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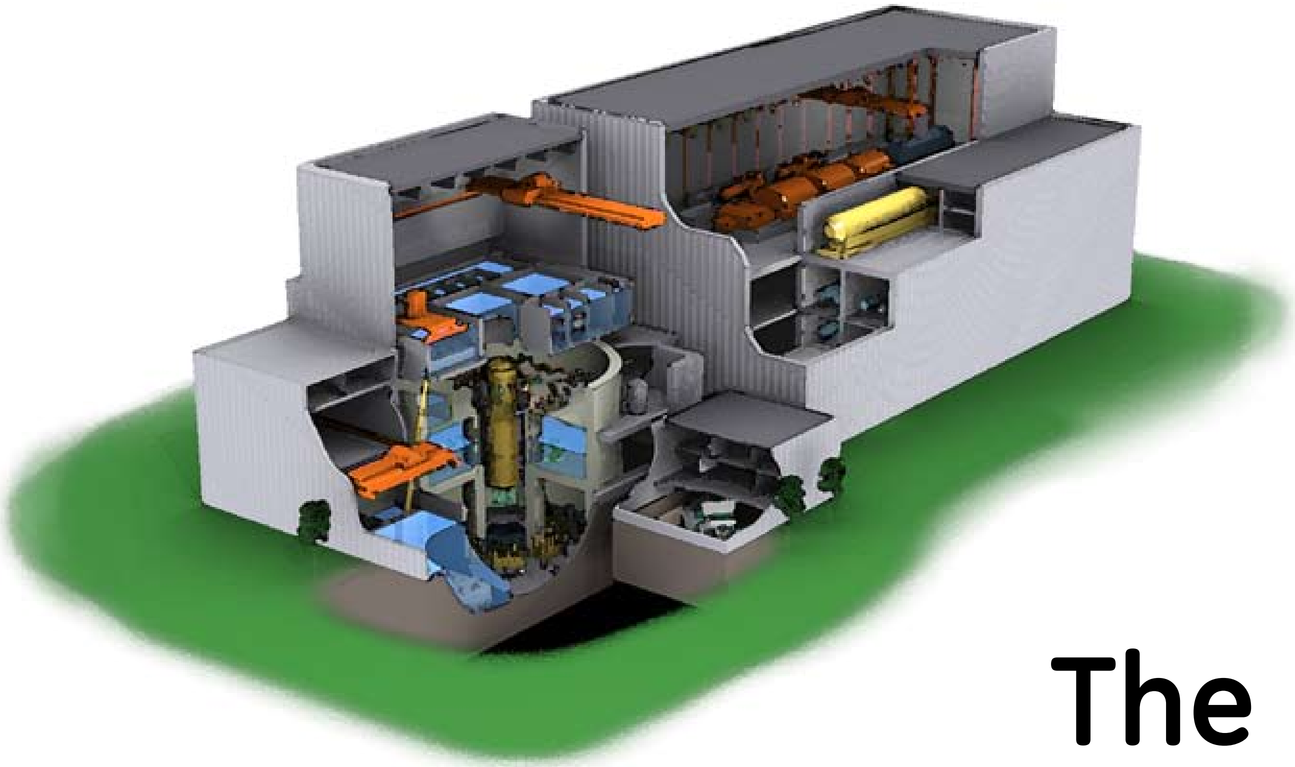

The ESBWR Plant General Description

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Economic Simplified Boiling Water Reactor Plant General Description

June 2006

DISCLAIMER OF RESPONSIBILITY

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Acronyms

ABWR	Advanced Boiling Water Reactor
ACRS	Advisory Committee on Reator Safeguards
ADS	Automatic Depressurization System
AFIP	Automatic Fixed In-Core Probe
AHS	Auxiliary Heat Sink
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
APR	Automatic Power Regulator System
APRM	Average Power Range Monitor
ARM	Area Radiation Monitoring
ARI	Alternate Rod Insertion
ASD	Adjustable Speed Drive
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
BAF	Bottom of Active Fuel
BiMAC	Basemat-internal Melt Arrest Coolability
BOP	Balance-of-Plant
BWR	Boiling Water Reactor
C&I	Control and Instrumentation
CB	Control Building
CCC	Control Cell Core
CCFP	Contingent Containment Failure Probability
CDF	Core Damage Frequency
C&FS	Condensate and Feedwater System
CIRC	Circulating Water System
CIV	Combined Intermediate Valves
CLAVS	Clean Air Ventilation Subsystem
CMS	Containment Monitoring System
COE	Cost of Electricity
COL	Combined Operating License
CONAVS	Controlled Area Ventilation Subsystem
CP	Construction Permit/Control Processor
CPR	Critical Power Ratio
CPS	Condensate Purification System
CRD	Control Rod Drive
CRDHS	Control Rod Drive Hydraulic System
CRHA	Control Room Habitability Area
CRT	Cathode Ray Tube
CST	Condensate Storage Tank
CWS	Chilled Water System
DAW	Dry Active Waste
DBA	Design Basis Accident
DCIS	Distributed Control & Information System
DCPS	DC Power Supply
DCS	Drywell Cooling System
DCV	Drywell Connecting Vent
DG	Diesel Generator
DMC	Digital Measurement Controller
DoE	Department of Energy
DPS	Divers Protection System
DPV	Depressurization Valve
DW	Drywell
EAB	Exclusion Area Boundary
EB	Electrical Building
EBAS	Emergengy Breathing Air System
ECCS	Emergency Core Cooling System
ECP	Electrochemical Potential
ECW	Emergency Chilled Water
E-DCIS	Essential DCIS
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
FAPCS	Fuel and Auxiliary Pool Cooling System
FB	Fuel Building
FCU	Fan Cooling Unit
FDA	Final Design Approval
FFTR	Final Feedwater Temperature Reduction
FIV	Flow-Induced Vibration
FMCRD	Fine Motion Control Rod Drive
FOAKE	First-of-a-Kind Engineering
FPS	Fire Protection System
FSAR	Final Safety Analysis Report
FSC	First Structural Concrete
FTDC	Fault Tolerant Digital Controller
FW	Feedwater

FWP	Feedwater Pump
FWC	Feedwater Control System
GE	General Electric Company
GETAB	General Electric Thermal Analysis Basis
GDCS	Gravity Driven Cooling System
GPM	Gallons per minute
GT	Gamma Thermometer
HCU	Hydraulic Control Unit
HCW	High-Conductivity Waste
HEPA	High Efficiency Particulate Air
HFF	Hollow Fiber FIlter
HIC	High Integrity Container
HPCP	High Pressure Condensate Pump
HPNSS	High Pressure Nitrogen Supply System
HSI	Human System Interface
HVAC	Heating, Ventilation and Air-Conditioning
HWC	Hydrogen Water Chemistry
I&C	Instrumentation and Control
IASCC	Irradiation-Assisted Stress Corrosion Cracking
ICS	Isolation Condenser System
IEEE	Institute of Electrical and Electronic Engineers
IFTS	Inclined Fuel Transfer
IGSCC	Intergranular Stress Corrosion Cracking
ILRT	Integrated Leak Rate Test
IMC	Induction Motor Controller
IMS	Information Management System
IRM	Intermediate Range Monitor
ISI	In-Service Inspection
LA	Low Activity
LCW	Low Conductivity Waste
LD	Laundry Drain
LDW	Lower Drywell
LD&IS	Leak Detection and Isolation System
LFCV	Low Flow Control Valve
LHGR	Linear Heat Generation Rate
LLRT	Local Leak Rate Test
LOCA	Loss-of-Coolant Accident
LOFW	Loss of Feedwater
LOOP	Loss of Offsite Power
LOPP	Loss of Preferred Power
LPCI	Low-Pressure Coolant Injection
LPCP	Low-Pressure Condensate Pump
LPCRD	Locking Piston Control Rod Drive
LPRM	Local Power Range Monitor
LPZ	Low Population Zone
LTP	Lower Tie Plate
LWMS	Liquid Radwaste Management System
MCC	Main Control Console/ Motor Control Center
MCOPS	Manual Containment Overpressure Protection System
MCES	Main Condenser Evacuation System
MCR	Main Control Room
MCPR	Minimum Critical Power Ratio
M-G	Motor-Generator
MITI	Ministry of International Trade and Industry (Japan)
MLHGR	Maximum Linear Heat Generation Rate
MMI	Man-Machine Interface
MO	Motor-Operated Valve
MRBM	Multi-Channel Rod Block Monitoring System
MS	Main Steam Subystem
MSIV	Main Steam Isolation Valve
MSR	Moisture Separator Reheater
MUX	Multiplexer
MWS	Makeup Water System
NBS	Nuclear Boiler System
NDT	Nil Ductility Temperature
NE-DCIS	Non-Essential DCIS
NMS	Neutron Monitoring System
NMO	Nitrogen Motor Operated Valve
NO	Nitrogen (piston) Operated Valve
NPHS	Normal Power Heat Sink
NRC	Nuclear Regulatory Commission
NRHX	Non-Regenerative Heat Exchanger
NSSS	Nuclear Steam Supply System
O&M	Operation and Maintenance
OGS	Offgas System
OPRM	Oscillation Power Range Monitor
PCCS	Passive Containment Cooling System
PCI	Pellet Clad Interaction
PCS	Plant Computer System
PCT	Peak Fuel Clad Temperature
PCV	Primary Containment Volume
PG	Power Generation (loads)
PGCS	Power Generation Control System
PIP	Plant Investment Protection (loads)
PIP	Position Indicator Probe

PLR	Part Length Fuel Rod
PRA	Probabilistic Risk Assessment
PRMS	Process Radiation Monitoring System
PRNM	Power Range Neutron Monitor System
PSWS	Plant Service Water System
PWR	Pressurized Water Reactor
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RBC	Rod Brake Controller
RCCWS	Reactor Component Cooling Water System
RC&IS	Rod Control and Information System
RCCV	Reinforced Concrete Containment Vessel
RCIC	Reactor Core Isolation System
RCPB	Reactor Coolant Pressure Boundary
RHR	Residual Heat Removal
RHX	Regenerative Heat Exchanger
RMU	Remote Multiplexer Unit
RO	Reverse Osmosis
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSS	Remote Shutdown System
RTNDT	Reference Nil Ductility Temperature
RW	Radwaste Building
RWCU/SDC	Reactor Water Cleanup/ Shutdown Cooling System
RWM	Rod Worth Minimizer System
S&PC	Steam and Power Conversion System
SA	Severe Accident
SAR	Safety Analysis Report
SB	Service Building
SBO	Station Blackout
SB&PC	Steam Bypass and Pressure Control System
SBWR	Simplified Boiling Water Reactor
SCRRI	Select Control Rod Run-in
SDV	Scram Discharge Volume
SJAE	Steam Jet Air Ejector
SLCS	Standby Liquid Control System
SOE	Sequence of Events
SP	Suppression Pool
SPC	Suppression Pool Cooling
SPDS	Safety Parameter Display System
SRM	Source Range Monitor
SRNM	Startup Range Neutron Monitor
SRV	Safety/Relief Valve
SSAR	Standard Safety Analysis Report
SSE	Safe Shutdown Earthquake
SSLC	Safety System Logic and Control
SWMS	Solid Waste Management System
TAF	Top of active Fuel
TB	Turbine Building
TBCE	Turbine Building Component Exhaust
TBS	Turbine Bypass System
TBV	Turbine Bypass Valve
TCCWS	Turbine Component Cooling Water System
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
UAT	Unit Auxilliary Transformer
UDW	Upper Drywell
UHS	Ultimate Heat Sink
UPS	Uninterruptable Power Supply
URD	Utility Requirements Document
UTP	Upper Tie Plate

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, with 25 more units under construction. These plants account for 17% of the world's electricity. The industry remains dynamic, as evidenced by the fact that several new plants enter commercial operation every year and there are, typically, 30 or more in various stages of construction at any given time.

Generating electricity with nuclear energy permits economic and social development to be sustainable; that is, not limited by encroaching environmental concerns. A non-nuclear, baseload power plant generates electricity by burning fossil fuels day in and day out and releasing the by-products to the environment. A nuclear plant, on the other hand, generates large amounts of electricity with virtually no impact on the environment. In quantitative terms, if the world's nuclear plants were replaced with coal-fired plants, global CO_2 emissions would increase by 8% every year. This would amount to 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 ESBWR advanced nuclear plant will play an important role in meeting the conflicting needs of developed and developing economies for massive amounts of new electricity and the need worldwide to limit CO_2 emissions. It continues to use advanced technologies first applied in the Advanced Boiling Water Reactor (ABWR) with simplifications in the recirculation system and ECCS. Two 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 ESBWR represents an entirely new approach to the way nuclear plant projects are undertaken, modelled after the successful process used for ABWR. The ABWR was licensed and designed in detail before construction ever began. Once construction did begin, it proceeded smoothly from start to finish in just four years.

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

Fifty Years in the Making

The Boiling Water Reactor (BWR) nuclear plant, like the Pressurized Water Reactor (PWR), has its origins in the technology developed in the 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 recirculation loops were needed. This change first appeared in the Dresden-2 BWR/3 plant.

Product Line	First Commercial Operation Date	Representative Plant/ Characteristics
BWR/1	1960	Dresden 1 Initial commercial-size BWR
BWR/2	1969	Oyster Creek Plants purchased solely on economics Large direct cycle
BWR/3	1971	Dresden 2 First jet pump application Improved ECCS: spray and flood capability
BWR/4	1972	Vermont Yankee Increased power density (20%)
BWR/5	1977	Tokai 2 Improved ECCS Valve flow control
BWR/6	1978	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 and fiber optic technology Improved ECCS: high/low pressure flooders
ESBWR	under review	Natural circulation Passive ECCS

Table 1-1. Evolution of the GE BWR

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

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

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

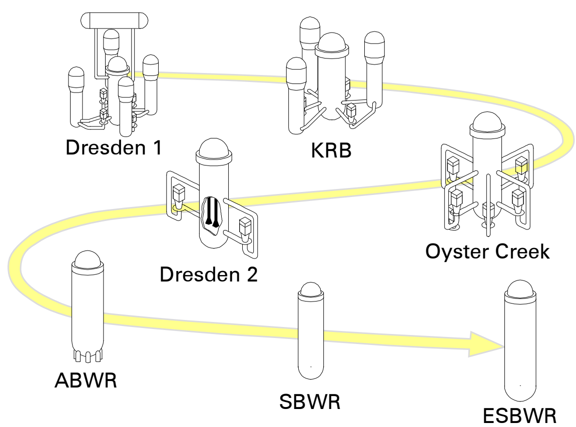


Figure 1-1. Evolution of the Reactor System Design

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

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

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

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

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

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 35 operating BWRs.

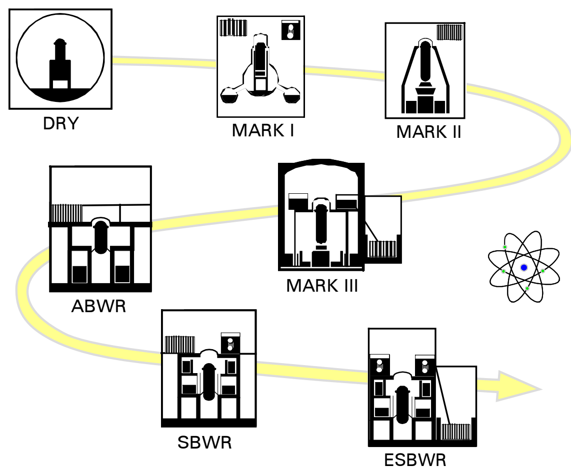


Figure 1-2. Evolution of the Containment Design

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

ESBWR Development and Design Approach

Following the Three Mile Island accident in 1979, there was a lot of interest in developing a reactor with passive safety features and less dependence on operator actions. Utilities also took this opportunity to request a reactor which was simpler to operate, had fewer components and no dependence on diesel-generators for safety actions. GE began an internal study of a new BWR concept based on these principles and the Simplified Boiling Water Reactor (SBWR) was born in the early 1980s. This concept attracted development support from the U.S. Department of Energy (DOE), EPRI and a number of US Utilities. Key new features, such as the Gravity Driven Core Cooling System (GDCS), Depressurization Valves (DPV), and leak-tight wetwell/drywell vacuum breakers were tested. As interest grew, an International Team was formed to complete the design, and additional separate effects, component and integrated system tests, particularly of the innovative new feature, the Passive Containment Cooling System (PCCS), were run in Europe and Japan.

A Design Certification Program was begun the late 1980s with the objective of obtaining a standardized license, similar to that obtained for the ABWR. However, as more of the design details became known, it became clear that, at 670 MWe, the SBWR was too small to be economically competitive with other Utility options for electrical generation. The certification program was stopped, but GE continued to look for ways to make an SBWR attractive for power generation. With European Utility support, the SBWR was uprated gradually to its current power level of approximately 1550 MWe. This was made possible by staying within the Reactor Pressure Vessel (RPV) size limit established by the ABWR, and by taking advantage of the modular approach to passive safety afforded by Isolation Condensers (IC) and PCCS.

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

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

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

Related Projects Worldwide

Operating ABWRs in Japan

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

Both TEPCO ABWR units were constructed in world record times. From first concrete to fuel load, it took just 36.5 months to construct Unit 6 and 38.3 months for Unit 7, the former being 10 months less than the best time achieved for any of the previous BWRs constructed in Japan. In addition, both units were built 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 will achieve 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. Kashiwasaki Units 6 & 7 (K5 to the right)

The ABWR in the United States

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

The ABWR in Taiwan

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

Commercial operation of Lungmen Unit 1 is expected to begin in July 2009. The schedule for Unit 2, including the start of commercial operation, is one year later.

ESBWR in the U.S.

The Design Certification application for the ESBWR was submitted to the U.S. Nuclear Regulatory Commission (NRC) in August 2005 and with one of the most thorough acceptance review processes conducted, was formally accepted for docketing in just 3 months. Initial scheduling between the U.S. NRC and GE, identifies completion of the preliminary

Safety Evaluation Report (SER) by 2007, which fits with current U.S. utility plans to submit Combined Construction and Operating License (COL) applications in 2007 and 2008, based on GE's ESBWR technology. The new plant review and licensing process has been improved, including allowance for parallel review of Design Certification and COL, with a focus on standardization, and reducing and eliminating re-reviews of the same open items. Based on recent licensing experience, Final Design Approval can be expected about 15 months after the preliminary SER (or around December 2008), and formal Design Certification is typically 12 months after that time frame (or around December 2009).

GE is participating with NuStart and Dominion Resources Inc. (who both selected ESBWR technology) in the Department of Energy's (DOE) Nuclear Power 2010 program, which was established by the DOE to act as a catalyst for new build nuclear energy in the U.S., and therefore help the U.S. meet long-term demand for electrical power generation. The preliminary SER is expected no later than October 2007, with a number of utilities preparing ESBWR COLs for submittal in 2007 and 2008. Once approved, a COL allows a utility to commence construction, followed by plant startup and commercial operation. Based on current schedules, this could mean operational ESBWR plants in the U.S. as early as 2014 and 2015.

Nuclear Plant Projects in the New Millennium

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

Design and Licensing

The ESBWR nuclear plant will be licensed and designed in its entirety prior to the start of construction. 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 ESBWR concept has been designed to higher levels of safety, including being designed to prevent and mitigate the consequences of a Severe Accident.

The ESBWR design will be captured electronically using the latest state-of-the-art information management technology. The benefits appear not only in construction, where it has been shown over and over with fossil plants that use of this engineering tool reduces construction time and cost, but also during the operation and maintenance of the plant. The approach described above is being fully utilized for the Lungmen ABWR project.

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 ESBWR is palnned to be built in only three years, from first concrete to the start of commercial operation. Design simplifications and the use of new construction technologies and techniques make this possible.

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.

Chapter 2
Plant Overview

ESBWR Program Goals

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

- Design life of 60 years.
- Plant availability factor of 87% or greater.
- Less than one unplanned scram per year.
- 18 to 24-month refueling interval.
- Operating personnel radiation exposure limit <1 Sv/year.
- Reduced calculated core damage frequency by at least a factor of 10 over previous BWRs (goal $<10^{-6}$/yr).
- Radwaste generation less than that of the 10% best operating BWRs
- 48-month construction schedule.
- 20% reduction in capital cost ($/kWh) vs. previous 1100 MWe class BWRs.

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

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

Summary of the ESBWR Key Features

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

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

Design Philosophy

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

By shortening the active fuel length, adding an approximately 9 m tall chimney above the core and lengthening the reactor vessel, the ESBWR eliminated the recirculation system, relying completely on natural circulation for core flow (see Figure 2-3). High pressure inventory control and heat removal

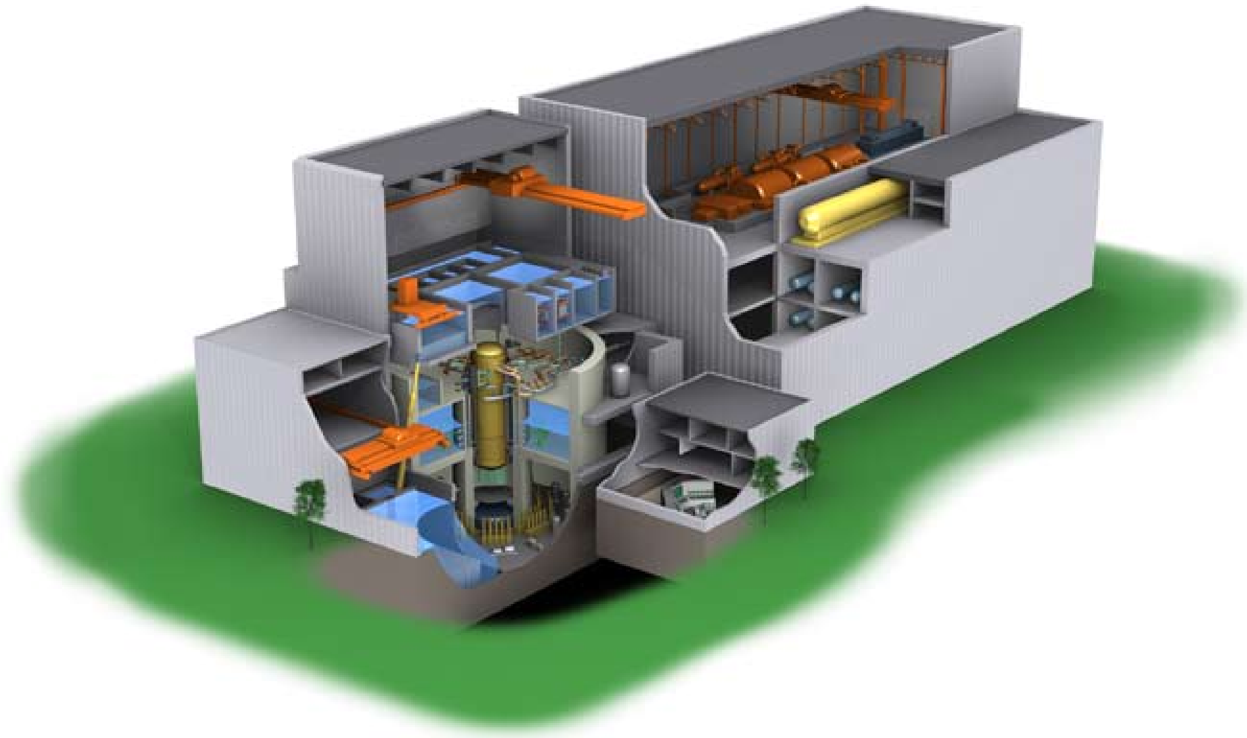


Figure 2-1. Cutaway Rendering of the ESBWR

is accomplished with the use of isolation condensers if the reactor becomes isolated from the normal heat sink.

