can be restored in a short time.

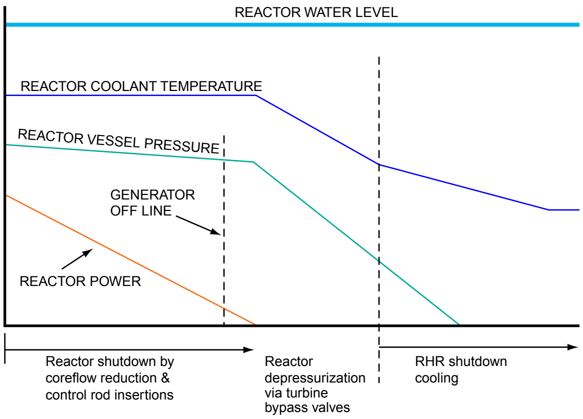


Figure 11-3 ABWR Shutdown Sequence

For plants electing to have a full generator load rejection without reactor scram capability, a 110% turbine bypass system is provided. The automatic response to a full load rejection consists of (1) rapid opening of the turbine bypass valves when the turbine control valves trip, (2) reduction of reactor core flow and power, and (3) insertion of a preselected number of control rods. The reactor power is reduced to approximately 60% by these automatic actions. The operator can achieve further power reduction through additional control rod insertions. If the generator can be re-loaded, full power operation can be rapidly restored.

Automatic Load-Following Capability

Grid requirements for frequency control, load regulation, and load following can be met by the ABWR. The automatic process control systems in the ABWR enable the reactor to respond to changes in turbine/load demand so that the generated steam flow matches that required by the turbine to maintain proper load and frequency operation. In the 65% to 100% power range, the power change rate capability is approximately 1% per second using only re-circulation flow changes. Below 65%, the power change rate is approximately 2.5% per minute using control rod motion. All of these operations are significantly faster than required to respond to load and frequency changes.

ABWR load-following operation is fully automated. In this manner, the ABWR operates in the "turbine-follow-reactor" mode.

Automated Response to Design Basis Accidents

In the event of a design basis accident, operator action is not required for an indefinite period. This capability is achieved by automatic initiation of (1) reactor protection systems for reactor shutdown, containment isolation, and emergency core cooling, (2) suppression pool cooling and reactor scram on high pool temperature (to mitigate an inadvertent safety-relief valve opening event) and (3) boron injection, feedwater flow runback, and insertion of control rods by electric-mechanical means should there be an anticipated transient without scram (ATWS) event. These features enhance overall plant safety and allow operators, if necessary or appropriate, to take control of the accident response.

Flexibility in Fuel Cycle Length

One of the advantages of ABWR and BWR operation in general is the degree of flexibility in designing the fuel management plan. This is due primarily to the strong negative void coefficient associated with the direct cycle (two-phase) flow in the core. At the end of a planned fuel cycle, when all control rods have been fully withdrawn, the ABWR can maintain full power operation for an extensive length of time by increasing the core flow rate (up to 110%) due to the built-in capacity margin of the ABWR internal re-circulation pumps. The increased core flow will reduce the average core void content and thus increase reactivity and maintain balanced reactor power.

At the end of this increased core flow operation, the reactor can still continue its operation at a gradually reduced power level, or a "coastdown operation", during which the reactor power will gradually decrease at a slow rate. This is again due to the negative void feedback mechanism where lower power level is automatically regulated (or balanced) due to slightly decreasing voids and thus slightly increasing core reactivity. BWR coastdown operation can maintain a coastdown final power level of about 85% rated power. Finally, an operation with reduced feedwater temperature can also increase core reactivity, thus maintaining a higher power level for a longer period of time at the end of the cycle. The ABWR incorporates design features to gradually reduce feedwater temperature.

All of these operations can extend the plant fuel cycle up to the order of a couple of months, thus providing significant flexibility for plant operation and outage planning.