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

Response to anticipated 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 feedwater runback and passive Standby Liquid Control System (SLCS) injection from borated water stored in pressurized accumulators.

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

Improvements to Operation and Maintenance

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

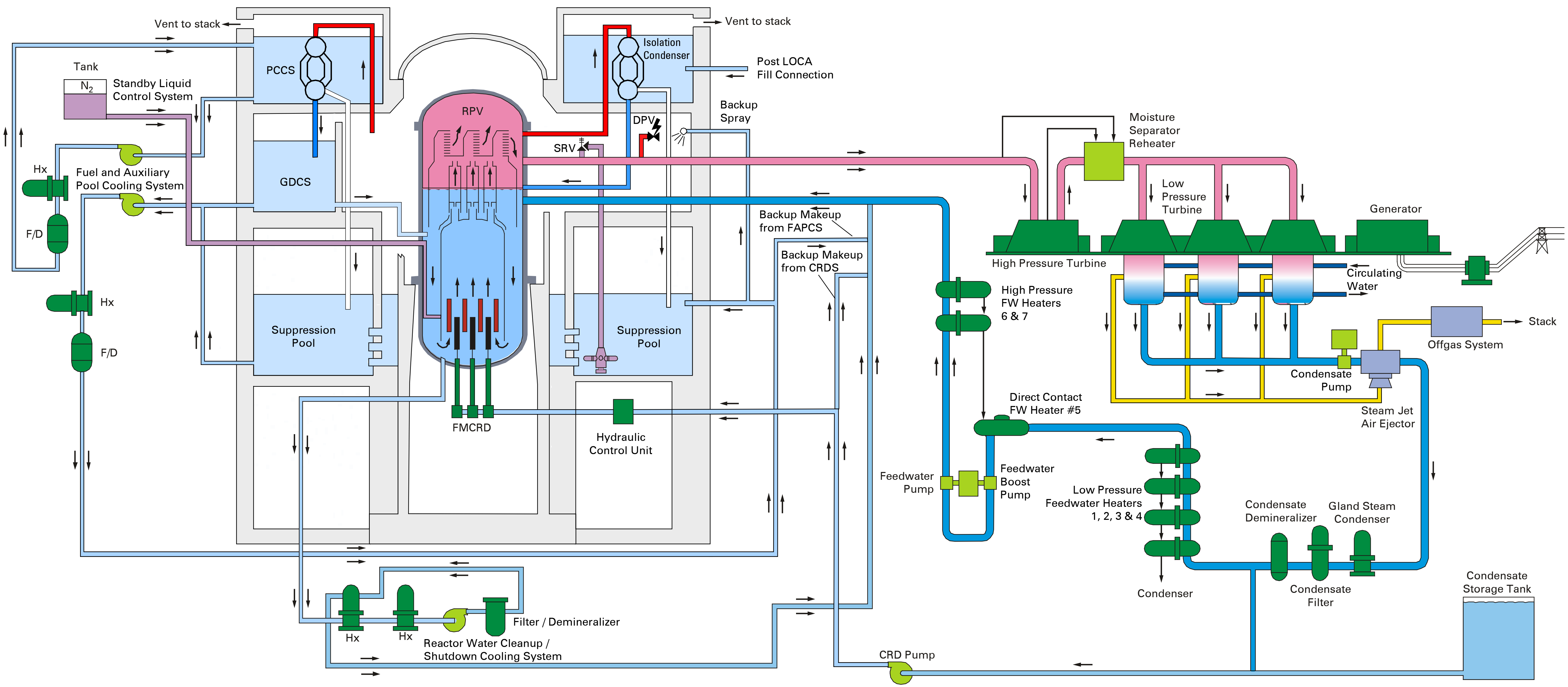


Figure 2-2. ESBWR Major Systems

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

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

plant, is focused on improved O&M.

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

The FMCRDs permit a number of simplifications. First, scram discharge piping and scram discharge volumes (SDVs) were eliminated, since the hydraulic scram water is discharged into the reactor vessel. By supporting the drives directly from the core plate, shootout steel located below the reactor vessel to mitigate the rod ejection accident was eliminated. The number of hydraulic control units (HCUs) was reduced by connecting two drives to each HCU, as was done on the ABWR. The number of rods per gang was increased up to 26 rods, greatly improving reactor startup times. Finally, since there

Figure 2-3. ESBWR Reactor Pressure Vessel and Internals

are no organic seals, only two or three drives will be inspected per outage, rather than the 30 specified in most current plants.

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

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

Lessons learned from operating experience were applied to the selection of ESBWR materials. Stainless steel materials which are qualified as resistant to intergranular stress corrosion cracking (IGSCC) were used. In areas of high neutron flux, materials were also specially selected for resistance to irradiation-assisted stress corrosion cracking (IASCC). Hydrogen Water Chemistry (HWC) is recommended for normal operation to further mitigate any potential for stress corrosion cracking.

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

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

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

Within the containment itself, no equipment requires servicing during plant operation and the

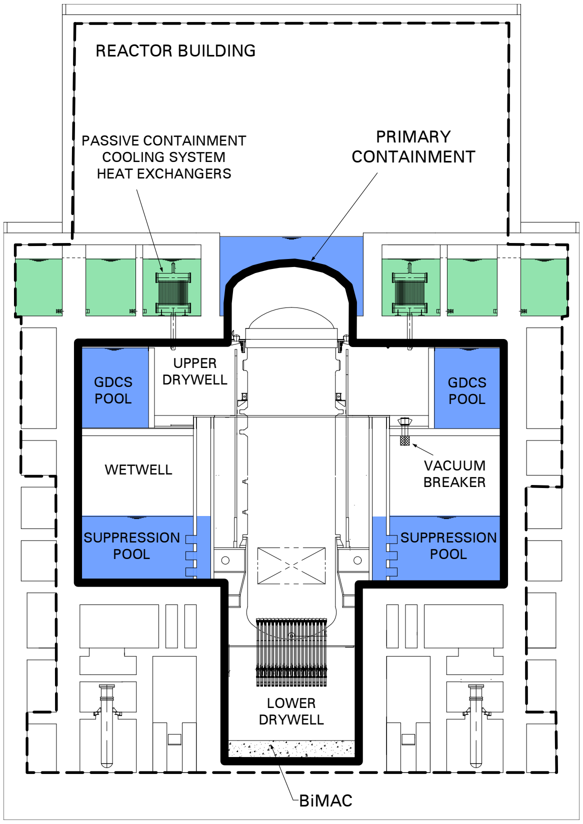


Figure 2-4. ESBWR Reactor Building and Containment

amount of equipment that requires maintenance during outages is significantly reduced. The containment is significantly smaller than that of the preceding BWR/6, but about the same size as ABWR. However, primarily due to the elimination of the recirculation system, there is actually more room to conduct maintenance operations. To simplify maintenance and surveillance during scheduled outages, permanently installed monorails and platforms permit 360° access, and both the upper and lower drywells have separate personnel and equipment hatches. To simplify FMCRD maintenance, a rotating platform is permanently installed in the lower drywell, and semi-automated equipment was specially designed to remove and install that equipment. The wetwell area is compact and isolated from the rest of containment, thus minimizing the chance for suppression pool contamination with foreign material.

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

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

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

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

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 ESBWR. Increased system redundancies will

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

also permit on-line maintenance. Thus, both forced outages and planned maintenance outages will be significantly reduced.

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

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

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

Chapter 3 Nuclear Steam Supply Systems

Overview

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

Reactor Vessel and Internals

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

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

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

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

The vessel contains the core support structure that extends to the top of the core. The presence of a large volume of steam and water results in two very important and beneficial characteristics. 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 ESBWR pressure rise that would occur after a rapid isolation of the reactor from the normal heat sink.

The following sections provide further descrip-

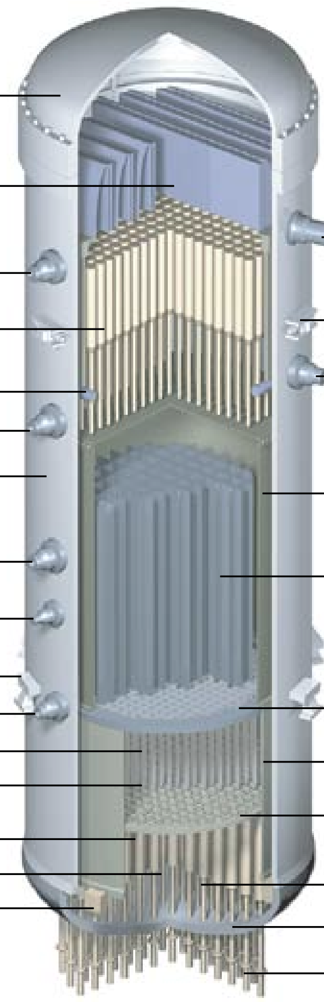


Figure 3-1. ESBWR Reactor Assembly

tions of the unique features of the ESBWR RPV and internals.

RPV Closure Head (1)[1]

The RPV closure head is elliptical in shape and is fabricated of low alloy steel, per ASME SA-508, Grade 3, Class 1. It is secured to the RPV by 80 sets of fasteners (studs and nuts). These nuts are tightened in groups of (typically) four at a time, using an automatic or semiautomatic four-stud tensioner device. The vessel closure seal consists of two concentric O-rings which perform without detectable leakage at all operating conditions, including hydrostatic testing.

Steam Nozzle with Flow Restrictor (2)

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

Feedwater Nozzle Thermal Sleeve (3)

There are three feedwater nozzles for each of the two feedwater lines which utilize double thermal sleeves welded to the nozzles. The double thermal sleeve protects the vessel nozzle inner blend radius

1. Numbers refer to Figure 3-1.

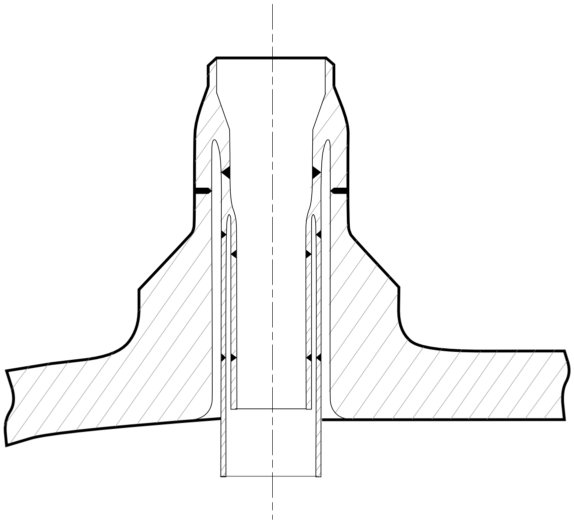


Figure 3-2. ESBWR Reactor Pressure Vessel Feedwater Nozzle

from the effects of high frequency thermal cycling. A schematic of the feedwater nozzle is shown in Figure 3-2.

Feedwater Spargers (4)

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

Vessel Support (5)

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

- Openings to permit water to pass from the upper to lower drywell
- Access for ISI of the bottom head weld

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

Reactor Vessel Bottom Head (6)

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

Stabilizers (7)

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

Forged Shell Rings (8)

The ESBWR RPV utilizes low alloy forged shell rings, adjacent to and below the core belt line region. The flanges and large nozzles are also low alloy steel. The shell rings above the core beltline region and the RPV closure head are made from low alloy steel forgings or plate per ASME SA-533, Type B, Class 1. The required Reference Nil Ductility Temperature, RT_{NDT}, of the vessel material is -20°C. Figure 3-3 shows one of the RPV forged shell rings during fabrication of an ABWR vessel.

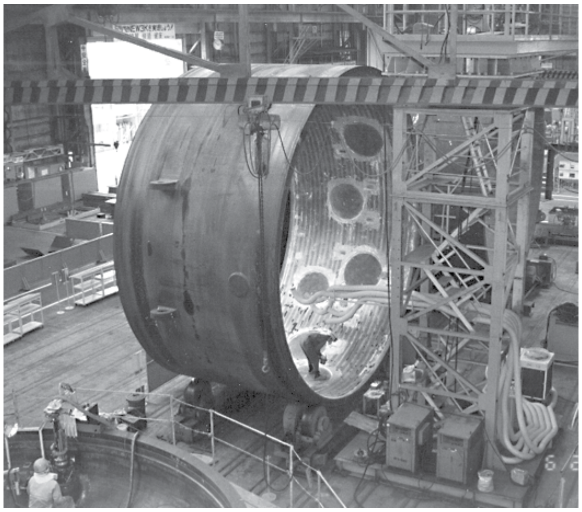


Figure 3-3. ABWR RPV Forged Steel Ring

Core Shroud (9)

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

Shroud Support Brackets (10)

There are 12 thick vertical support brackets welded to the vessel wall near the bottom inside region of the RPV cylindrical portion. Besides the weight of the shroud, these brackets support the weights of the core plate, fuel and fuel supports, top guide, chimney, chimney partitions and the steam separator assembly.

Core Plate (11)

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

Top Guide (12)

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

Fuel Supports (13)

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

Control Rod Drive Housing (14)

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

Control Rod Guide Tubes (15)

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

In-Core Housing (16)

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.

Chimney (17)

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

Chimney Partitions (18)

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

Steam Separator Assembly (19)

The steam separator assembly consists of a flat base on top of which is welded an array of standpipes with a three-stage steam separator located at the top of each standpipe. The steam separator assembly rests on the top flange of the chimney and forms the cover of the core discharge plenum region. The separator assembly is bolted to the chimney flange by long hold down bolts which, for ease of removal, extend above the separators. During installation, the separator base is aligned on the chimney flange with guide rods and finally positioned with locating pins. The objective of the long-bolt design is to provide direct access to the bolts during reactor refueling operations with minimum-depth underwater tool manipulation during the removal and installation of the assemblies. It is not necessary to engage threads in mating up the shroud head. A tee-bolt engages in the chimney flange and its nut is tightened to only nominal torque. Final loading is established through differential expansion of the bolt and compression sleeve. The fixed axial flow type steam separators have no moving parts and are made of stainless steel. In each separator, the steam-water mixture rising through the standpipe impinges on vanes which give the mixture a spin to establish a vortex wherein the centrifugal forces separate the water from the steam in each of three stages. Steam leaves the separator at the top and passes into the wet steam plenum below the dryer (Figure 3-4). The separated liquid exits from the lower end of each stage of the separator and enters the pool that surrounds the standpipes to join the downcomer annulus flow.

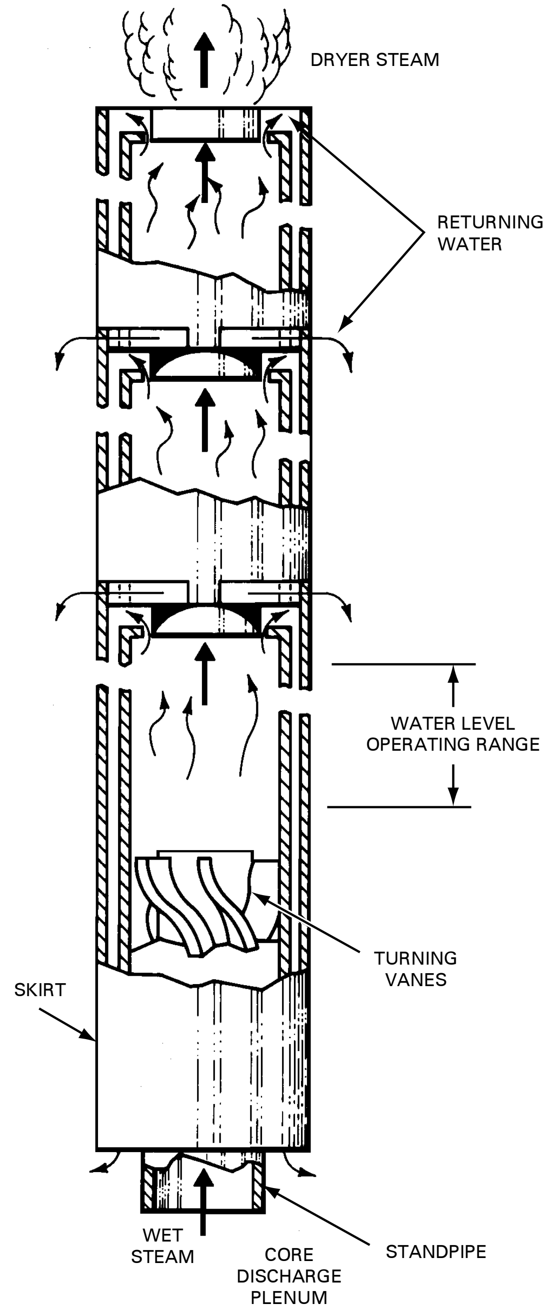


Figure 3-4. Schematic of Steam Flow Through Separator

Steam Dryer Assembly (20)

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 steam separators flows upward to the steam dryer and outward through the drying vanes (Figure 3-5). These vanes are attached to a top and bottom supporting member forming a rigid, integral unit. Moisture is removed and carried by a system of troughs and drains to the pool surrounding

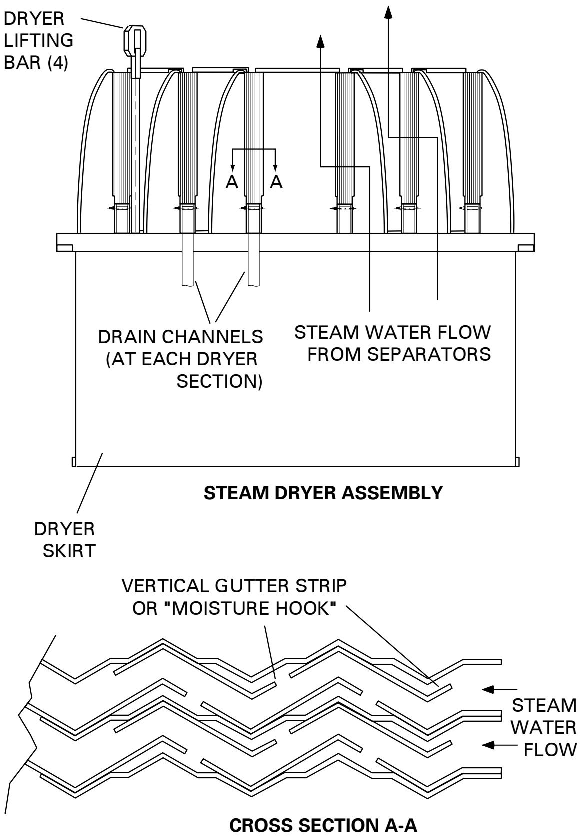


Figure 3-5. Schematic of Steam Flow Through Dryer

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

DPV/IC Outlet (21) and IC Return (22)

There are four 450 mm nozzles spaced around the RPV for connection to each of the four isolation condenser (IC) subsystems and to four of the depressurization valves (DPV). The IC return line nozzles are 200 mm diameter.

GDCS Inlet (23)

There are eight 150 mm nozzles spaced around the RPV for connection to each of the four divisions of the GDCS injection lines. In addition, four 150 mm flow restrictors are designed into each nozzle to limit the flow in the event of a postulated GDCS line break.

GDCS Equalizing Line Inlet (24)

There are four 150 mm nozzles spaced around the RPV for connection to each of the four divisions of the GDCS equalizing lines. Flow restrictors are designed into each nozzle to limit the flow in the event of a postulated GDCS equalizing line break.

RWCU/SDC Outlet (25)

There are two 300 mm nozzles provided for connection to each of the trains of the RWCU/SDC system.

Control Rod Drive System

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

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

The CRD System consists of three major elements:

- Electrohydraulic fine motion control rod drive (FMCRD) mechanisms.
- Hydraulic control unit (HCU) assemblies

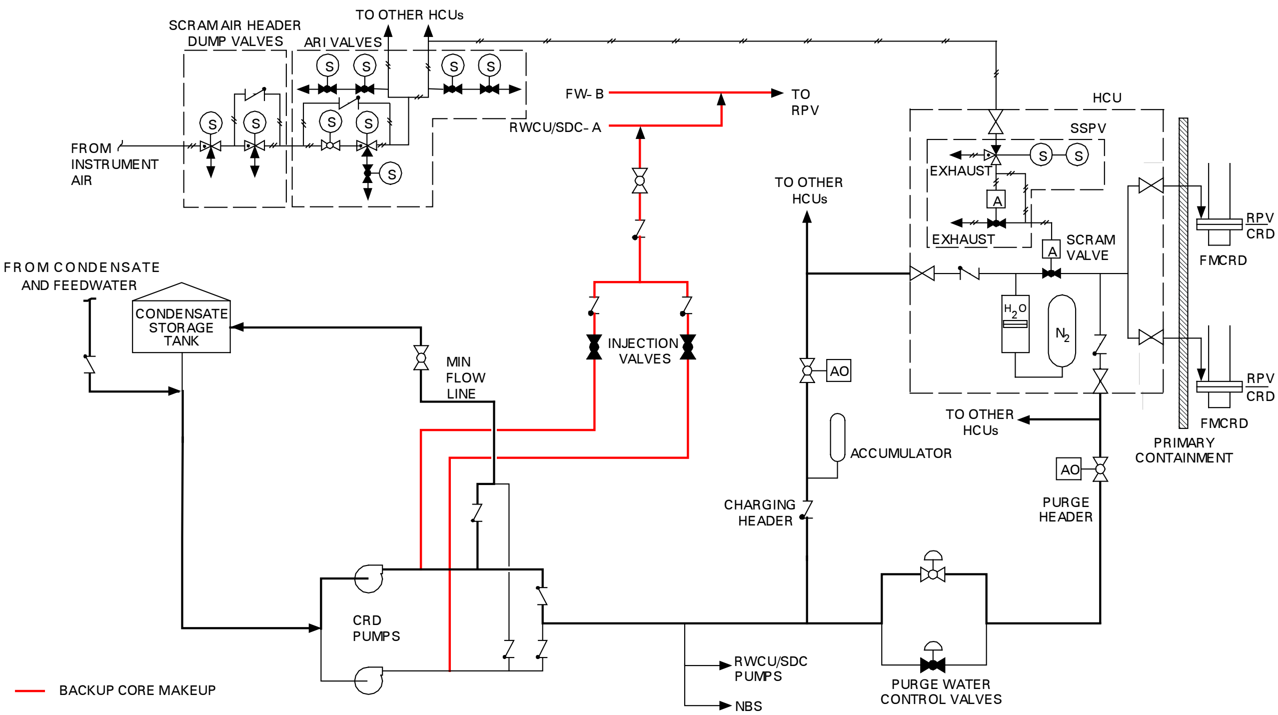


Figure 3-6. CRD System Schematic

- Control Rod Drive Hydraulic System (CRDHS)

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

Fine Motion Control Rod Drives

The ESBWR FMCRDs are distinguished from the locking piston CRDs (which are in operation in almost 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, and is basically the same drive design that is in use in ABWR.

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

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

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

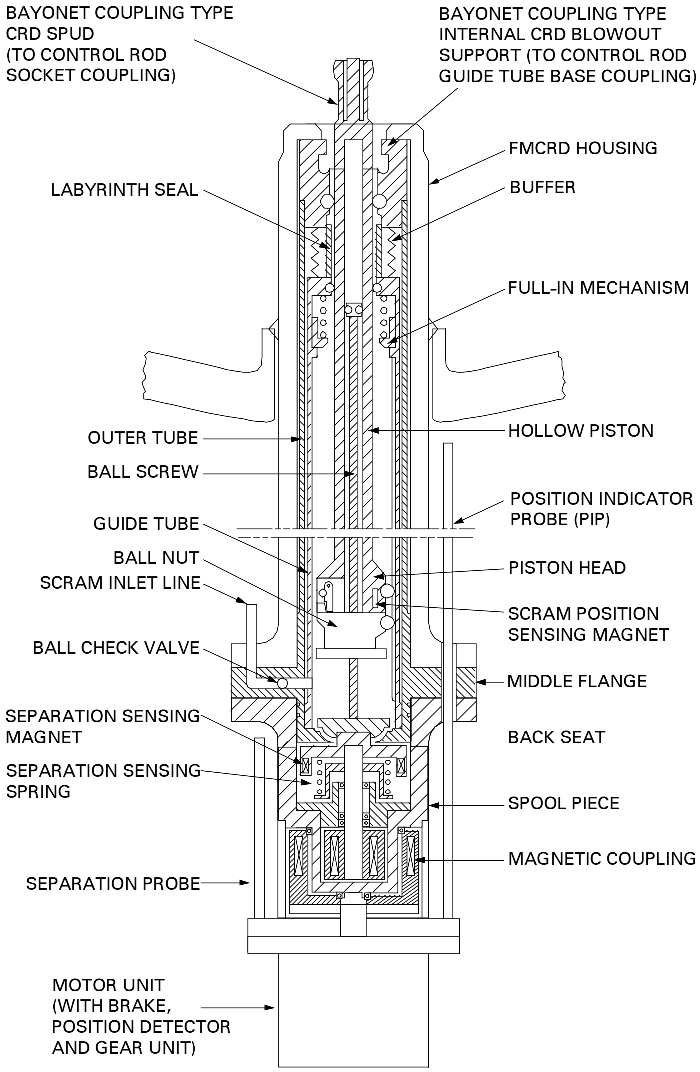


Figure 3-7. Fine Motion Control Rod Drive Cross-Section

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

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

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

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

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

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

The spool piece also contains a weighing device, which is a spring-loaded platform with two magnets located on it. In normal service, 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. Thus, the reactivity addition from a postulated Rod Drop Accident is limited to a few cents, and this event is no longer analyzed in Safety Analysis Reports.

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

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

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

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

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

The probes use reed switches similar to the Locking Piston Control Rod Drive (LPCRD), as do the separation switch probes that are mounted on the side of the spool piece. The separation probes and associated circuits and equipment are considered important to safety and are therefore categorized as Class 1E.

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

Induction Motor Control (IMC) equipment in the Rod Control & Information System (RC&IS) provides the control power to the FMCRDs for performing normal control rod movements. The IMC equipment provides for AC phase direction change of the 3-Phase AC power provided to each AC induction motor so that both insertion and withdrawal movements can be accomplished.

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

Hydraulic Control Units

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

Control Rod Drive Hydraulic System

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

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

Nuclear Boiler System

The purpose of the Nuclear Boiler System (NBS) is to direct steam flow from the RPV steam outlet nozzles to the main turbine. A main steam line flow restrictor is provided in each steam outlet nozzle. It is designed to limit the flow rate in the event of a postulated steam line break. The system incorporates provisions for relief of overpressure conditions in the RPV. Also included in the NBS is the Nuclear Island portion of the Feedwater System.

Main Steam Subsystem (MS)

In the ESBWR design, four 700 mm steam lines transport steam from the steam outlet nozzles on the RPV through Reinforced Concrete Containment Vessel (RCCV) penetrations and then through the steam tunnel to the turbine. Main steam isolation valves (MSIVs) are installed in each steam line inboard and outboard of the RCCV penetrations. Eighteen safety/relief valves (SRVs) are installed vertically on the main steam lines. Of the 18 SRVs, 10 provide the Automatic Depressurization System (ADS) function during an accident condition, and the discharge from each SRV is routed through the associated SRV discharge line to quenchers located in the suppression pool. The remaining 8 SRVs are spring-actuated only to provide overpressure protection in the case of a postulated Anticipated Transient Without Scram (ATWS) event. The discharge from these valves is routed into two discharge headers located in the drywell equipped with rupture disks. Each discharge header is also routed to the suppression pool with a discharge line terminating in a quencher.

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

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

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

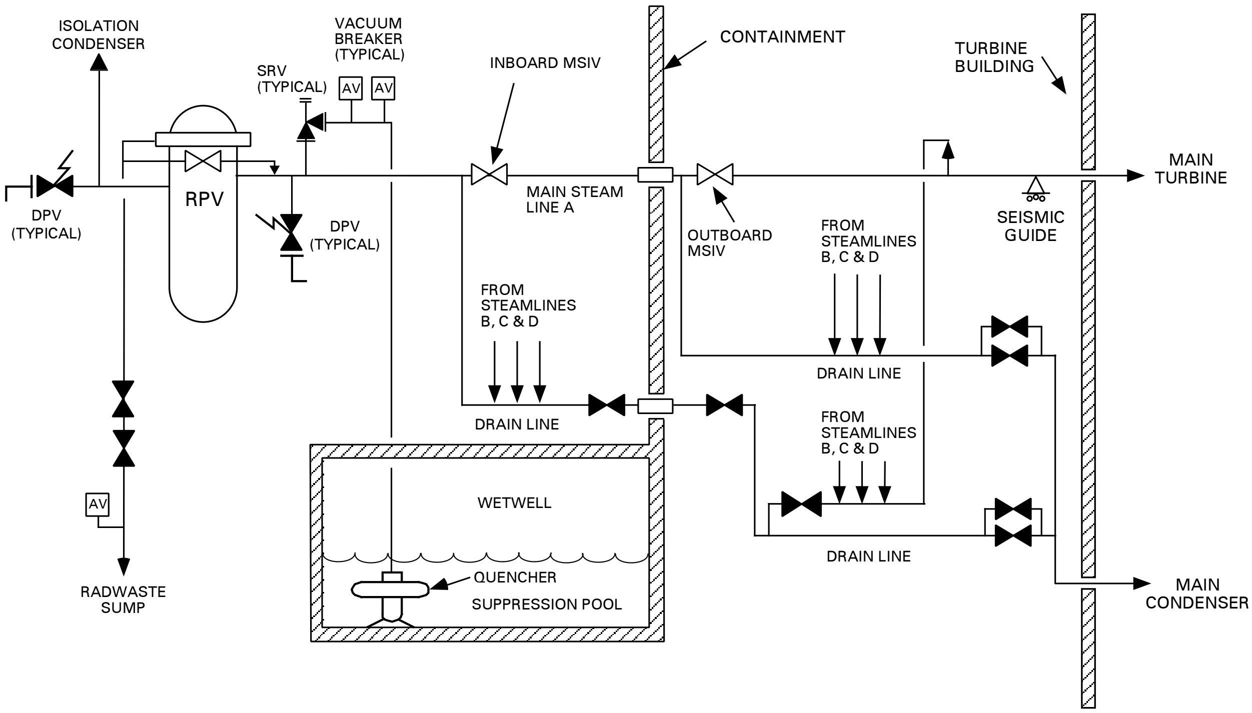


Figure 3-8. Main Steam Subsystem

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 3-9). 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 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 a pneumatic 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 MSIVs 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 steam lines 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

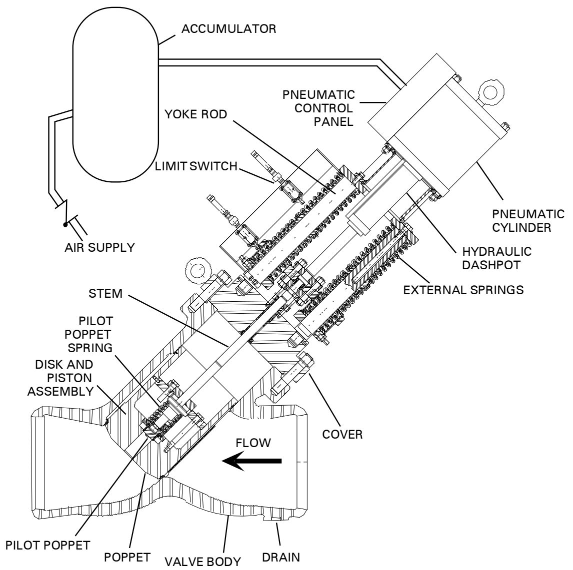


Figure 3-9. Main Steam Isolation Valve

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

A SRV is in principle a dual function, direct-acting valve and is classified as safety-related (Figure 3-10). In the ESBWR 10 valves (ADS-SRV) include the necessary accumulators and pneumatic actuators necessary for automatic or manual actuation in addition to opening on spring pressure (dual function). The other 8 SRVs have the same body, but do not include the extra actuators and open on spring pressure only (direct acting). The SRV is considered as part of the RCPB because the inlet side of the valve is connected to the steam line prior to the inboard MSIV. The ADS-SRV logic and two of the 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. The third solenoid is classified as non-Class 1E.

Due to the use of the IC System in ESBWR, SRVs are not normally needed to maintain primary system pressure below the ASME Code design

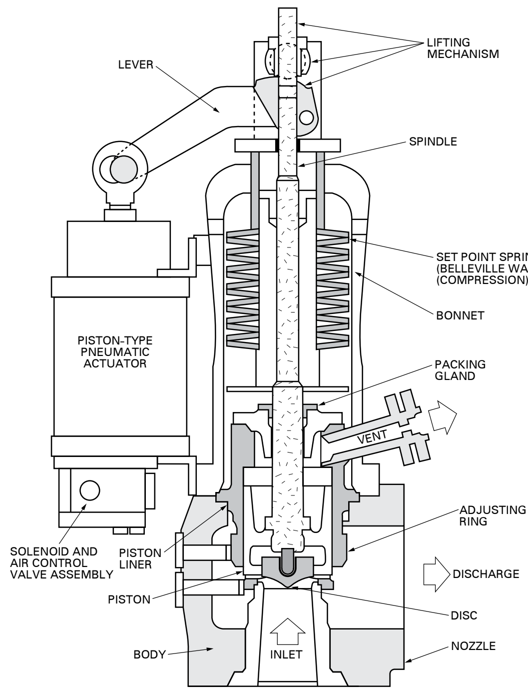


Figure 3-10. Safety Relief Valve With Pneumatic Actuator

limits.

The SRVs are located on the main steam lines between the RPV and the inboard MSIV. These valves provide two main protection functions:

- **Overpressure Safety Operation:** The 18 valves function as spring-loaded safety valves and open to prevent RCPB overpressurization. The valves are self-actuated by inlet steam pressure. This will occur only during certain ATWS events.

- **Automatic Depressurization System (ADS) Operation:** The 10 ADS-SRV valves are opened using a pneumatic actuator upon receipt of an automatic or manually initiated signal at the solenoid valve located on the pneumatic actuator assembly. 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. They are opened automatically or manually in the power actuated mode when required during a LOCA. The ADS designated SRVs open automatically as part of the Emergency Core Cooling System (ECCS) as required to mitigate a LOCA when it becomes necessary to reduce RCPB pressure to admit low pressure ECCS coolant flow to the reactor.

The SRVs are divided into two spring setpoint groups to relieve the RPV pressure in accordance with the RPV overpressure protection evaluations for ATWS.

A pneumatic accumulator is provided for each ADS-SRV function and is located close to each SRV to supply pressure for the purpose of valve actuation.

The ADS-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 ADS-SRV has its own discharge line with two vacuum breakers. The discharge lines are sized so that the critical flow conditions occur through the valve. This prevents the conditions in the discharge lines of water hammer and pressure instability. For the ADS-SRVs, the SRV discharge lines terminate at the quenchers located below the surface of the suppression pool (SP). The remaining 8 SRVs discharge lines are routed to two headers in the drywell which are equipped with rupture disks. Each header has a discharge line with two vacuum breakers, sized for the steam relief of one SRV, which terminates in a quencher located below the surface of the SP.

Depressurization Valves

There are eight DPVs, 4 located on the main steam lines and 4 on stub lines (Figure 3-11), whose sole purpose is to aid the ADS subsystem in rapidly reducing RCPB pressure during a LOCA in order for the low pressure ECCS to add water to the RPV.

The DPVs are of a non-leak/non-simmer/non-maintenance design. They are straight-through, squib-actuated, non-reclosing valves with a metal

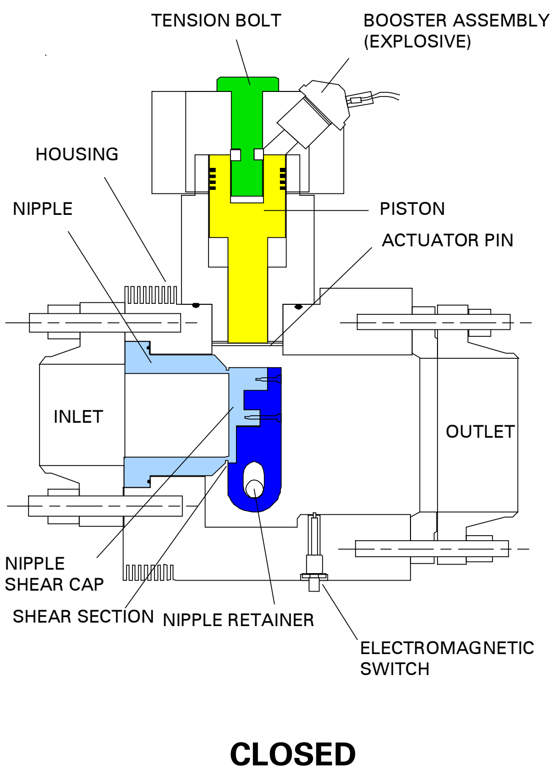


Figure 3-11. Depressurization Valve

diaphragm seal. The valve size provides about twice the depressurization capacity as an SRV. Each DPV is closed with a cap covering the inlet chamber. The cap shears off when pushed by a valve plunger that is actuated by the explosive initiator-booster. This opens the inlet hole through the plug. The sheared cap is hinged such that it drops out of the flow path and does not block the valve. The DPVs are designed so that there is no leakage across the cap throughout the life of the valve.

Two initiators (squibs), singly or jointly, actuate a booster, which actuates the shearing plunger. The squibs are initiated by either one or both of, two battery-powered, independent firing circuits. The firing of one initiator-booster is adequate to activate the plunger.

The DPV has undergone engineering development testing using a prototype to demonstrate the proper operability, reliability, and flow capability of the design. Figure 3-12 shows the DPV test facility.

Figure 3-12. DPV Under Test

Figure 3-13 shows the SRV, DPV and MSIV locations on the main steam lines and stub lines.

Feedwater Subsystem (Nuclear Island)

Two 550 mm feedwater lines transport feedwater from the feedwater pipes in the steam tunnel through RCCV penetrations to horizontal headers in the upper drywell which have three 300 mm riser lines that connect to nozzles on the RPV (Figure 3-14).

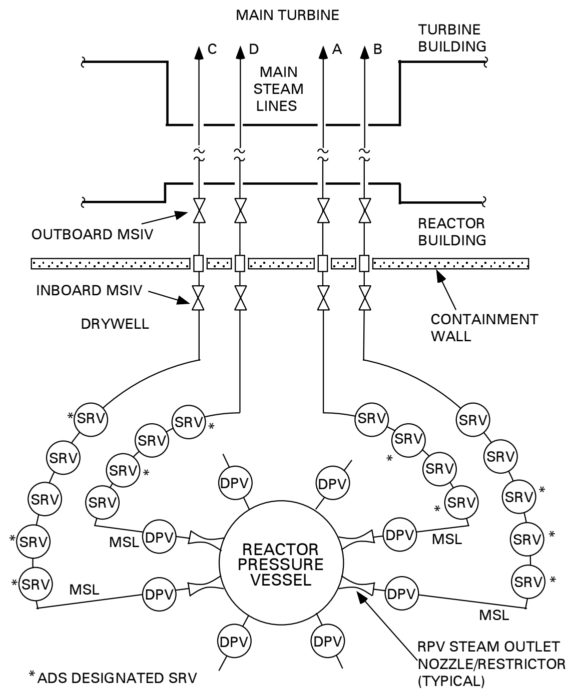


Figure 3-13. MSIV, SRV and DPV Configuration

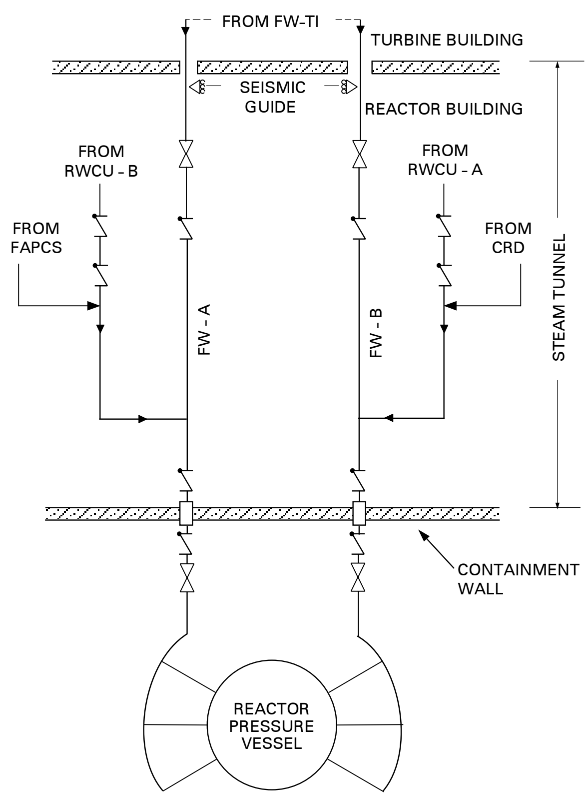


Figure 3-14. Feedwater Configuration (Nuclear Island)

Isolation check valves are installed upstream and downstream of the RCCV penetrations, and manual maintenance gate valves are installed in the 550 mm lines upstream of the horizontal headers. Also shown in the figure are the interconnections from the Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDC), CRD and Fuel and Auxiliary Pool Cooling System (FAPCS).

Isolation Condenser System

The primary function of the Isolation Condenser System (ICS) is to limit reactor pressure and prevent Safety Relief Valve (SRV) operation following an isolation of the main steam lines. The ICS, together with the water stored in the RPV, conserves sufficient reactor coolant volumes to avoid automatic depressurization caused by low reactor water level. The ICS removes excess sensible and core decay heat from the reactor, in a passive way and with minimal loss of coolant inventory from the reactor, when the normal heat removal system is unavailable. The ICS is designed as a safety-related system to remove reactor decay heat following reactor shutdown and isolation. It also prevents unnecessary reactor depressurization and operation of ECCS, which can also perform this function.

The ICS consists of four totally independent trains, each containing an isolation condenser (IC) that condenses steam on the tube side and transfers heat to a large IC/PCCS pool positioned immediately outside the containment, which is vented to the atmosphere as shown on the ICS schematic (Figure 3-15). The IC, connected by piping to the reactor pressure vessel, is placed at an elevation above the source of steam (vessel) and, when the steam is condensed, the condensate is returned to the vessel via

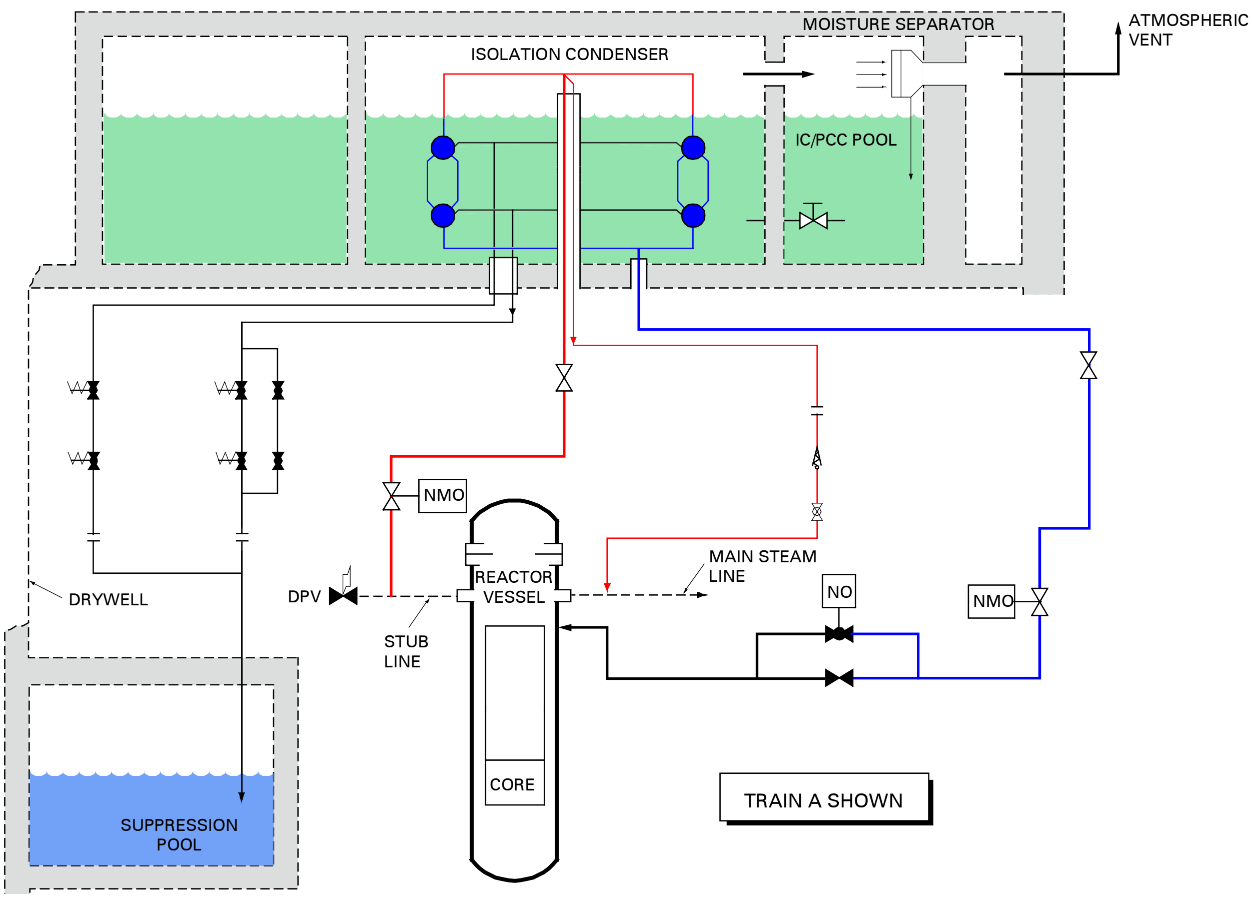


Figure 3-15. Isolation Condenser System (Standby Mode)

a condensate return pipe. The steam side connection between the vessel and the IC is normally open and the condensate line is normally closed. This allows the IC and drain piping to fill with condensate, which is maintained at a subcooled temperature by the pool water during normal reactor operation. The IC is started into operation by opening condensate return valves and draining the condensate to the reactor, thus causing steam from the reactor to fill the tubes which transfer heat to the cooler pool water. Each IC is made of two identical modules.