Technical Specifications

Plant operation (and maintenance) is governed by a set of technical specifications (Tech Specs). The purpose of these Tech Specs is to ensure that (1) required systems will be available in the event of an accident and (2) the plant will always remain within analyzed conditions. Technical specifications also govern maintenance, and the ABWR Tech Specs, in particular, allow for online maintenance, a key element in achieving very short outage lengths. By taking full advantage of design features such as the three ECCS divisions, four divisions of safety instrumentation and control, and low core damage frequency, the ABWR Tech Specs provide a sufficient time to perform maintenance of inoperable equipment during plant operation. For example, one full ECCS division may be taken out of service for inspection and maintenance for fourteen days before it must be returned to service. Thirty days are allowed for a division of safety instrumentation out of service because the remaining three divisions of instrumentation still provide a single-failure-proof capability. The comparable time for one emergency diesel generator is fourteen days.

Emergency Plant Operation

Symptom-based emergency procedure guidelines for the ABWR enable a very simple approach for the operators to deal with accidents, without having to identify the cause of the accident. For the ABWR, only a few key plant parameters need be monitored. These parameters provide simple and clear entry conditions into Emergency Operating Procedures (EOPs). Only a few parameters will need to be controlled in an ABWR during an emergency: reactor pressure, reactor power, reactor water level, and containment conditions such as pressure and temperature. The objective of the EOPs is to provide the necessary steps for the operator to bring the reactor to a safe, cold shutdown condition. The ABWR EOP is thus very much simplified.

There are primarily four simple control strategies, one each for control of the RPV, primary containment, secondary containment, and site radioactivity. Each of these strategies provides for the proper operations to control specified parameters. In extreme degraded conditions, contingencies are also specified to bring the plant to a safe condition.

Summary

Due to inherently safe BWR design principles and improved design features, the ABWR provides for substantial operation simplicity and flexibility. Simple plant automatic processes, for example, control reactor water level, pressure, and power. Power changes are easily made by changing reactor re-circulation flow rate or control rod position. Automatic load-following capability provides for system frequency control, load regulation, and periodic load cycling. Full load rejection capability allows for fast power recovery after grid disturbances. Highly automated normal plant operation provides simple startup, shutdown, and transient response capability. For design basis accidents, the ABWR is designed to not require operator action for 72 hours through automation of selected system automation. Through safety system redundancy in the design, the technical specifications provide for extended periods for maintenance of equipment during operation. Simple symptom-based emergency operating strategies and procedures enable the operators to place the plant in a safe condition should an accident occur.

Appendix
Key Design Characteristics

A

This appendix lists key design characteristics for the ABWR, using the standard design licensed in the US as a reference. Further details can be obtained from the ABWR Standard Safety Analysis Report, Rev. 9.

Overall Design	
Site Envelope	
Safe shutdown earthquake, g	0.3 envelope
Wind design, km/h	197
Maximum tornado, km/h	483
Max dry bulb/wet bulb ambient temperature, °C	46/27
Thermal and Hydraulic	
Rated Power, MWt	3926
Generator Output, MWe	1371
Steam flow rate, Mkg/h	7.64
Core coolant flow rate, Mkg/h	52.2
System operating pressure, MPa	7.17
Average core power density, kW/l	49.2
Maximum linear heat generation rate, kW/m	44.0
Average linear heat generation rate, kW/m	13.6
Minimum critical power ratio (MCPR)	1.2 - 1.3*
Core average exit quality, %	14.5
Feedwater temperature, °C	215.6

* Depending on the cycle length

Core Design	
Fuel Assembly	
Number of fuel assemblies	872
Fuel rod array	10 x 10
Overall length, cm	447
Weight of UO2 per assembly, kg	183
Number of fuel rods per assembly	92
Rod diameter, cm	1.026
Cladding material	Zircaloy-2
Fuel Channel	
Thickness corner/wall, mm	2.5/1.6
Dimensions, cm	14 X 14
Material	Zircaloy-4
Reactor Control System	
Method of variation of reactor power	Moveable control rods and variable forced coolant flow
Number of control rods	205
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	18
Number of hydraulic accumulators	103
Hydraulic scram speed, sec to 60% insert	1.44
Electric drive speed, mm/sec	30
Type of temporary reactivity control	Burnable poison; gadolinia urania fuel rods
Incore Neutron Instrumentation	
Total number of LPRM detectors	208
Number of incore LPRM penetrations	52
Number of LPRM detectors per penetration	4
Number of SRNM penetrations	10

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	21.0
Steam Separators and Dryers	
Separator type	AS-2B
Number of separators	349
Dryer type	Chevron
Reactor Recirculation	
Number of recirculation pumps	10
Pump motor type	Wet motor
Design flow rate m^3/hr/pump	7700
Flow Control	solid state variable frequency drive
Pump inertia, kg-m^2	19.5*
Main Steam	
Number of steam lines	4
Diameter of steam lines, cm	70
Number of safety/relief valves	18

* Coastdown is determined by 4400 kg-m^2 M-G set flywheel, not pump

Emergency Core Cooling	
Reactor Core Isolation Cooling	
Number of pumps	1
Type	Steam turbine driven pump
Flow rate, m^3/h	182
Pressure range, MPa	1.1 - 8.2
High Pressure Core Flooder	
Number of Pumps	2
Flow rate at 8.1 MPa, m^3/hr	182
Flow rate at 0.7 MPa, m^3/hr	727
Low Pressure Core Flooder/ Residual Heat Removal	
Number of Loops	3
Flow rate at 275 kPa, m^3/hr	954
Automatic Depressurization	
Number of relief valves	8
Reactor Building Cooling Water	
Number of loops	3
Flow rate per loop (LOCA), m^3/hr	1822
Heat removal duty per loop (LOCA), MWt	30
Service Water	
Number of loops	3
Hydrogen Recombiner	
Type	Thermal reaction
Number of units	2
Flow rate per unit, m^3/hr	128
Standby Liquid Control	
Number of pumps	2
Flow rate per pump, m^3/h	11.4

Containment	
Primary	
Type	Pressure suppression
Construction	Reinforced concrete with steel liner
Drywell	Concrete cylinder
Wetwell	Concrete cylinder
Design pressure, MPa	0.31
Design leak rate, % free volume/day	0.5*
Drywell free volume, m^3	7350
Wetwell free volume, m^3	5960
Suppression pool water volume, m^3	3580
Number of vertical vents	10
Vertical vent diameter, m	1.2
Number of horizontal vents/vertical vent	3
Horizontal vent diameter, m	0.7
Secondary	
Type	Controlled leakage
Construction	Reinforced concrete/steel
Design pressure, Pa	1724
Design in leakage rate at 6.4 mm water, %/day	50

* Excluding MSIV leakage

Auxiliary Systems	
Reactor Water Cleanup	
Number of pumps	2
Type	Canned rotor
Flow rate per pump, m^3/h/ % of feedwater	154/2
No. of regenerative heat exchangers	1
No. of non-regenerative heat exchangers	2
Return water temperature (cleanup mode), °C	225
Fuel Pool Cooling and Cleanup	
Number of pumps	2
Flow rate per pump, m^3/h	250
Number of heat exchangers	2
Total heat removal capability, MWt	4.0
Suppression Pool Cleanup	
Number of pumps	1
Flow rate, m^3/hr	250
Reactor Building Cooling Water	
Number of loops	3
Flow rate per loop (normal), m^3/h	1200
Heat removal duty (normal), MWt	15
Flow rate per loop (shutdown), m^3/h	2400
Heat removal duty (shutdown), MWt	34
Drywell Cooling	
Number of fans	3
Flow rate per fan, m^3/h	51480
Heat removal duty, MWt	1.25

Appendix Frequently Asked Questions B

What has the ABWR done to reduce the potential for Intergranular Stress Corrosion Cracking?