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

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

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

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

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

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

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

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

The IC has undergone engineering development testing using a prototype to demonstrate the proper operability, reliability, and heat removal capability of the design. Figure 3-16 shows one of the modules under test.

Figure 3-16. IC Test Module

Chapter 4
Safety Systems

Overview

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

The RPV has no external recirculation loops or large pipe nozzles below the top of the core region. This, together with a high capacity ADS allowed the incorporation of an ECCS driven solely by gravity, not needing any pumps. The water source needed for the ECCS function is stored in the containment upper drywell, with sufficient water to insure core coverage to 1 meter above the top of active fuel as well as flooding the lower drywell.

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

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

In addition to the systems mentioned above, there are other important safety systems in the ESBWR, including the Containment Inerting System (CIS) and the Emergency Breathing Air System (EBAS).

Emergency Core Cooling Systems

Gravity Driven Core Cooling

General

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

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

Each division of the long-term system consists

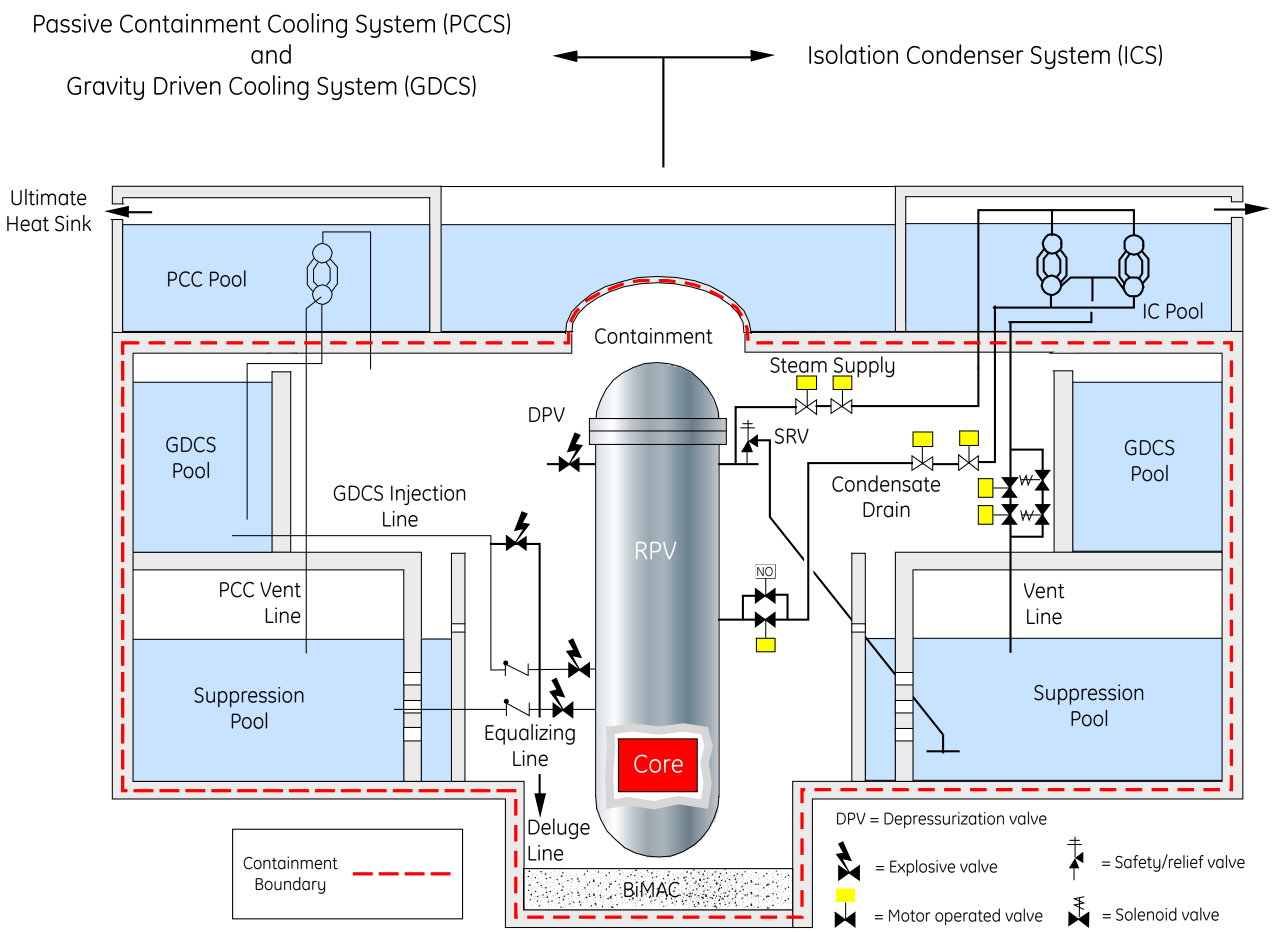


Figure 4-1. ESBWR Key Safety Systems

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

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

The check valves mitigate the consequences of spurious GDCS squib valve operation and minimize the loss of RPV inventory after the squib valves are actuated and the vessel pressure is still higher than the GDCS pool pressure plus its gravity head. Once the vessel has depressurized below GDCS pool surface pressure plus its gravity head, the differential pressure opens the check valve and allow water to begin flowing into the vessel.

The GDCS deluge lines provide a means of flooding the lower drywell region with GDCS pool water in the event of a postulated core melt sequence which causes failure of the lower vessel head and allows the molten fuel to reach the lower drywell floor. A core melt sequence would result from a common mode failure of the short-term and long-term systems, which prevents them from performing their intended function. Deluge line flow is initiated by thermocouples, which sense high lower drywell region basemat temperature indicative of molten

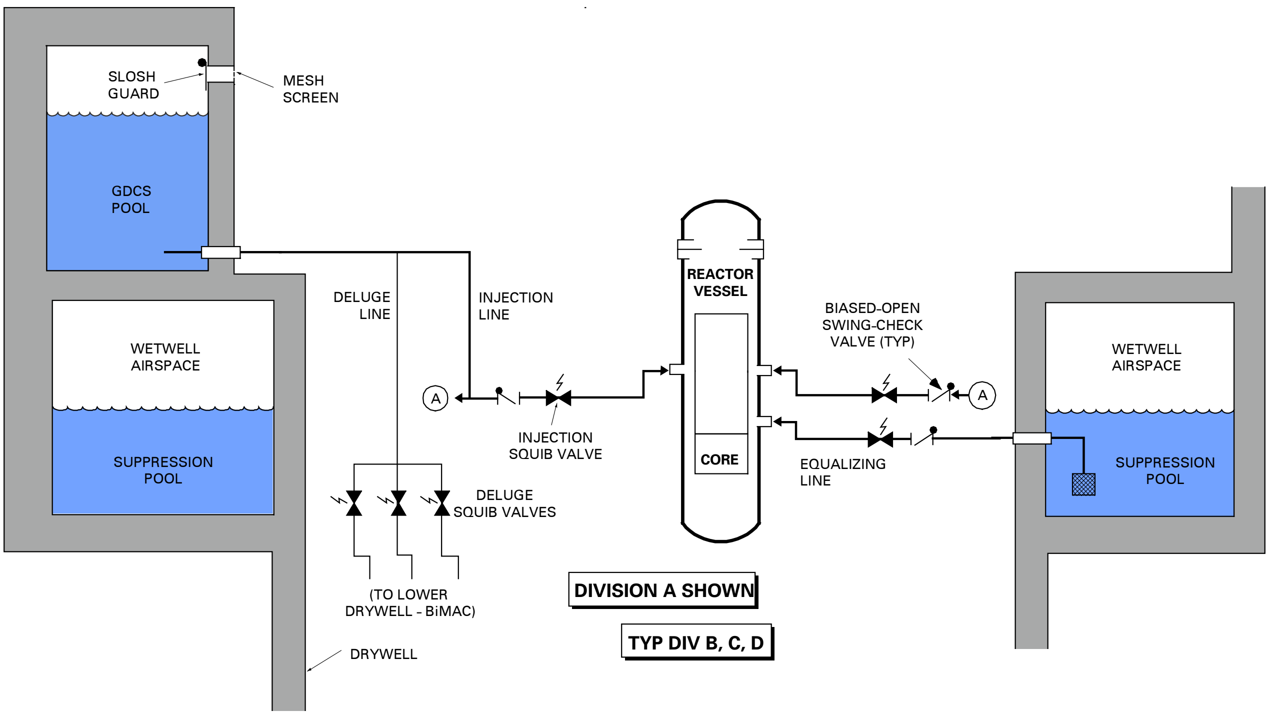


Figure 4-2. Gravity Driven Cooling System Schematic

fuel on the lower drywell floor. Squib-type valves in the deluge lines are actuated upon detection of high basemat temperatures. The deluge lines do not require the actuation of squib-actuated valves on the injection lines of the GDCS piping to perform their function.

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

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

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

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

Equipment and Component Description

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

Squib Valve

The function of the squib valve is to open upon an externally applied signal and to remain in its full open position without any continuing external power source in order to admit reactor coolant makeup into the reactor pressure vessel in the event of a LOCA. These valves also function in the closed position to maintain RPV backflow leak-tight and maintain the reactor coolant pressure boundary during normal plant operation. The valve is a horizontally mounted, straight through, long duration submersible, pyrotechnic actuated, non-reclosing valve with metal diaphragm seals and flanged ends. The valve design is such that no leakage is possible across the diaphragm seals throughout the 60-year life of the valve. The squib valve is classified as Quality Group A, Seismic Category I, and ASME Section III Class 1. The valve diaphragm forms part of the reactor pressure boundary and as such is designed for RPV service level conditions.

Illustrated in Figure 4-3 is a typical squib valve design that satisfies GDCS system requirements. This valve has similar design features to the ADS depressurization valve. Valve actuation initiates upon the actuation of either of two squib valve initiators, a pyrotechnic booster charge is ignited, and hot gases are produced. When these gases reach a designed pressure, a tension bolt holding a piston breaks allowing the piston to travel downward until it impacts the ram and nipple shear caps. Once the piston impacts the ram and nipple shear caps, the nipples are sheared. The ram and shear caps are then driven forward and are locked in place at the end of stroke by an interference fit with the nipple retainer. This lock ensures that the nipples cannot block the flow stream and provides a simple means of refurbishment by simply unthreading the plug. A switch located on the bottom of the valve provides a method of indication to the control room of an actuated valve. The shear nipple sections are designed to produce clean shear planes. The piston is allowed to backup after shearing the nipples, but in any case its forward motion is limited by the housing so that it will not create flow resistance. Standard metal seals are installed on the piston to reduce the potential of ballistic products from entering the flow stream. The squib valve can be completely refurbished once fired. The squib valve housing, nipples, adapter flanges, actuator housing, indicator switch body, indicator plunger, head cap, coupling,

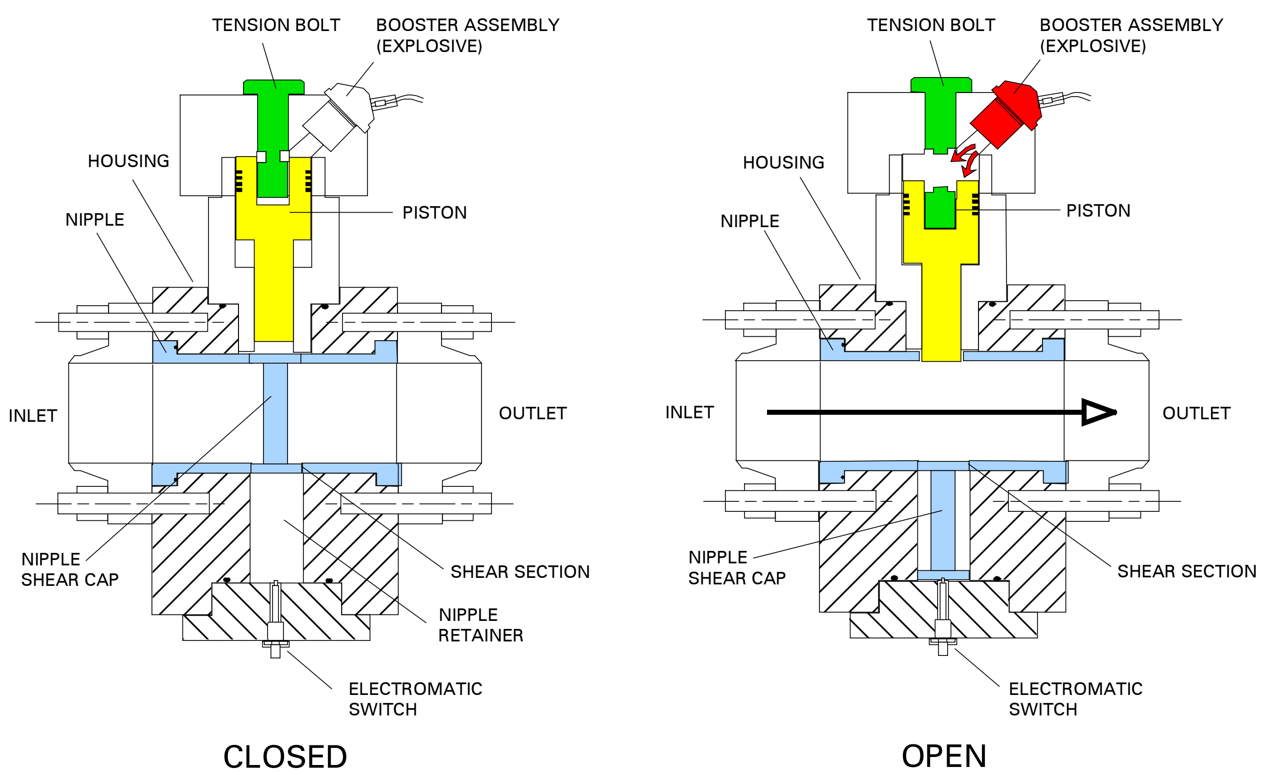


Figure 4-3. GDCS Squib Valve

collar and adapter are machined. The piston, ram, and tension bolt is made from heat treated material for necessary strength.

Biased-Open Check Valve

The GDCS biased-open check valves are designed such that the check valve active mechanism is off the seat when zero differential pressure is applied across the check valve. The check valve fully closes to prevent excess backflow when a low reverse differential pressure is applied across it. The biased-open check valve is a straight through, horizontally mounted, long duration submersible valve. The valve meets the minimum flow requirements for a valve stuck in the "valve biased" open position. The biased-open check valve illustrated in Figure 4-4 is a tilting disc check valve, biased open by gravity forces on the disc and counterweight. No torsion spring is used on the hinge pin and the hinge pin does not extend through any packing. The check valve is fabricated out of low carbon stainless steel. Remote check valve position indication is provided in the main control room by an eccentric ferromagnetic target on an extension from the tilting disc, which rotates within a non-magnetic (stainless steel) pressure shell. A micro switch proximity sensor locates the magnetic target when it is in the proper position. Two switches are required to differentiate full-closed and biased open indication. This indicator requires no pressure boundary penetration.

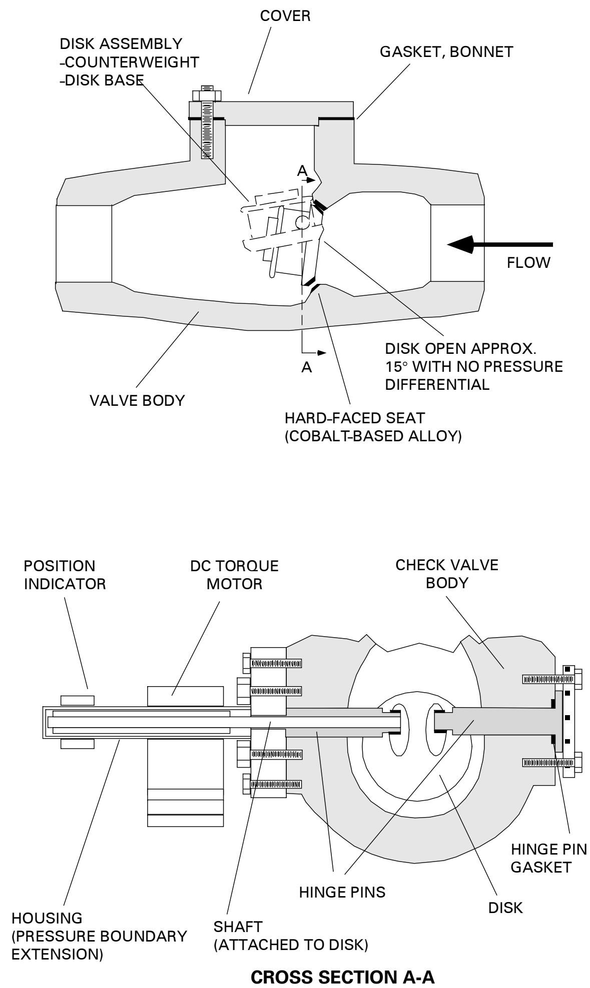


Figure 4-4. GDCS Check Valve

Deluge Valve

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

Automatic Depressurization System

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 high pressure backup systems (such as the CRD hydraulic system or parts of the Feedwater System) a chance to restore the RPV water level and thus avoid the ADS actuation.

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 Supply System (HPNSS). In addition to the 10 ADS-SRVs, the 8 Depressurization Valves (DPVs) are also initiated from one of two squibs located on each valve, controlled by the same trip channels. The ADS valve openings are staggered in time to control the blowdown rate and prevent excessive level swell.

The DC power for the logic and squib firing power 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. Description of the ADS valves can be found in Chapter 3.

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

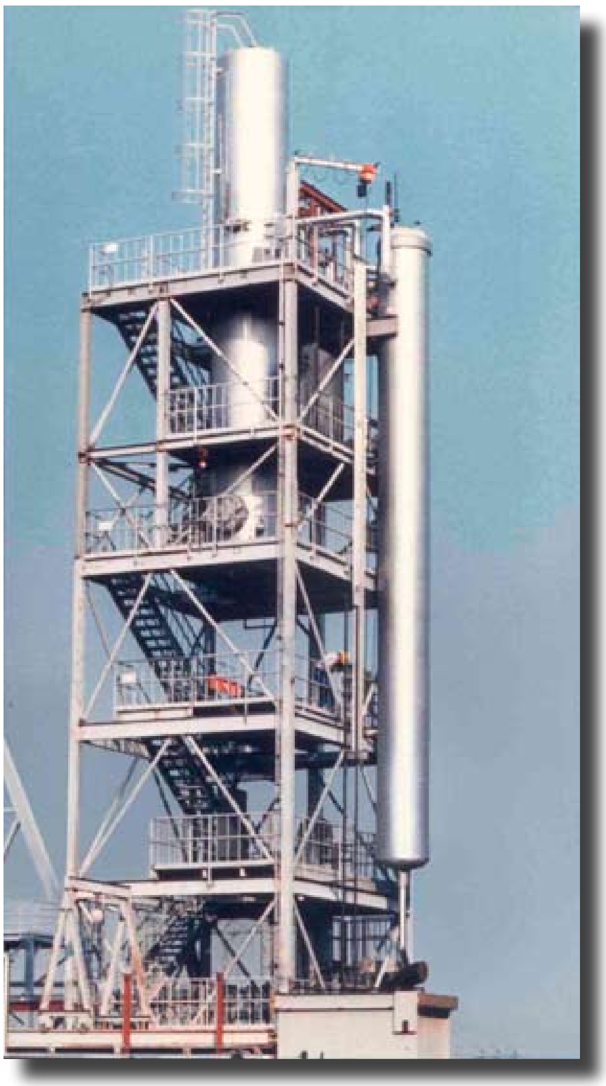


Figure 4-5. GIST Facility

Passive Containment Cooling System

The PCCS maintains the containment within its pressure limits for DBAs. The system is designed as a passive system with no components that must actively function, and it is also designed for conditions that equal or exceed the upper limits of containment severe accident capability. The PCCS consists of six, low-pressure, totally independent loops, each containing a steam condenser (Passive Containment Cooling Condenser), as shown Figure 4-6. Each PCCS condenser loop is designed for 11 MWt capacity and is made of two identical modules. Together with the pressure suppression containment, the PCCS condensers limit containment pressure to less than its design pressure for at least 72 hours after a LOCA without makeup to the IC/PCC pool, and beyond 72 hours with pool makeup. The PCCS condensers are located in a large pool (IC/PCC pool) positioned above, and outside, the ESBWR containment (DW).

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

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

The PCCS loops receive a steam-gas mixture supply directly from the DW. The PCCS loops are initially driven by the pressure difference created

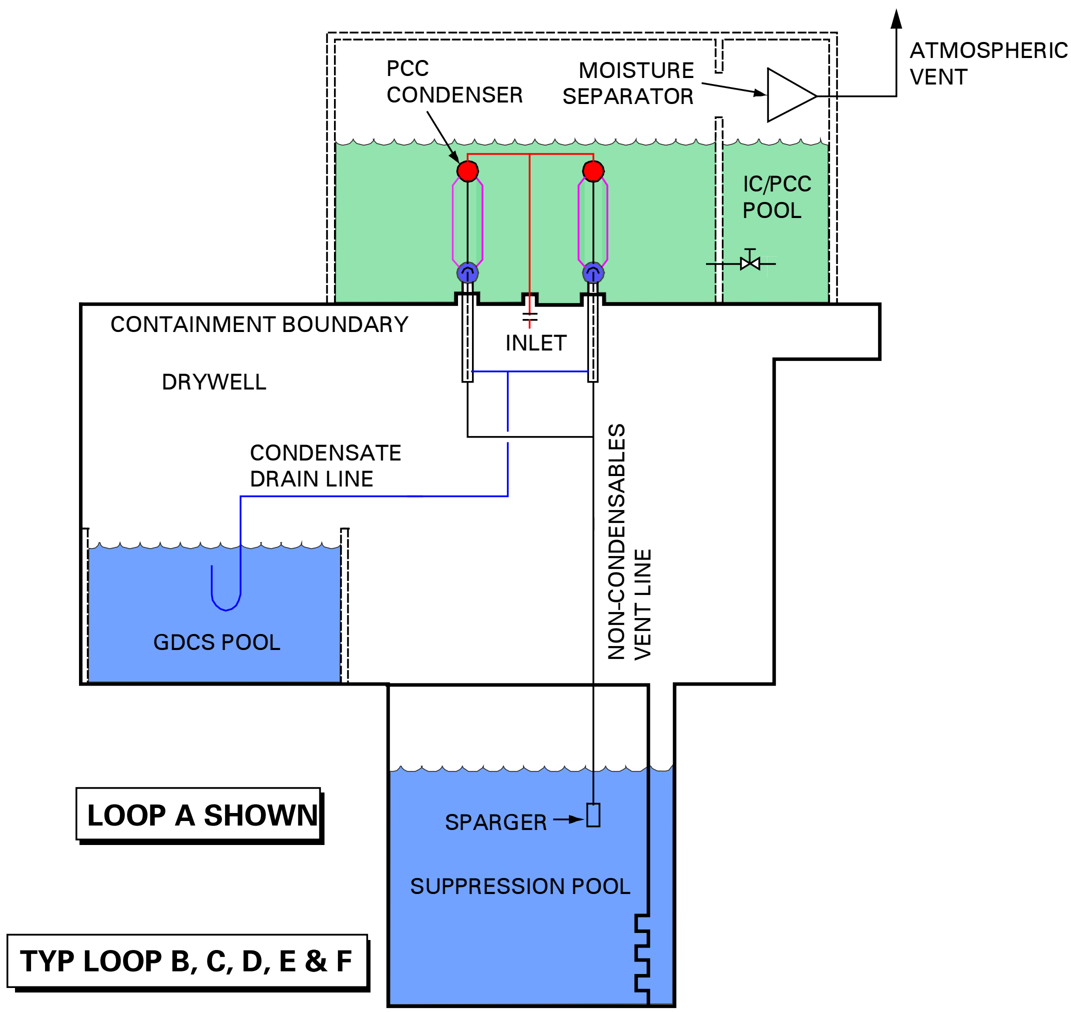


Figure 4-6. Passive Containment Cooling System Schematic

between the containment DW and the suppression pool during a LOCA and then by gravity drainage of steam condensed in the tubes, so they require no sensing, control, logic or power-actuated devices to function. The PCCS loops are an extension of the safety-related containment and do not have isolation valves.