Boiling Water Reactors (BWR) have been supplying commercial electric power for about 30 years. In that time, the understanding of both the materials of construction and the degradation mechanisms has increased significantly. This understanding has been used to make the ABWR materials robust while preserving the experience of the operating BWR fleet where possible. The ABWR makes use of austenitic stainless steels and nickel base alloys as well as low alloy and carbon steel for all the major components. The plant design takes into account the potential effects of stress corrosion cracking, irradiation embrittlement, erosion/corrosion and radiation buildup in the careful application of these materials. This application of materials is made in conjunction with good fabrication practice and strict operating control of the coolant environment chemistry to minimize the potential for long-term degradation and radiation buildup. The following is a more detailed explanation of each area.

Austenitic Stainless Steel Materials for ABWR Internals and Components

Austenitic stainless steels, because of high general corrosion resistance, high toughness, ease of fabrication, and acceptable strength, have been commonly used for reactor internals and some piping systems. In the ABWR, the primary consideration in the use of these materials is the development of resistance to intergranular stress corrosion cracking (IGSCC) both in low and high fluence locations. Based on field and laboratory efforts, this resistance is tied to control of the material's composition, control of the fabrication processes and control of the coolant environment.

The extensive understanding has now thoroughly demonstrated that reduction of carbon content in austenitic stainless steel reduces susceptibility to IGSCC. Three main austenitic materials are being used: Type 316NG/316L wrought material, CF3M cast materials, and XM-19 high strength stainless steel.

For the majority of the reactor internal structures, resistance to sensitization for the ABWR is achieved by using a special Type 316NG (Nuclear Grade). This alloy has carbon restricted to a maximum 0.020% by weight to prevent sensitization, has nitrogen additions up to a maximum of 0.12% to maintain the desired strength levels and has specific fabrication and processing controls to increase the resistance to any crack initiation. Alternately, where design does not need the mechanical strength of Type 316, Type 316L is used with the same carbon limit and controls on fabrication processes.

Types 308L/316L weld metal and CF3M austenitic stainless steel castings are also used in the reactor as well as for many of the complex shape components such as valve body castings. These alloys have a duplex microstructure which has been found to have very high resistance to IGSCC. Their resistance is maximized by again limiting carbon content to less than 0.03% while requiring a minimum ferrite levels (greater than 8%) to assure adequate resistance to IGSCC.

For any highly irradiated stainless steel in-core components such as control blades and instrument tubes, it is desirable to provide additional potential for the component to reach the intended design life. Consequently, for the ABWR both the component design and materials of construction have been altered to provide this higher resistance against IASCC. The ABWR designs are crevice-free and incorporate control of trace element chemistry such as sulfur and phosphorus which provide this resistance.

The final material selected for specialized ABWR applications is XM-19. This is a high chromium, high manganese austenitic stainless steel alloy that is strengthened with specific nitrogen addition (as opposed to carbon). It has an established record of excellent performance in very demanding applications in BWR service where both high strength and extremely high resistance to IGSCC are required. Example applications in the ABWR include components in the FMCRD as well as core plate and top guide bolting.

Nickel Base Alloys for ABWR Application

Nickel base alloys, particularly Alloy 600 and X-750, are also used extensively in BWRs. In addition to their excellent corrosion resistance, nickel alloys have a thermal expansion coefficient more similar to the low alloy steel used in the construction of the reactor pressure vessel and nozzles. This feature has led to the application of Alloy 600 for several parts of the core internals, including the shroud support structure as a transition material placed between the alloy steel reactor vessel and stainless steel internals. The other desirable feature of Alloy 600 is that it is resistant to stress corrosion crack initiation in the post-weld heat treated condition, thereby allowing attachment to low alloy steel prior to stress relief. For the ABWR, to add margin against IGSCC these alloys and their weld metals have been modified with stabilizing additions of niobium to reduce the potential for chromium depletion. For the wrought structures, the ABWR employs an Alloy 600 (designated as 600M) which has a niobium content on the order of 1 to 3%. For weld metal, Alloy 82 with high stabilizing ratios is used, leading to high IGSCC resistance as well.