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

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

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

Figure 4-7. PCCS Heat Exchanger Testing

Level control is accomplished by using an air-operated valve in the makeup water supply line. The valve opening and closing is controlled by water level signal sent by a level transmitter sensing water level in the IC/PCC pool. Cooling and cleanup of IC/PCC pool water is performed by the Fuel and Auxiliary Pools Cooling System (FAPCS) (see Chapter 5). The FAPCS provides safety-related dedicated makeup piping, independent of any other piping, which provides an attachment connection at grade elevation in the station yard outside the reactor building, whereby a post-LOCA water supply can be connected.

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

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

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

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

In addition to the accumulators and injection valves, supporting non-safety grade equipment includes a high pressure nitrogen charging system for pressurization and to make up for losses, and a mixing and boron solution makeup system.

The boron absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the fuel. The specified neutron absorber solution is sodium pentaborate using 94% of the isotope B^{10} at a concentration of 12.5%. This combination not only minimizes the quantity of liquid to be injected, but also assures no auxiliary heating is needed to prevent precipitation of the sodium pentaborate out of solution in the accumulator and piping. At all times, when it is possible to make the reactor critical, the SLCS will be able to deliver enough sodium pentaborate solution into the reactor to assure reactor shutdown.

Upon completion of injecting the boron solution, redundant accumulator level measurement instrumentation using 2 out of 4 logic closes the injection line shut-off valve in each SLCS division. Closure of these valves prevents injection of nitrogen

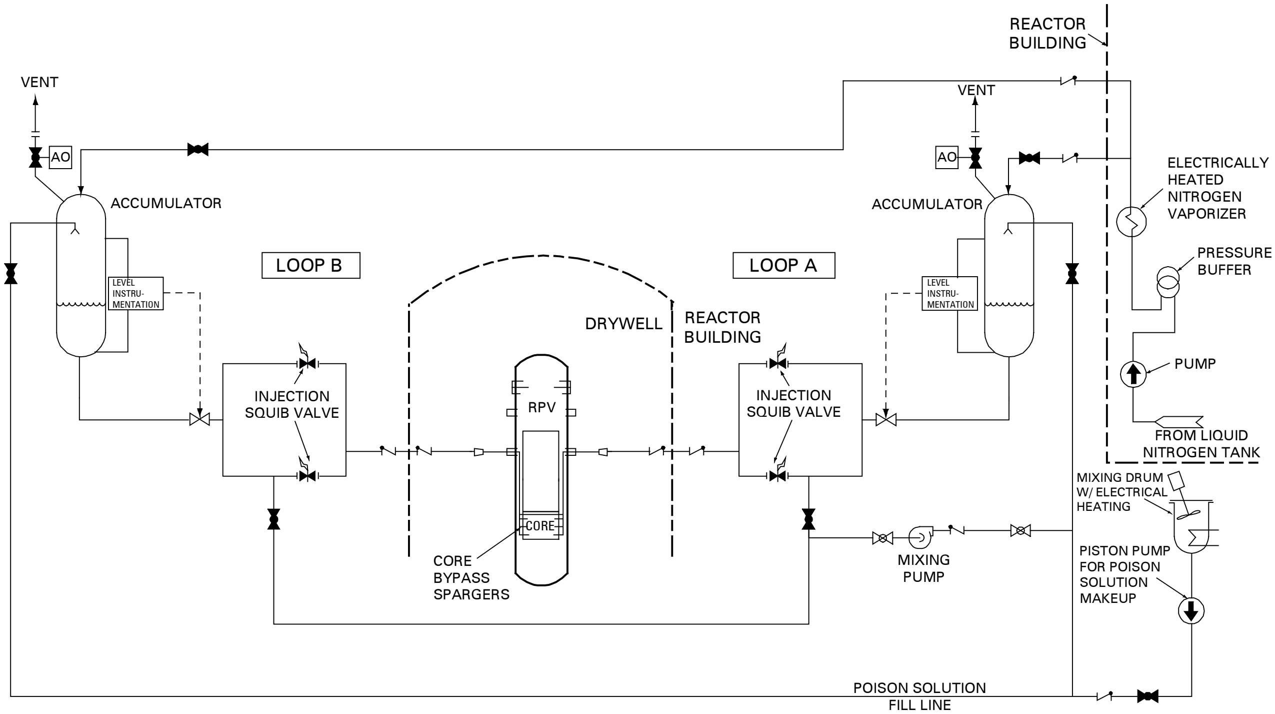


Figure 4-8. Standby Liquid Control System Schematic

from the accumulator into the reactor vessel that could interfere with Isolation Condenser System operation, or cause additional containment pressurization. As a backup, the accumulator vent valves are also opened at the same time.

Emergency Breathing Air System

Protection of the control room operators for events which could threaten the control room atmosphere (high radiation, toxic gases or smoke) is provided by passive means. If such an event occurs the normal Control Room Habitability Area (CRHA) HVAC is isolated and the Emergency Breathing Air System (EBAS) is initiated.

The CRHA* boundary envelope structures are designed with low leakage construction. The access doors are designed with self-closing devices, which close and latch the doors automatically following use. There are double door air locks for access and egress during emergencies.

EBAS is a redundant safety-related system that supplies stored, compressed air to the Control Room Habitability Area (CRHA) for breathing and for pressurization to minimize in-leakage. The EBAS is automatically initiated.

There are multiple trains of EBAS to provide the required redundancy. Each train of EBAS consists of compressed breathing air tanks and associated piping and components. EBAS has been sized to provide sufficient breathing quality air to maintain a positive pressure in the CRHA for a minimum of 72 hours.

* The CRHA includes all instrumentation and controls necessary for safe shutdown of the plant and is limited to those areas requiring operator access during and after a design basis accident (DBA).

Chapter Auxiliary Systems 5

Overview

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

Reactor Water Cleanup/ Shutdown Cooling System

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

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

System Description

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

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

Each train of the RWCU/SDC system performs the two functions of reactor water cleanup and shutdown cooling with a common piping system.

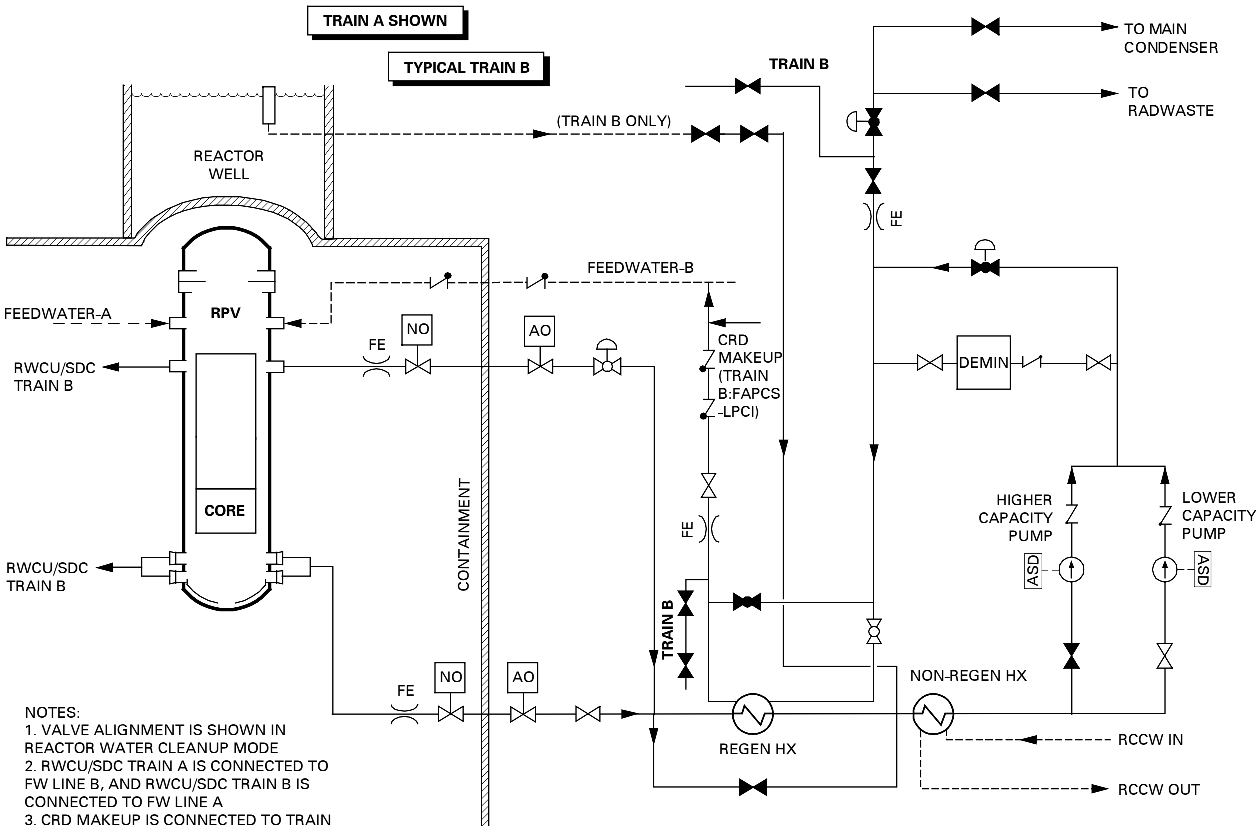


Figure 5-1. Reactor Water Cleanup/Shutdown Cooling System Schematic

The RWCU/SDC system suction line from reactor bottom head up to and including the outboard isolation valve, reactor bottom flow sample line up to and including the outboard isolation valve, pumps, demineralizer, pump suction line including suction valves up to and including the demineralizer downstream isolation valve, demineralizer bypass valve and upstream piping are constructed of stainless steel. The remaining system is constructed of carbon steel.

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

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

System Components

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

The RWCU/SDC system includes the following major components:

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

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

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

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

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

System Operation - Cleanup Mode

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

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

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

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

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

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

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

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

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

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

System Operation - Shutdown Cooling Mode

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

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

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

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

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

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

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

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

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

Refueling — The RWCU/SDC system can be used to supplement the FAPCS spent fuel heat removal capacity during refueling (or other times). It also can provide additional cooling of the reactor well water when the RPV head is off in preparation for removing spent fuel from the core.

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

In addition to the MS and Feedwater Systems, RWCU/SDC is the only other normally operating process system with primary system high pressure water located outside the 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 outside the containment which can be used to effect isolation.

Fuel and Auxiliary Pools Cooling System

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

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

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

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

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

Each FAPCS C/C train has sufficient flow and cooling capacity to maintain spent fuel pool bulk water temperature below 48.9°C (120°F) under

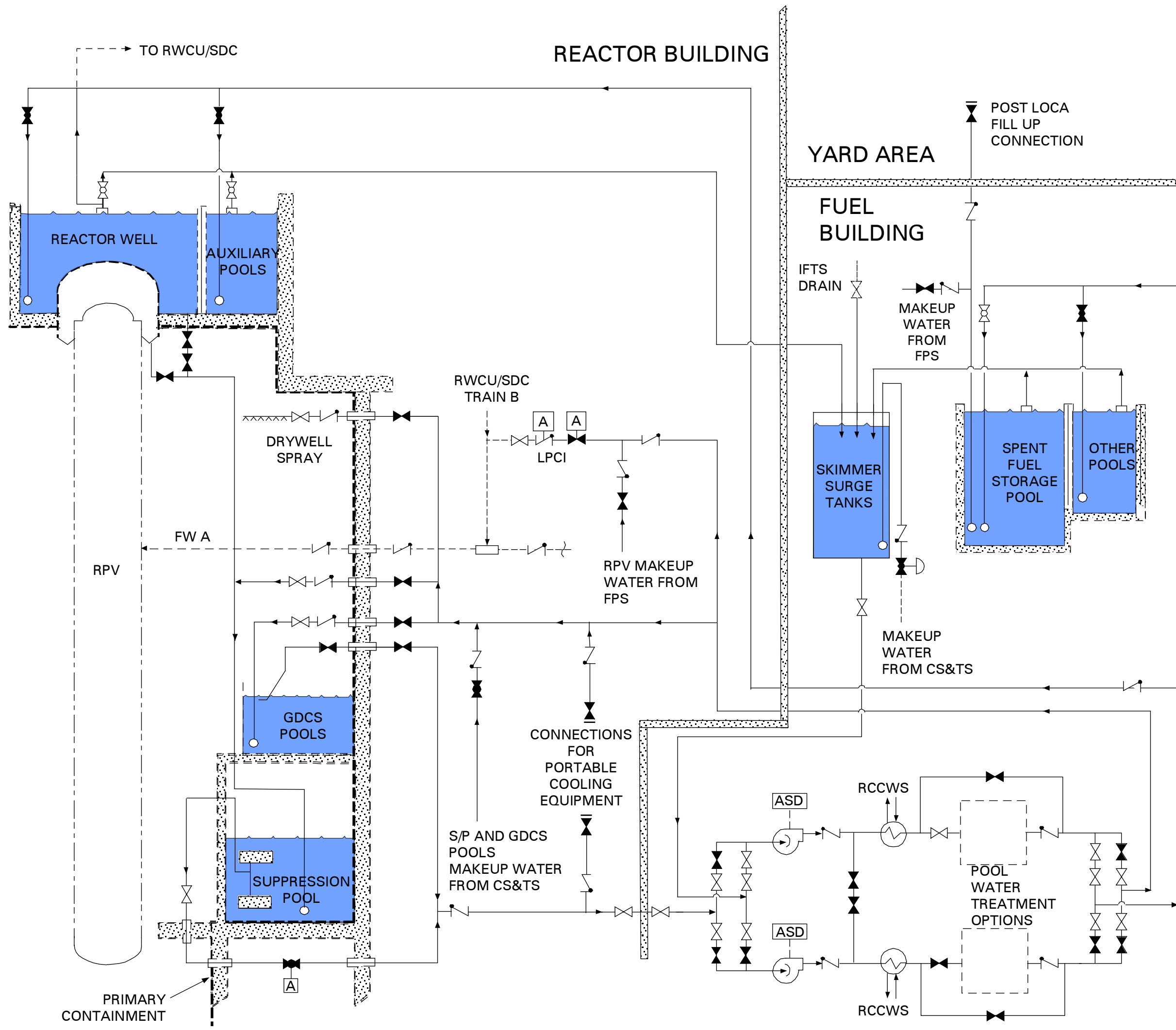


Figure 5-2. Fuel and Auxiliary Pools Cooling System Schematic

normal spent fuel pool heat load conditions. During the maximum spent fuel pool heat load conditions of a full core off-load plus irradiated fuel in the spent fuel pool resulting from 10 years of plant operations, both FAPCS C/C trains are needed to maintains the bulk temperature below 60°C (140°F).

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

System Operation

The following discuss the major design operating modes of FAPCS.

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

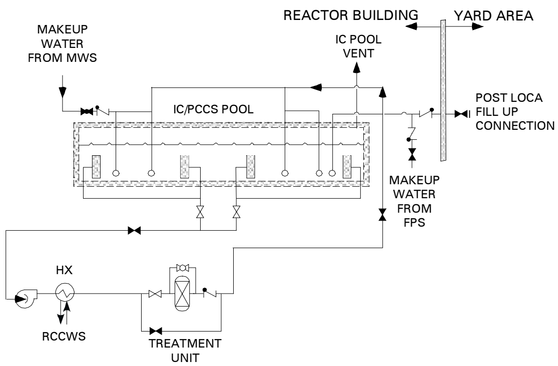


Figure 5-3. FAPCS IC Subsystem Schematic

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

- Upper fuel transfer pool
- Buffer pool
- Reactor well
- Dryer and separator storage pool

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

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

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

Low Pressure Coolant Injection (LPCI) – This mode may be initiated following an accident after the reactor has been depressurized to provide reactor makeup water for accident recovery. In this mode the FAPCS pump takes suction from the suppression pool and pumps it into the reactor vessel via RWCU/SDC loop B and then Feedwater loop A.

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

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

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

Reactor Component Cooling Water System

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

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

Each RCCWS train consists of 3 parallel pumps, 3 parallel heat exchangers, one surge tank, connecting piping, and instrumentation. Both trains share a chemical addition tank. The two trains are normally connected by cross-tie piping during operation for flexibility, but may be isolated for individual train operation or maintenance of either train. The pumps in each train discharge through check valves and butterfly valves to a common header leading to the RCCWS heat exchanger header. Crosstie lines between each train are provided up and downstream of the heat exchangers; at the pump suction and discharge headers; and downstream of the Radwaste Building cooling water supplies. The heat exchanger outlet isolation valves are automatic. The heat exchanger flow control valves, bypass temperature control valves, and cross-tie isolation valves are pneumatically operated.

RCCWS cooling water is supplied to the following major users:

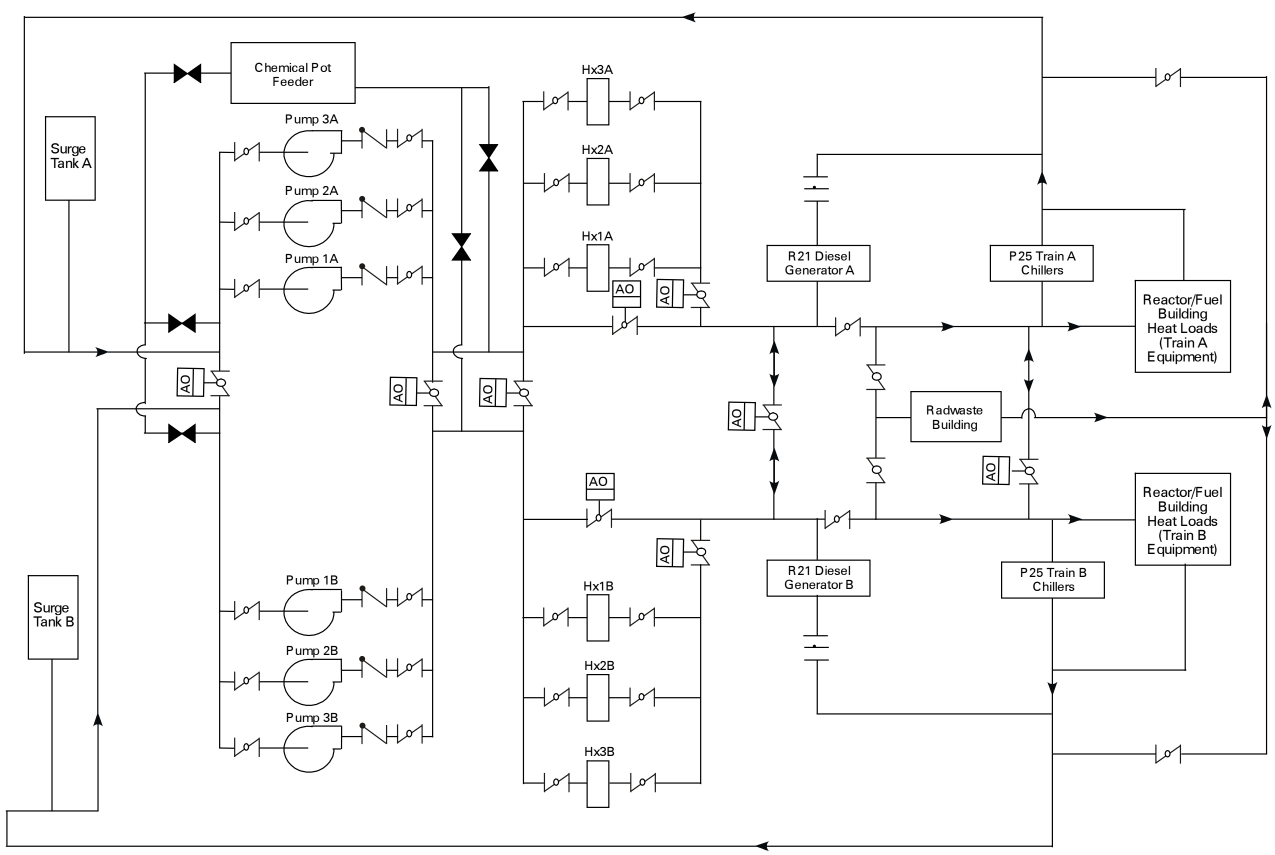


Figure 5-4. Reactor Component Cooling Water System Schematic

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

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

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

Normally, the pumps in each train are powered from independent buses. During a LOPP, the pumps in either train can be powered from the Standby On Site AC Power System.

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

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

System Operation

The RCCWS operates during startup, normal power, hot standby, normal and extended cooldown, shutdown, and LOPP. If any of the redundant users requires cooling in addition to the primary users, additional pumps may need to be started.