Alloy X-750 is also used to a limited extent in the ABWR where high strength is required. Alloy X-750 is a precipitation hardening high strength nickel base alloy which will perform well in the ABWR environment under the proper application. To improve stress corrosion cracking resistance for the ABWR, both the heat treatment and stresses of Alloy X-750 are controlled based on the extensive operating plant experience and material understanding.

Component Fabrication and Design Considerations

Aside from selection of materials that are intrinsically resistant to IGSCC, the fabrication of components will be controlled to ensure that this high resistance is maintained in the finished part. Low heat input welding processes will be used to decrease the likelihood of any sensitization due to the welding process. The margin of IGSCC resistance will be reduced even for the intrinsically resistant materials by abusive fabrication practices such as excessive cold working. Post-manufacturing surface processing such as polishing can be applied to the weld heat-affected zones to remove surface cold work, residual stresses and strains, thereby further increasing the resistance to IGSCC initiation. Contamination with in-process materials having high levels of species such as chlorine, fluorine and sulfur can degrade performance. Consequently, to protect the high level of resistance obtained by using materials such as described above, for the ABWR, the entire fabrication and installation process is controlled to prevent detrimental practices.

A final consideration in control of IGSCC is the design itself. Crevices have been eliminated where possible and the number of welds has been reduced. The top grid structure, for example, is manufactured from a single solution annealed plate, thereby eliminating the risk of IGSCC. The core shroud is constructed in a manner to locate all welds away from the highest fluence locations.

Water Chemistry Control

With respect to materials performance in the BWR, a substantial body of data now exists that shows that water chemistry is a key factor in materials degradation as well as radiation buildup processes, especially for core internal components. It has become very clear that the presence of oxidizing species in the high purity coolant (such as oxygen and hydrogen peroxide), as well as anionic species that contribute to the coolant conductivity, can be correlated with incidence of IGSCC cracking as well as the rate of progression of any initiated cracks. Events such as resin intrusions can also reduce the resistance to SCC initiation. The benefits of operating with good water chemistry are very clear and the owners of operating BWRs (even plants that have not experienced severe degradation in water chemistry such as resin intrusions) have adopted practices to obtain low conductivity. Thus, the Electric Power Research Institute (EPRI) has developed guidelines for water chemistry control that provide specific recommended limits on overall conductivity as well as specific species. For the ABWR, recommended water chemistry is applied such that the units operate at or below a target conductivity of 0.1 μS/cm whenever the reactor system is greater than 200°F (93°C).

These guidelines also recommend the application of hydrogen water chemistry (HWC) when possible. The main function of HWC is to inject sufficient hydrogen into the condensate to reduce oxidizing species and maintain the Electrochemical Potential (ECP) of the reactor water below -0.23 V Standard Hydrogen Electrode (SHE). The addition of hydrogen has been documented to effectively control Stress Corrosion Cracking (SCC). While HWC can result in additional main steam line radiation, the use of noble metal technology such as noble metal chemical addition (NobleChemTM) can reduce the hydrogen needed for SCC control through catalytic action while limiting any steam line radiation issues.

Can the reactor vessel and attached piping really last 60 years?

Alloy Steel Pressure Vessel and Carbon Steel Piping

The reactor vessel is also a very important component where 30 years of BWR operating experience and materials understanding has been used in the ABWR. The vessel draws from the understanding of materials degradation mechanisms and fabrication experience. Of key interest are the effects of neutron irradiation over the life of a reactor vessel beltline region (the region immediately around the core), which results in a progressive loss of fracture toughness. This shift in NDT (nil ductility temperature) to higher temperatures can influence reactor operation with respect to such items as bolt-up temperature and hydrostatic test temperature. Consequently, it is advisable to start with a low NDT and use material resistant to neutron radiation damage. Copper and phosphorus have been identified as detrimental elements in alloys steels with respect to neutron damage, and nickel is considered to also be a factor. The corrosion behavior of reactor vessel steels is also well understood. There is significant experience with these steels to assure that there is high resistance to any SCC processes under the ABWR operating coolant environment.