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

Plant Service Water System

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

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

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

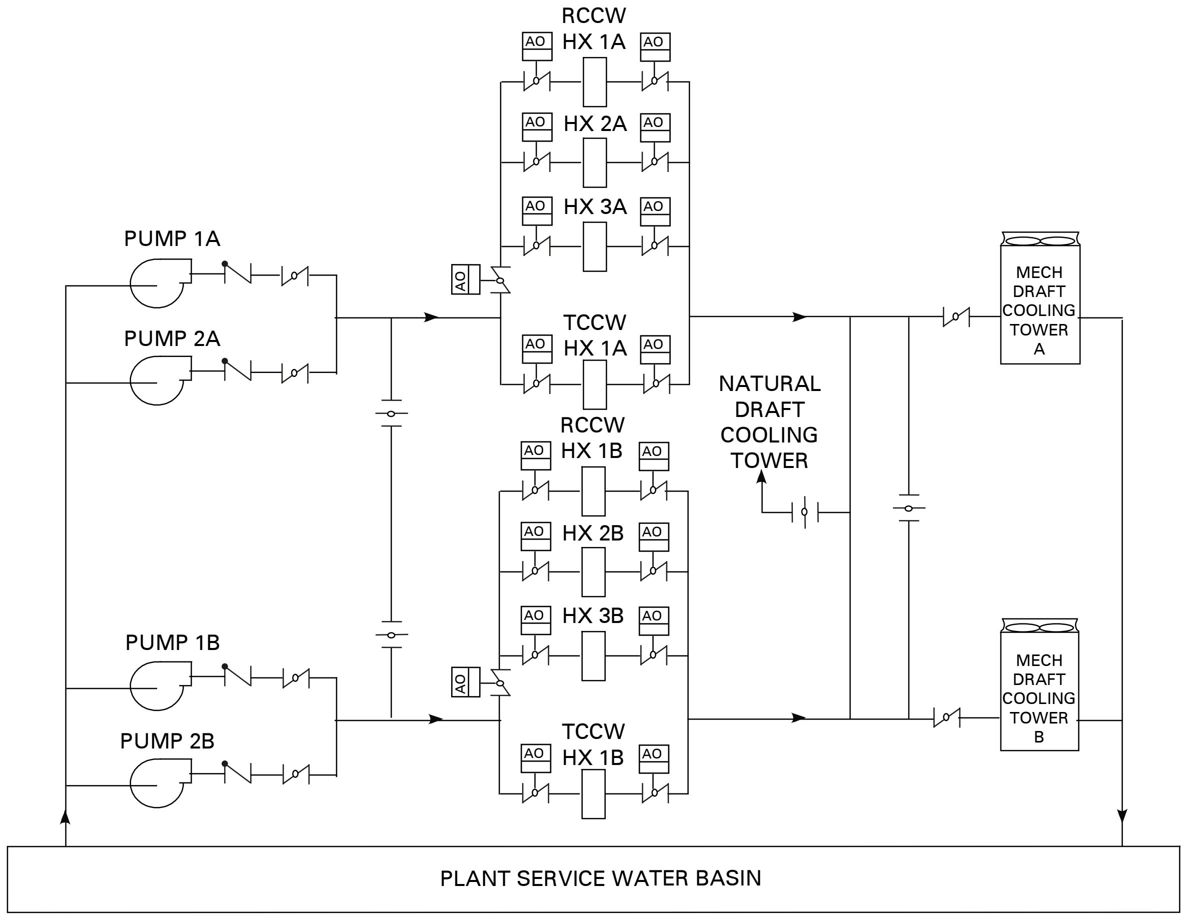


Figure 5-5 Plant Service Water System Schematic

a LOPP, the pumps are powered from the two non safety-related standby diesel-generators.

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

The PSWS cooling towers and PSWS basins are located inside the plant security protected area. Each PSWS train is provided with a separate, multi-celled mechanical draft cooling tower with 50% of the cell fans supplied by one of the redundant electrical buses. During a LOPP, the fans are powered from the two non safety-related standby diesel-generators. The adjustable-speed, reversible motor fan units can be controlled for cold weather conditions to prevent freezing in the basin. The mechanical and electrical isolation of the cooling towers allows maintenance, including complete disassembly, during full power operation. Makeup, for blowdown, drift, and evaporation losses to the basin is from the Station Water System. Provision for anti-fouling treatment of the PSWS is provided.

Blowdown from the PSWS basins is by gravity into the main cooling tower basin or directly to the plant waste effluent system.

System Operation

During normal operation the primary source of cooling water for the PSWS is the cooling tower makeup pumps, with the PSWS pumps serving as a backup. Heat removed from the RCCWS and TCCWS is rejected to the main cooling tower basin when the cooling tower makeup pumps are in operation. If the PSWS pumps are in operation, the PSWS mechanical draft cooling towers are used to reject the heat removed from RCCWS and TCCWS to the environment.

During periods when the required makeup water for the main cooling tower basin is reduced (e.g. winter months) or when the cooling tower makeup pumps are unavailable, the PSWS pumps are the source of cooling water for the RCCWS and TC-CWS heat exchangers.

Operation of any two of the four cooling tower makeup or PSWS pumps is sufficient for the design heat load removal in any normal operating mode with the exception of the normal cooldown mode, when three pumps are initially required.

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

Drywell Cooling System

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

Each upper drywell FCU has a cooling capacity of 50% of the upper drywell design heat load during

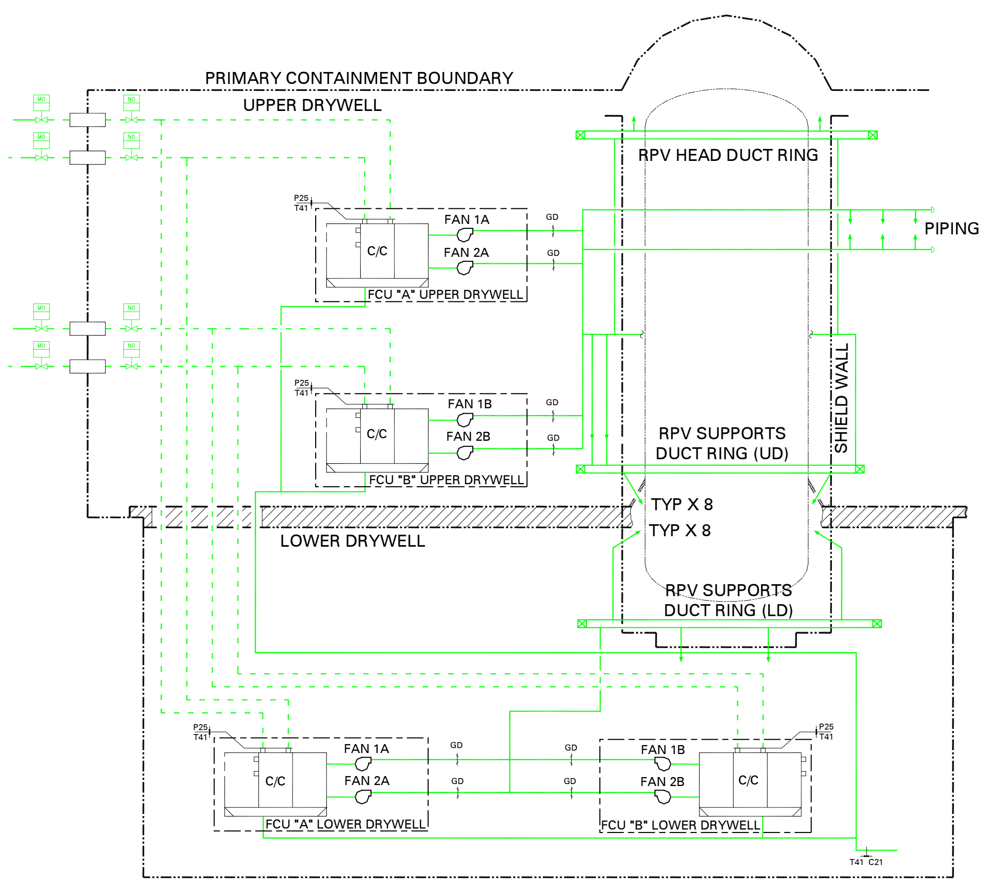


Figure 5-6 Drywell Cooling System Schematic

normal plant operating conditions. Both FCUs are normally operating. Each FCU comprises a cooling coil and two fans downstream of the coil. Nuclear Island subsystem of CWS train A supplies one FCU, and Nuclear Island subsystem of CWS train B supplies the other. One of the fans operates while the other is on standby status. The fan on standby automatically starts upon loss of the lead fan. Cooled air/nitrogen leaving the FCUs enters a common plenum and is distributed to the various zones in the upper drywell through distribution ducts. Return ducts are not provided. The FCUs draw air/nitrogen directly from the upper drywell. Each FCU is equipped with a condensate collection pan.

Each lower drywell FCU has a cooling capacity of 50% of the lower drywell design heat load. Each FCU comprises a cooling coil and two fans downstream of the coil. One of the fans operates while the other is on standby status. The fan on standby automatically starts upon loss of the lead fan. Nuclear Island subsystem of CWS train A supplies one FCU, while Nuclear Island of subsystem CWS train B supplies the other. Cooled air/nitrogen is supplied below the RPV and in the RPV support area through supply ducts. Return ducts are not provided. The FCUs draw air/nitrogen directly from the lower drywell.

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

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

System Operation

During normal plant operating condition, two FCUs in the upper drywell and two FCUs in the lower drywell are continuously operating to maintain the required ambient conditions.

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

Non-safety related onsite diesel generators power the FCUs during a LOPP as long as there is no loss of coolant accident (LOCA) signal.

Containment Inerting System

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

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

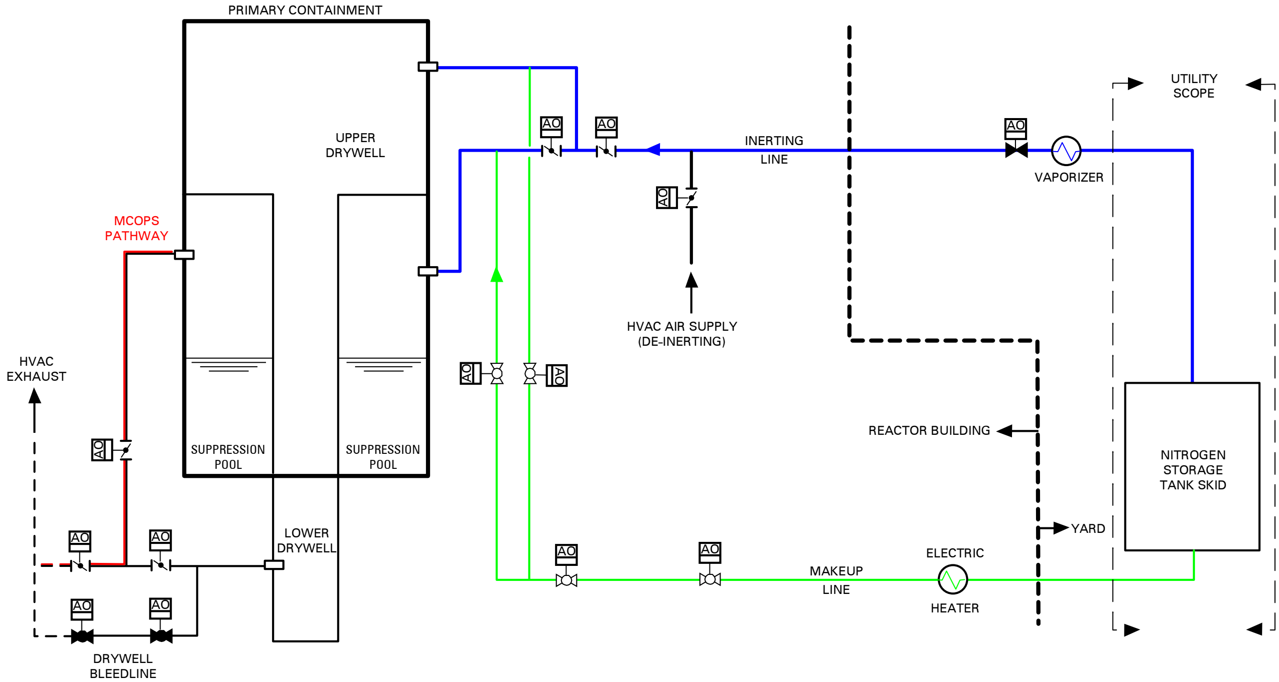


Figure 5-7. Containment Inerting System Schematic

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

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

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

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

Chapter 6
Core and Fuel Design

Introduction and Summary

The design of the ESBWR core and fuel is based on the proper combination of many design variables and operating experience. These factors contribute to the achievement of high reliability, excellent performance, and improved fuel cycle economics.

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 ESBWR core. As these technological improvements are added, the core and fuel design parameters are optimized to achieve better fuel cycle economics, while improving fuel integrity and reliability and while maintaining overall reactor safety.

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

- The ESBWR core mechanical design is based on conservative application of stress limits, operating experience, and experimental test results. The moderate pressure levels characteristic of a direct cycle reactor, approximately 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 ESBWR are significant, advantageous factors in minimizing Zircaloy clad temperature and associated temperature-dependent corrosion and hydride buildup. This results in improved cladding performance at high burnup.

- The basic thermal and mechanical criteria applied in the ESBWR design have been proven by irradiation of statistically significant quantities of fuel. The design heat fluxes and linear heat generation rates are similar to values proven in fuel assembly irradiation in the large fleet of operating BWRs.

- In-reactor experience of fuel components acquired in the existing fleet is applicable to the ESBWR.

- Because of the large negative moderator density (void) coefficient of reactivity, the ESBWR has a number of inherent advantages, including (1) inherent self-flattening of the radial power distribution, (2) spatial xenon stability, and (2) ability to override xenon in order to follow load. The inherent spatial xenon stability of the ESBWR is particularly important for large-sized plants, and permits daily load following over a large range of core power levels.

- The moderate power density and the power distributions used in sizing the ESBWR core includes margins providing for operational flexibility.

- The ESBWR 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 ESBWR is arranged as an upright cylinder containing a large number of fuel assemblies (1132) located within the reactor vessel. The coolant flows upward through the core. The core arrangement (plan view) and the lattice configuration are shown in Figures 6-1 and 6-2, respectively. Important components of this arrangement are described in the following pages.

As can be seen from Figure 6-1, the ESBWR reactor core is comprised of fuel assemblies, control rods and nuclear instrumentation. The fuel assembly and control rod mechanical designs are basically the same as used in all but the earliest GE boiling water reactors; however, evolutionary improvements have been made to these components throughout the history of the GE BWR. The current generation of these components will be described below for application to the ESBWR.

GE14 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 GE12 and GE14 designs both 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. Thus, GE provides two 10x10 fuel designs for the ESBWR.

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.

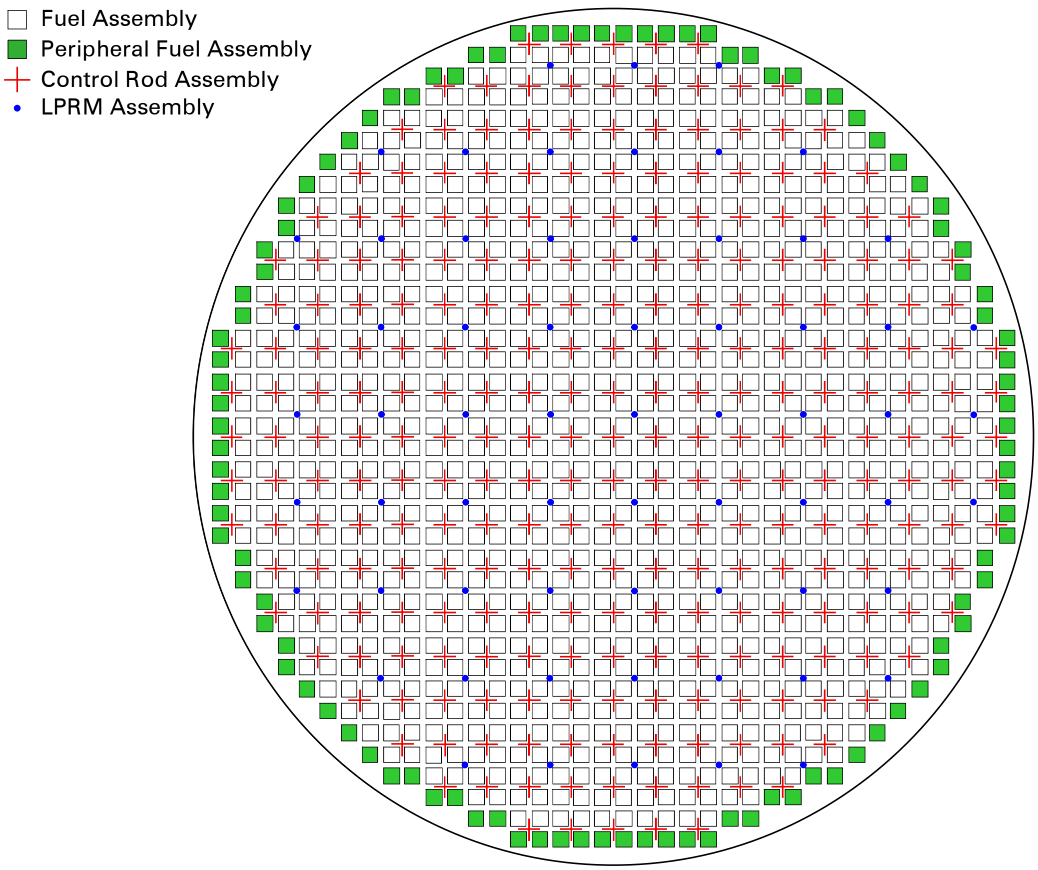


Figure 6-1. ESBWR Core Configuration

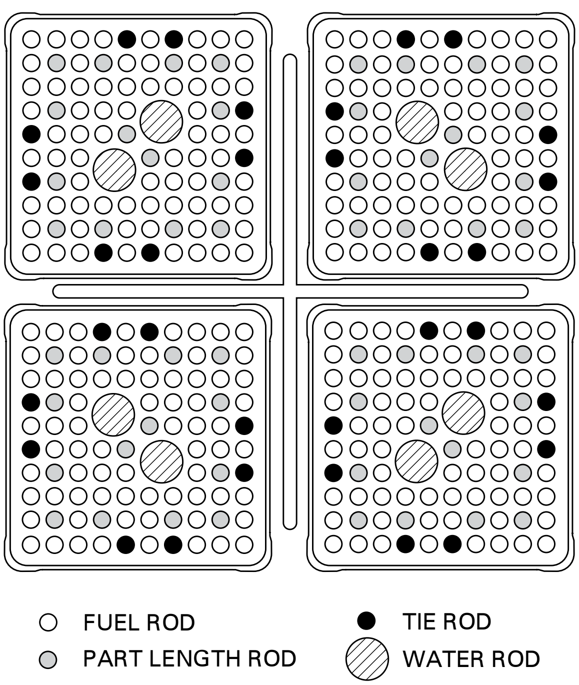


Figure 6-2. Four Bundle Fuel Module (Cell)

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

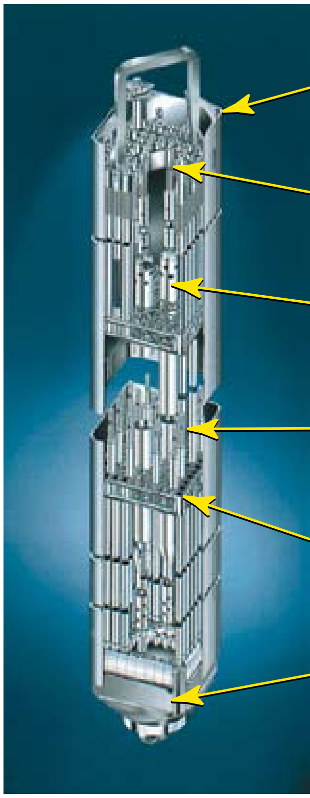


Figure 6-3. GE14 Fuel Assembly

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

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

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

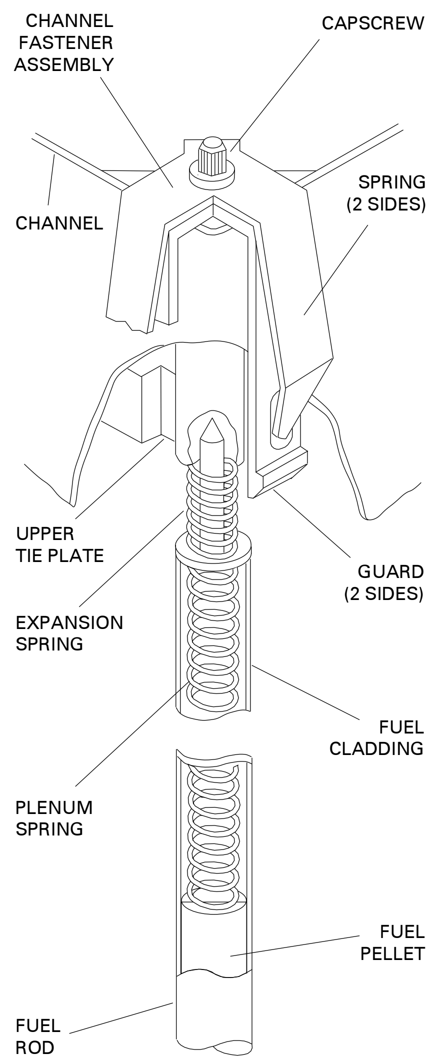


Figure 6-4. Channel Fastener Assembly

to fuel ratio in the top of the core to improve cold shutdown margins and fuel efficiency.

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 GE10, GE11, and GE13. 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.

High Pressure Drop 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 an increase in single-phase pressure drop at the lower tie plate. This trade-off provides improved stability with essentially the same overall pressure drop as previous designs. In addition, it allows for the use of very small flow holes in the lower tie plate (LTP), which act as a very effective debris filter. Figure 6-5 shows a top view of one of the three possible debris filter LTPs. The bundle flow passes through the small holes which are only 0.125 inches in diameter, and of which there are 444. Debris filter LTP designs are standard with the GE14 design, and have been provided as an option for the GE11 through GE13 designs. Any of GE's debris filter LTPs can be applied to ESBWR, 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.

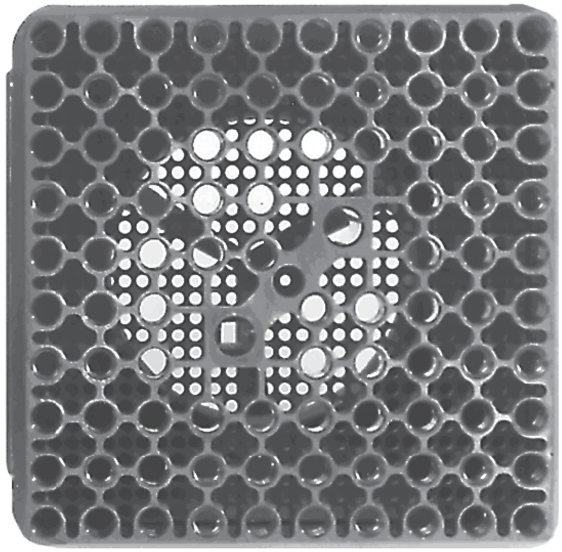


Figure 6-5. Channel Fastener Assembly

Interactive Channels

The interactive fuel channel concept used with the GE10 through GE13 fuel bundles is also included as an integral part of the GE14 design. This 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.

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

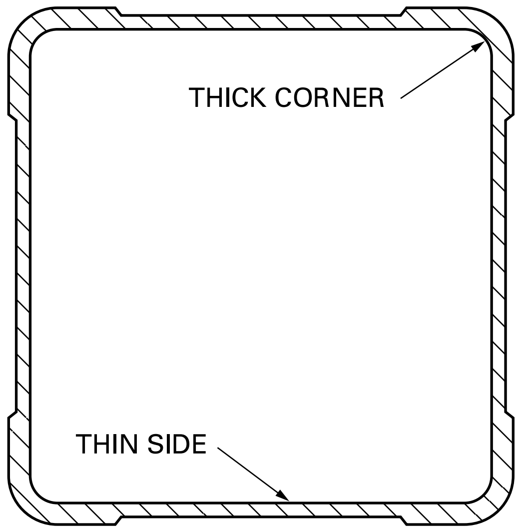


Figure 6-6. Cross Section of Interactive Channel

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

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

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.

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.