For the ABWR reactor vessel, high toughness and retention of that toughness in service is provided. Strict controls are specified on initial fracture toughness of vessel components. Reactor vessel shell courses are required to have RTNDT (reference nil ductility transition temperature) of -20°C or less. In addition, copper content is controlled to 0.05% maximum for base metal and weld metal. Phosphorus is limited to 0.006 and 0.008%, respectively. With these very low levels of copper, the shift in NDT for the ABWR vessels is expected to be very small over the operating life of the units. The estimated shift in RTNDT is estimated to be less than 11°C over a 60 year operating life.

Low alloy steels are also used in many piping applications. The reduced corrosion rates make it attractive for several systems. However, the ABWR also employs carbon steel piping in some systems. This selection has been made to aid in plant fabrication while taking advantage of the steel's inherent toughness. Although erosion-corrosion in carbon steel power plant piping systems has been experienced in some units, oxygen injection is used for the ABWR to minimize this concern. Corrosion allowances for 60 years operation are included in the piping design.

What has ABWR done to address worker radiation exposure?

Materials Considerations

With the increasing focus on managing radiation buildup, the ABWR focuses on both alloy selection and water chemistry modification and control to minimize this effect. 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 base 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 base 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 base alloys are almost completely eliminated inside the reactor pressure vessel of the ABWR. 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 base alloys for control blade guide rollers and pins rather than cobalt base 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 Reactor Water Cleanup (RWCU) 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 ABWR 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 and precoat operations in the reactor water cleanup and fuel pool cleanup systems).
- Equipment is designed to facilitate maintenance. Equipment such as the RHR 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.
- 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 and condensate demineralizer on the reactor feedwater.
- External recirculation pumps and recirculation piping have been replaced with internally mounted recirculation pumps. The RIPs can be easily removed for maintenance outside the lower drywell radiation zone.
- Clean purge water is continuously supplied to the RIPs and FMCRDs to keep the equipment from accumulating radioactive contaminants.

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 radioactivity, 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 ABWR will be significantly reduced relative to conventional light water reactors due to these materials technologies. Based on early feedback form the operating ABWRs in Japan, it is expected that the US Utility Requirements Document goal of less than 100 manRem/yr (1manSv/yr) will be met by a wide margin.

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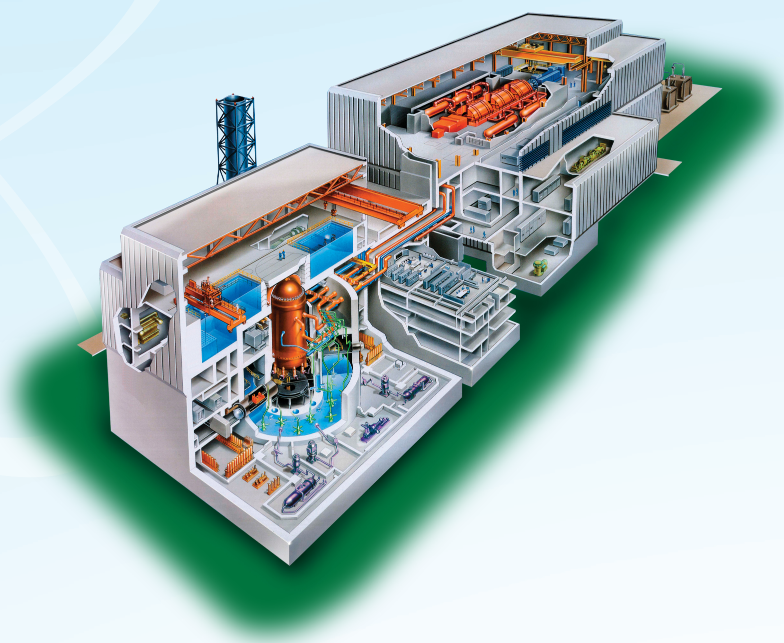

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