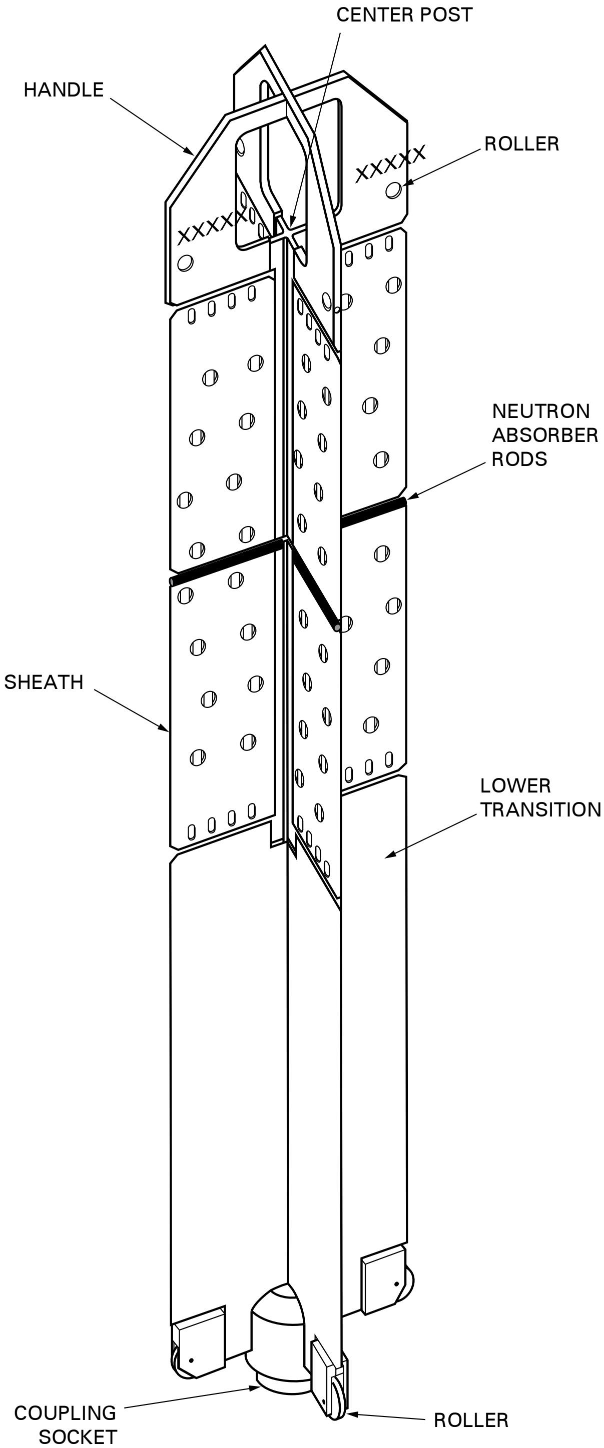


Figure 6-7. ESBWR Control Rod

Other Reactor Core Components

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

SRNM Assembly

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

LPRM Assembly

There are 64 Local Power Range Monitoring (LPRM) assemblies evenly distributed throughout the reactor core. Each assembly extends vertically in the core bypass water region at every fourth intersection of the fuel assemblies and contains four fission chamber detectors evenly spaced at four axial positions adjacent to the active fuel. Detector signal cables are routed within the assembly toward the bottom of the reactor pressure vessel where the assembly penetrates the vessel pressure boundary. Below the vessel bottom, the pressure boundary is formed by an extended portion of the in-core instrument housing tube that houses the assembly.

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

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

Neutron Sources

Several antimony-beryllium startup sources are located within the core. They are positioned verti-

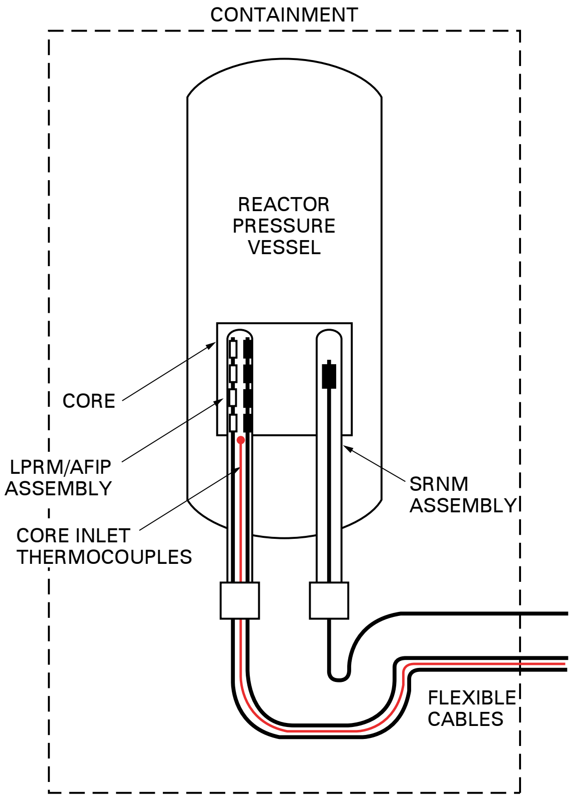


Figure 6-8. AFIP, LPRM and SRNM Schematic

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

The active portion of each source consists of a 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

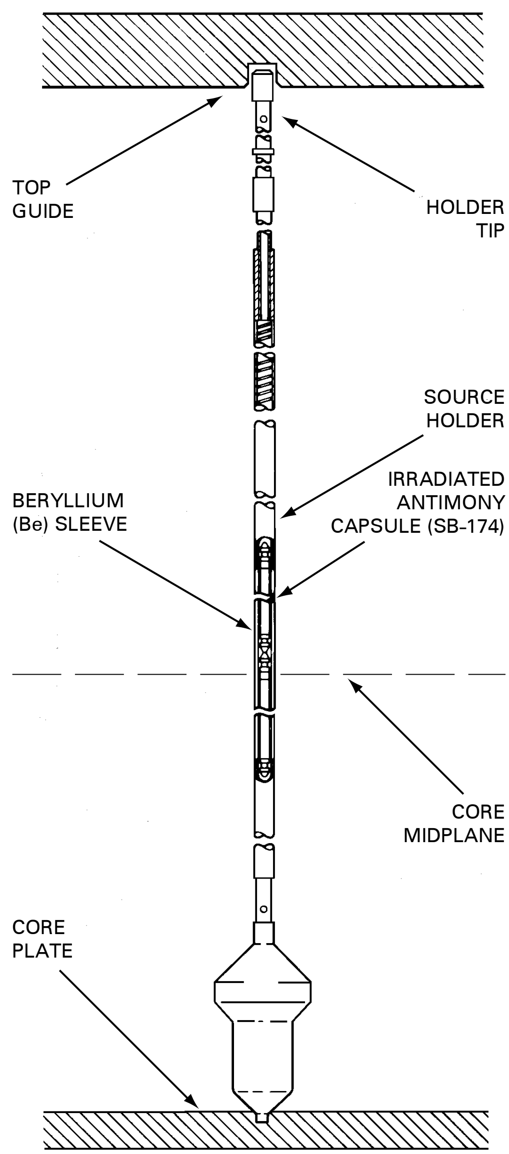


Figure 6-9. Neutron Source Schematic

due to the use of undue conservatism.

Core Configuration

The ESBWR core map is illustrated in Figure 6-1. There are 1132 fuel assemblies, 269 control rods and 64 LPRM assemblies. Also the core periphery zone with more restrictive inlet flow orifices is shown.

As an option, ESBWR can employ the Control Cell Core (CCC) operating strategy in which control rod movement to offset reactivity changes during power operations is limited to a fixed group of control rods. Each of these control rods and its 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 ESBWR core has the capability of overriding the negative reactivity introduced by the build-up of xenon following a power decrease.

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

  –**Load Changing by Control Rod Movement:** The ESBWR is capable of daily load following between 100% and 50% power by adjusting control rod density within the core.

Reactivity Control

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

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

Fuel Management

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

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

Neutron Monitoring System

The Neutron Monitoring System (NMS) is a system of in-core neutron detectors and out-of-core electronic monitoring equipment. The system provides indication of neutron flux, which can be correlated to thermal power level for the entire range of flux conditions that can exist in the core. There are four subsystems in the NMS: the Startup Range Neutron Monitoring (SRNM) Subsystem, the Power Range Neutron Monitoring (PRNM) Subsystem [comprised of the Local Power Range Monitors (LPRM) and Average Power Range Monitors (APRM)], the Automatic Fixed In-Core Probe (AFIP) Subsystem, and the Multi-Channel Rod Block Monitoring (MRBM) Subsystem.

The NMS design has been greatly simplified for ESBWR application. Key simplification features include the SRNM, period-based trip logic, and the Automatic Fixed In-Core Probe (AFIP) System. The SRNMs replace the separate Source Range Monitor (SRM) and Intermediate Range Monitor (IRM) found in conventional BWRs. Use of these fixed in-core SRNM detectors eliminates the drive mechanism and the associated control systems for the moveable SRM and IRM detectors. IRM range switches have been eliminated by incorporating a period-based trip design in the startup power range. Hence, operability is greatly improved and accidental trips due to manual range switching are eliminated. The AFIP uses fixed in-core gamma thermometers for automatic core flux mapping and calibrating the power range monitors in the ESBWR design, thereby substantially reducing reactor room space for the old TIP system, enhancing operability, and reducing personnel radiation dosage.

Startup Range Neutron Monitoring (SRNM) Subsystem

The SRNM Subsystem monitors the neutron flux from the source range to approximately 100%

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

Power Range Neutron Monitoring (PRNM) Subsystem

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

The PRNM Subsystem consists of two subsystems:

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

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

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

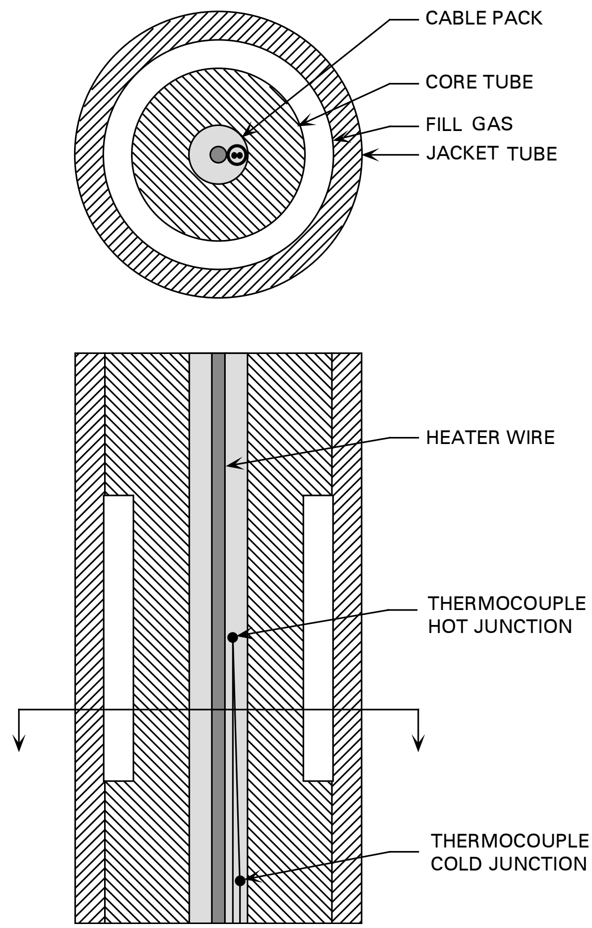


Figure 6-10. Gamma Thermometer Cross-section

Automatic Fixed In-Core Probe (AFIP) Subsystem

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

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

During core power and LPRM calibration, the AFIP signals are collected automatically to the AFIP data processing and control unit, where the data are properly amplified and compensated by applying correct sensor calibration adjustment factors. Such data are then sent to the plant computer function of the NE-DCIS for core local power and thermal limits calculations. The calculated local power data are then used subsequently for LPRM calibration. The AFIP data collection and processing sequences are fully automated, with manual control available. The AFIP gamma thermometer sensor has near constant, very stable detector sensitivity due to its operation principle, and its sensitivity does not depend upon fissile material depletion or radiation exposure.

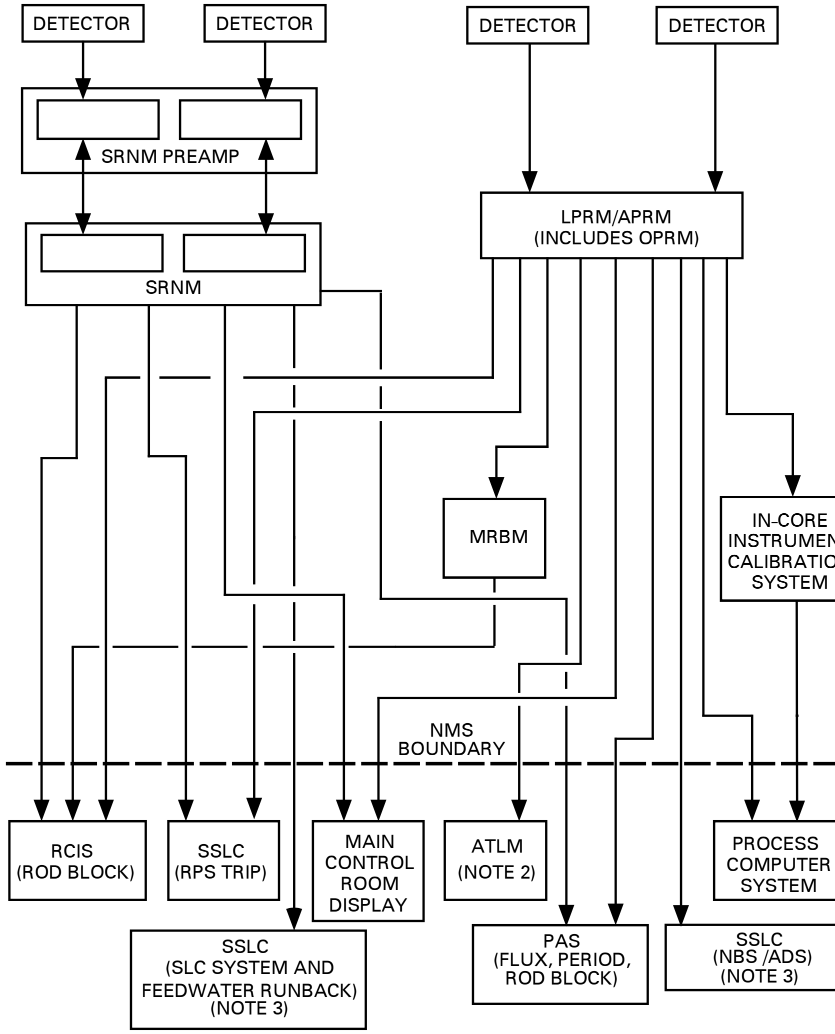
The AFIP gamma thermometer, however, can be calibrated, either manually or automatically, by using a built-in calibration device inside the gamma thermometer/LPRM assembly. The calibrated new sensitivity data of the AFIP sensors are stored in the AFIP control unit and are readily applied to the newly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

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

Multi-Channel Rod Block Monitor (MRBM) Subsystem

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

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



NOTES:

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

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

Chapter 7 Instrumentation and Control

Overview

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

Previous BWRs used hard wired point-to-point connections from field instrumentation to control systems and panels in the control room. 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 ESBWR, instead, is designed with an I&C system that uses extensive multiplexing and fiber optics; this allows far fewer cables to the control room and elimination of the cable spreading room.

The system design comprises:

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

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

Design goals of the I&C System include:

- Minimize reactor trips/system unavailability due to human errors and eliminate scrams and trips from single active component failures
- Design any systems necessary for power generation (except the electrical system) to be single-failure proof for both control and trips
- 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, reduce surveillance requirements and frequencies, 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 ESBWR monitoring and control functions. The standardized Class 1E (safety-related) Logic Units (LU), non-Class 1E Control Processors (CP) , 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 LU chassis, CP chassis, RMU chassis and Human System Interface (HSI) chassis are standard for all similar ESBWR applications; only modular, plug-in interchangeable, circuit boards differ between systems. Functional features provided in the I&C design include:

- Sensor signal processing
- Redundant sensor power supplies to meet the requirements of all sensors
- 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 and power feeds for both the safety and non safety electronics
- Fiber optic and other interfaces, allowing the LU/CP and HSI units to communicate directly with plant multiplexing networks
- Menu-driven front panel for operator/technician interface

Data Control Networks

The DCIS provides redundant and distributed control and instrumentation data communications networks to support the monitoring and control of interfacing plant systems. The system contains Essential DCIS (E-DCIS) cabinets and Non-Essential DCIS (NE-DCIS) cabinets that respectively acquire and output signals to/from safety and non-safety/BOP systems. 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 DCIS also includes the associated data acquisition and communication software required to support its function of plant-wide data and control distribution; all data control networks use redundant data paths and power supplies to increase reliability. As shown on Figure 7-1, digital technology and multiplexed fiber optic signal transmission technology have been combined in the ESBWR design to integrate control and data acquisition for all of the plant buildings.

Signals from various plant process sensors provide input to RMUs located near those 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 E-DCIS has four control data networks (each of which is redundant), one per division with the NE-DCIS being a control data network with dual (and sometimes triple) redundancy. Whether E-DCIS or NE-DCIS, redundancy is such that a single cable or power feed can be lost or any RMU can 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 to keep plant wiring as short as possible.

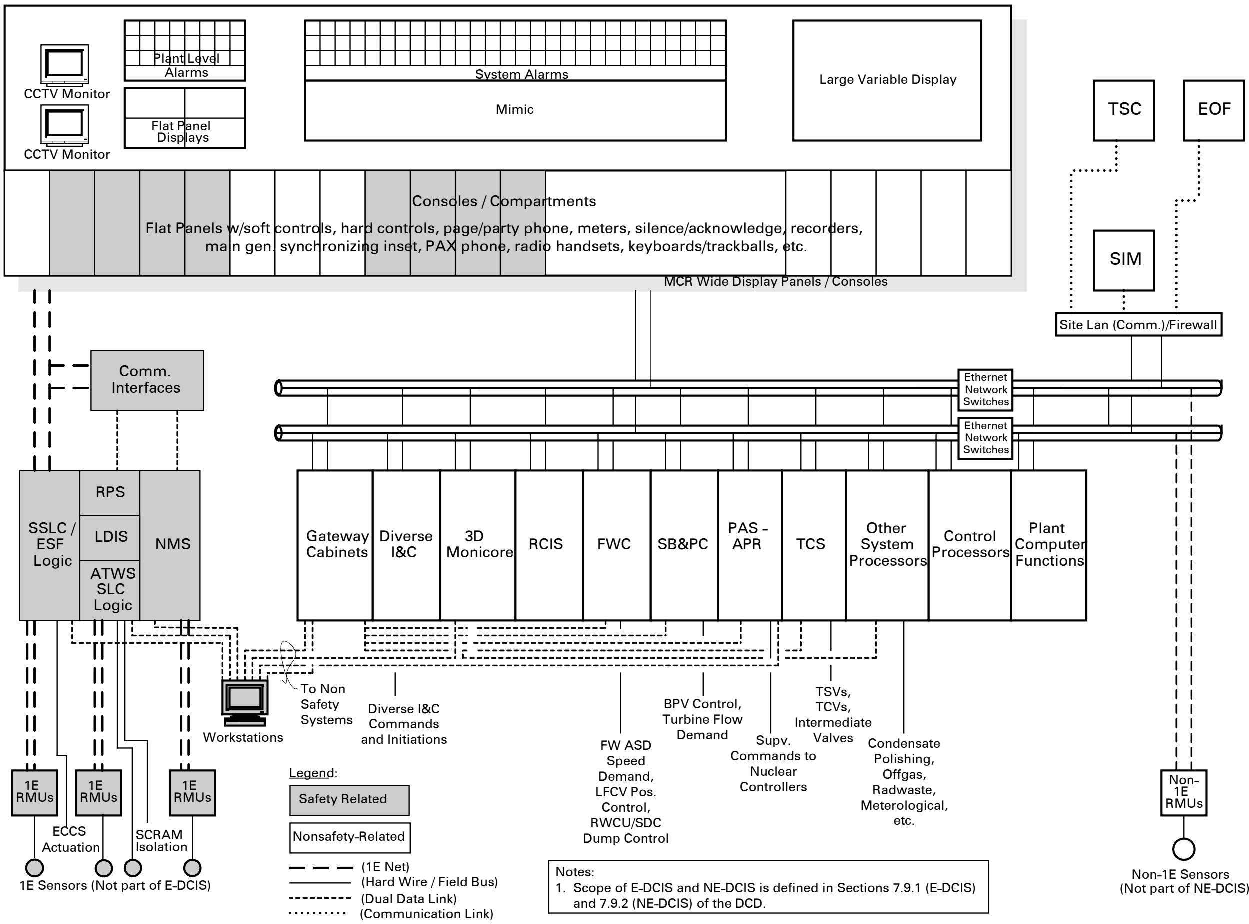


Figure 7-1. ESBWR Digital Control and Information System

Digital Protection System Applications

Advanced Safety Systems Design

The Reactor Protection System (RPS), Neutron Monitoring System (NMS), Leak Detection and Isolation System (LD&IS) and the ECCS (ICS, GDCS and ADS) are four-channel (divisional) systems actuated by two-out-of-four logic from four-channel (divisional) sensor input. NMS is described in Chapter 6.

Safety System Logic and Control

Safety System Logic and Control (SSLC) and the associated E-DCIS equipment is divided into four divisions. Each division is physically and electrically separated from the other divisions. Communications between divisions 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 data control networks when the signal design requirements are such that processing the signal through them would cause the established design requirements (signal processing speed) to be exceeded.

The SSLC also controls the automatic actuation and operation of the following Emergency Core Cooling Systems (ECCS) during emergency operation:

- Isolation Condenser System (ICS)
- Safety relief valves and depressurization valves of the Automatic Depressurization System (ADS)
- Gravity Driven Cooling System (GDCS)

Standby Liquid Control System (SLCS) logic is separate and diverse 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 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 alarms.

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

Leak Detection and Isolation System

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

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

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

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

Fault-Tolerant Process Control Systems

The ESBWR 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 may be used where the function is not important to power generation.

In general, the key ESBWR boiler control systems such as the feedwater control, reactor pressure regulator and plant automation systems are based on the triplicated, microprocessor-based FTDC; the main turbine is also controlled with a triply redundant control system to minimize DCIS failures causing either lost generation or reactor transients. 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 uses dedicated triply redundant RMU data acquisition; all control inputs are made available to all processors and all 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). Additionally the triply redundant controllers are furnished with three independent uninterruptible power feeds. 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"; both processors receive all inputs but only one processor at a time provides an output to the NE-DCIS network and to the RMUs. 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; 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 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 NE-DCIS networks (triply redundant controllers use triply redundant data acquisition).
- 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 Technician Interface Unit (TIU) for certain controllers providing a menu-driven system which allows the technician to inject test signals, perform troubleshooting and calibrate process parameters; all other controllers can be accessed centrally for troubleshooting purposes.

The fault-tolerant architecture of the FTDC design provides assurance that no single active component failure within the sensing, control, or communication equipment can result in loss of system function or plant power generation. The 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.

Plant Automation System

The primary objective of the Plant Automation System (PAS) is to both control the balance of plant and to set reactor temperature, pressure, power and neutron flux to those values required by the automation scheme. This control is only possible in normal plant operation and PAS controller failures will "trip" the plant out of automation and leave the plant "as is" for manual control. It should be noted that system control is only in that system's logic and the PAS only issues "supervisorial" commands to be carried out by system controllers; all automatic and manual reactor protection and BOP protection functions are always operative and cannot be bypassed by any automation control.

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 PAS 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 PAS 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 PAS commands to the Steam Bypass and Pressure Control System (SB&PC) and Rod Control and Information System (RCIS).

The PAS consists of both dual and triply redundant process controllers; these receive information from the various plant sensors and issue commands to the BOP system controllers and to RCIS to position control rods and to the SB&PC System to set pressure.

While triply redundant controllers provide commands to the main reactor control systems, the dual controllers provide 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.

Feedwater Control System

The Feedwater Control System (FWCS) 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 adversely affects plant thermal limits).

A fault-tolerant triplicated, digital controller using a conventional three-element control scheme, provides control signals to the four adjustable speed drives (ASDs) that, in turn, control the four feedwater pump motors. Feed flow to the reactor (and thereby reactor level) is regulated by changing the speed of the feedpump motor; a lower capacity Low Flow Control Valve (LFCV) is used to control level at low reactor powers.

The FWCS 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 FWCS automatically controls both the Reactor Water Cleanup/Shutdown Cooling system (RWCU/SDC) 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 in the reactor. At higher flow rates, the FWCS 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 (SB&PC) 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 SB&PC System to Automatic, but either the operator or the APR can switch the SB&PC System to Manual.

Unlike previous BWRs, reactor pressure and not turbine inlet pressure is controlled by the SB&PC System. In normal power generation, reactor pressure is controlled by automatically positioning the turbine control valves - the pressure control signal “passes through” the SB&PC 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.

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 SB&PC 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.

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 PAS; this will also follow a predefined control rod sequence. As with all automated control systems, only the operator can switch the RCIS controllers to Automatic, but either the operator or the PAS 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.

All of the rod blocks, sequence controls and thermal limit monitoring remain operative whether or not the plant is being controlled by PAS.

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 and Fuel 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 Monitoring System

The Containment Monitoring (CMS) 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 CMS provides normal plant shutdown and post-accident monitoring for gross gamma radiation and hydrogen/oxygen concentration levels in both drywell and suppression chamber. The CMS 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.

Plant Computer

On-line networked computers are provided to monitor and log process variables and make certain analytical computations. The process computer function is distributed in various workstations and cabinets on the same network as the plant controllers; the important functions are redundant and there is no longer a centralized process computer. The process computer 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.
- SPDS (Safety Parameters Display System)
- Historian (normal slow speed recording)
- TRA (Transient Recording & Analysis)
- SOE (Sequence of Events)
- Alarm & Annunciator
- Firewalls to support offsite simulator and emergency response functions
- General Displays

Remote Shutdown System (RSS)

In the event that the control room becomes inaccessible, the reactor can be brought from power range operation to cold shutdown conditions by use of controls and equipment that are available outside the control room. The reactor can be manually scrammed from the RSS but all of the remaining RSS control is accomplished with a division 1 (or division 2 depending on the RSS panel) Video Display Unit (VDU) and a non safety VDU. These VDUs remain operational with a complete loss of all control room equipment since neither the safety or non safety networks are physically in the main control room nor does the loss of the fiber and Ethernet cables connecting the control room VDUs to the network adversely affect the network. Additionally control room fires or smoke will not adversely affect the safety-related or non-safety related controllers/ network equipment nor their power feeds.

As a result, all safety-related control and monitoring (division 1 and 2 only) and all non-safety related control and monitoring remain available to the remote shutdown panel operator in the same way that he would operate that equipment from the main control room. Since it is battery operated, the safety-related equipment is always available, the non-safety related systems' availability (i.e., control, not monitoring) is dependent on the status of offsite power and the diesel generators.

The remote shutdown system panels are located in enclosed rooms in their corresponding divisional areas of the reactor building and are normally locked; any access to these rooms is alarmed in the main control room.

Main Control Room

The key elements of the ESBWR 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. The Lungmen (ABWR) simulator demonstrates the control approach.

Total plant control is achieved from the Main Control Console (MCC) for all phases of operation. The console design incorporates flat panel touchscreen and display devices, and a limited number of hard switches as the primary operator interface devices. The flat panel displays are driven by the DCIS. The main control console has a low profile so that the operators can perform their duties from a seated position and still view the wide display panel.

The Wide Display Panel (WDP) provides summary information on plant status parameters and key alarms to the operators, supervisors and other technical support personnel in the MCR. The WDP is located immediately in front of the operators when they are at their normal work station seated at the main control console. This WDP includes a fixed mimic display, an approximately 250 cm large variable display, top-level plant alarms, summary and detailed system level alarms, touch-control flat panel displays and system level annunciators. The WDP incorporates the Safety Parameter Display System (SPDS) as part of the plant status summary information.

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

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 (only one is necessary to operate the plant), 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 WDP 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 non safety reactor and 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, touchscreen flat panel displays mounted on the MCC. The displays are driven by the DCIS. There are many types of system display formats which can be shown on

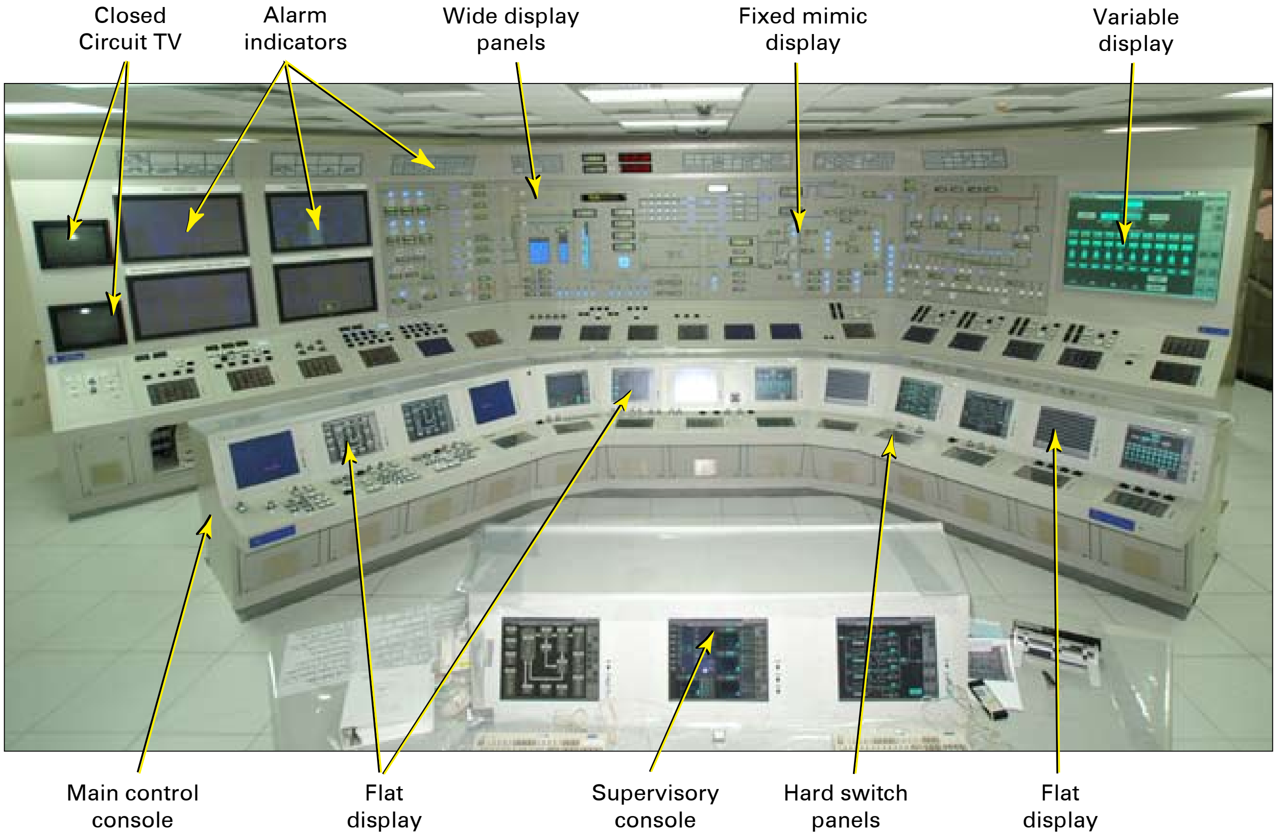


Figure 7-2. Lungmen Simulator

the flat panels and all are accessible from menu displays; displays include summary plant status displays, P&ID/control displays, trend plots, system status formats, alarm summaries, plant operating procedure guidance displays, and plant automation guidance displays.

Although each flat panel is nominally assigned to a specific system or systems, it should be understood that this is under the control of the operator; any non-safety VDU can show any non-safety display and each safety VDU can display anything within its associated division. Any system can be controlled by only one VDU at a time; this multi-redundant display capability ensures continued normal plant operation in the event of a failure of one or more of the flat panels.

The system status displays provide information on individual plant systems. The touch screens on the safety/non safety flat panels provide direct control for safety-related and 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 flat panel displays 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 flat panel display devices are used to support both safety-related and non-safety related 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 and provide for control and monitoring of the four redundant and independent divisions of the Emergency Core Cooling System (ECCS) and reactor primary containment heat removal. There are two flat panel displays per division, one on the MCC and the other on the left side of the horizontal portion of the WDP. Flat panel displays for monitoring and control of major non-safety systems are also located on the MCC.

In addition to the touchscreen 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 WDP 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 WDP provides a fixed mimic display, and a large (~250 cm diagonal) variable display. Spatially dedicated alarm windows for critical, plant-level alarms are also provided on the left-hand side WDP. Spatially-dedicated detailed system level annunciators are located above their respective systems on the fixed-mimic display. On the horizontal portion of the WDP, there are multiple flat display devices for individual system surveillance, monitoring and control and specialized functions like main and diesel generator sequencing.

The fixed mimic display is arranged on two, adjacent, upright panels which comprise the center and right-hand sections of the WDP. 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 (although any non safety VDU can also indicate SPDS parameters and monitoring). 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 flat panel displays can also be displayed on the large variable display. Examples of the full color graphic displays that can be shown on the variable display are the various flat panel display formats (including SPDS and plant normal and emergency procedures) which would be selected under plant emergency conditions.

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 and are radiation hardened where appropriate.

The touch-control flat displays located at the base of the WDP 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.

Plant Automation

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

The ESBWR automation design provides for three distinct automation modes: Automatic, Semi-Automatic, and Manual. In the Automatic mode, the operator initiates automated sequences of operation from the MCC. Periodic breakpoints are inserted in the automated sequence 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 the Manual mode of operation, no automated operator guidance or prompts are provided. The operator can completely stop an automatic operation at any time by selecting the Manual mode of operation; this will also happen automatically for any abnormal events, such as turbine trips or reactor scrams.

Operation

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

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

Diversity

To preclude common mode failures and to satisfy NRC requirements, the ESBWR DCIS is deliberately configured using different hardware and software platforms. The NMS and RPS safety systems are diverse from the ECCS safety systems and both are diverse from the investment protection and BOP non safety systems. Further, there is a diverse protection (DPS) system which, although non safety, duplicates many of the RPS scram functions and several ECCS functions but does not use the same hardware/software of those safety systems. Additionally the monitoring and control of the "A" and "B" plant investment protection systems (like the PSW and RWCU/SDC systems) are arranged in different networks such that they can be operated independently (and separate from BOP control and monitoring) should one of the A or B DCIS systems fail. Finally there is a non software based safety system that operates the ATWS and SLCS logic.

Despite these different systems, the DCIS appears seamless to the operator in that all of his interfaces through the VDU displays have the same operating format and menus. The non safety network is such that an A or B plant investment protection system can be normally operated on any control room non safety VDU; however should a DCIS failure occur such that half of the DCIS is lost (more than a single failure is required), the other half will remain operational. There are no common mode or single failures or control room evacuations that will prevent the operators from safely shutting down the plant using either safety or non safety systems. There are no single failures that will result in the loss of power generation.

Surveillance

Unlike previous BWR designs, the ESBWR DCIS is specifically designed to perform tech spec required surveillance tests during normal operation without operator intervention. For example redundant parameters between the four safety divisions are automatically and continuously checked for consistency and alarmed when not consistent; the daily channel checks are either the absence of an alarm or an automatically generated printout of the appropriate parameters. Similarly, although digital setpoint do not "drift", the DCIS automatically and continuously checks the various trip setpoints and alarms when they are not correct.

Maintenance

As with all modern DCIS, the ESBWR incorporates extensive self diagnostics and self calibration and will direct the operator to the lowest level of failed module/card; these can be replaced on line with no loss of power generation.

Chapter 8
Plant Layout and Arrangement

Plant Layout

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

- Reactor Building – houses safety-related structures, systems and components (SSC), except for the main control room, safety-related Distributed Control and Information System equipment rooms in the Control Building and spent fuel storage pool and associated auxiliary equipment in the Fuel Building. The Reactor Building includes the reactor, containment, refueling area and auxiliary equipment.

- Fuel Building – houses the spent fuel storage pool and its associated auxiliary equipment.

- Control Building – houses the main control room and safety-related controls outside the reactor building.

- Turbine Building – houses equipment associated with the main turbine and generator, and their auxiliary systems and equipment, including the condensate purification system and the process offgas treatment system.

- Electrical Building – houses the two non-safety-related standby diesel generators and their associated auxiliary equipment. It also houses the non-safety grade batteries.

- Radwaste Building – houses equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.

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

- Retain the passive and well established BWR pressure suppression containment technology. Use of the horizontal vent configuration was confirmed for the Mark III containments.

- Separate clean and controlled radiation areas to minimize personnel exposure during operation and maintenance.

- Emphasize improved layout of systems to improve access and equipment maintenance activities.

- Locate major equipment for early installation using 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.

- Place the passive safety systems (GDCS, PCCS, ICS) within and adjacent to the primary containment.

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

The site plan of the ESBWR includes the Reactor, Control, Fuel, Turbine, Radwaste, Electrical

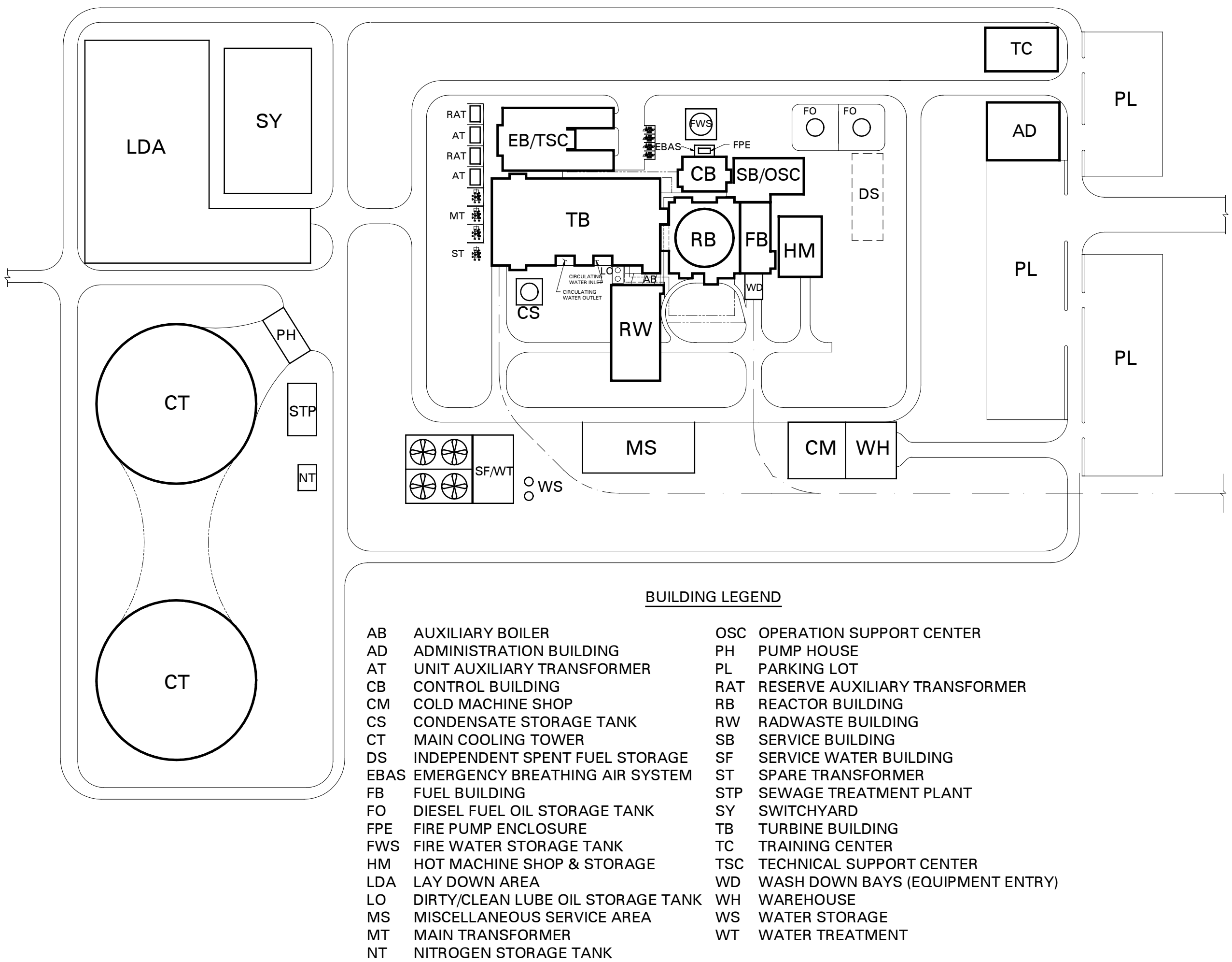


Figure 8-1. ESBWR Site Plan

Buildings and supporting buildings. Provision is made within the Fuel Building for ten years plus a full core offload of spent fuel storage. Separate buildings can be provided for additional onsite waste storage. Figure 8-1 illustrates a site plan of the ESBWR for a single unit arrangement. Although the heat sink is based on cooling towers in the figure, other heat sinks are possible.

A 3-dimensional perspective of most of the power block (the Reactor, Fuel, Control and Turbine Buildings) is shown in Figure 8-2.

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

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

- Personnel access for all normal operating and maintenance activities is a primary concern. Access routes from the change room to contaminated Reactor and Turbine Building areas are as direct as possible and clearly separated from clean routes. At each floor, 360° access is provided, if practical, to enhance daily inspections and normal work activities. Access to equipment not reachable from floor level is via platform and stair access wherever possible.

- Equipment access is provided for all surveil-

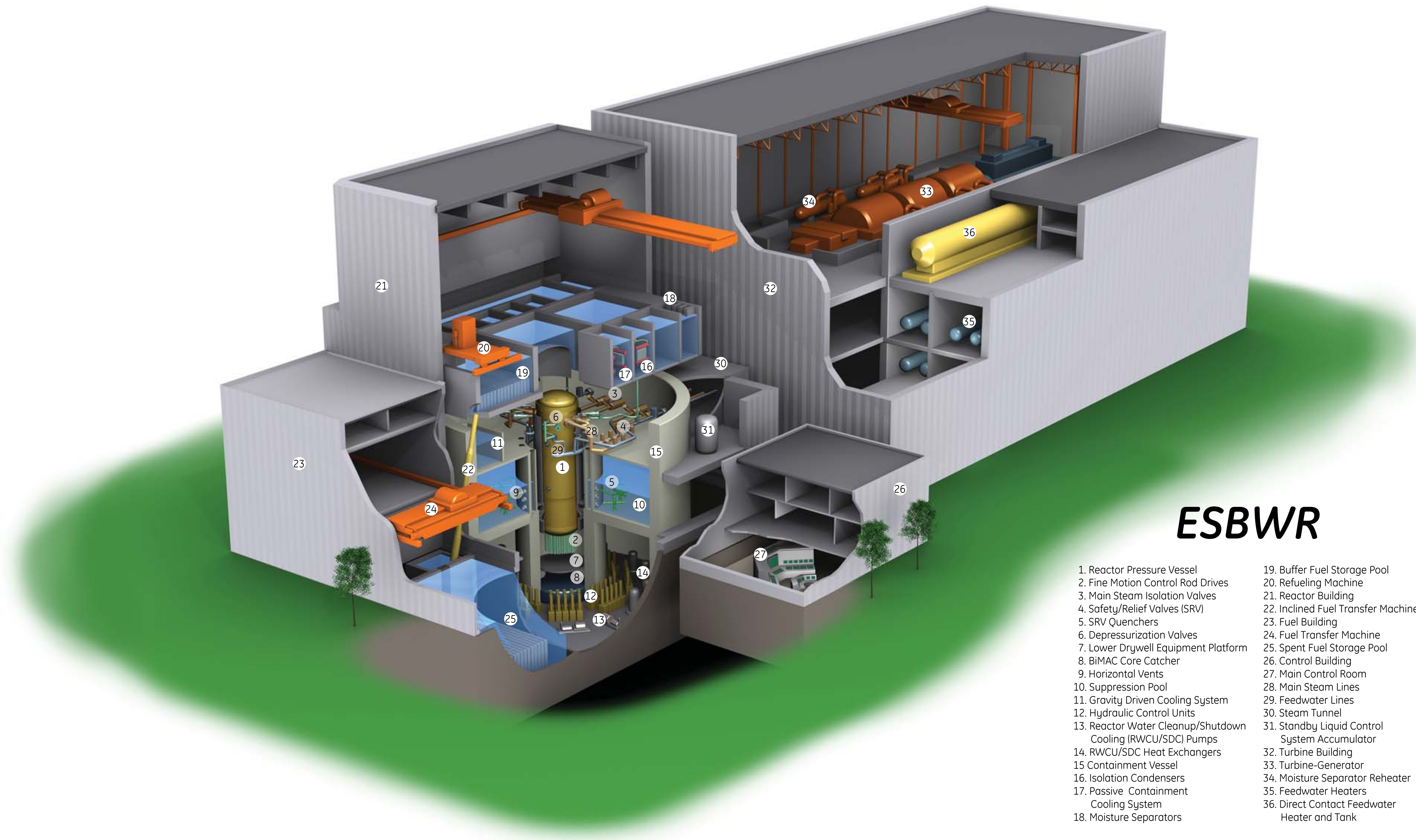


Figure 8-2. ESBWR Cutaway View of the Reactor, Fuel, Control and Turbine Buildings

lance, maintenance and replacement activities with local service areas and laydown space. Adequate hallways and other equipment removal paths, including vertical access hatches, are provided for moving equipment from its installed position to service areas or out of the building for repair. Lifting points, monorails and other installed devices are provided to facilitate equipment handling and minimize the need for re-rigging individual equipment movements. Equipment access also considers the need for temporary construction access.

- Radiation levels are controlled and minimized. The Reactor Building is divided into clean and controlled areas. Once personnel enter a clean or controlled area, it is not possible to crossover to the other area without returning to the change area. Redundant equipment is located in shielded cells to permit servicing one piece of equipment while the plant continues to operate. Valve galleries are provided to minimize personnel exposure during system operation or preparation for maintenance.

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

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

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

Safety Buildings

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

Figure 8-3 shows an elevation view of the Reactor Building and Fuel Building These two buildings have a common basemat. Visible in this view is the Inclined Fuel Transfer System (IFTS) which manages the transfer of fuel between the Reactor Building and the Fuel Building.

Figure 8-4 shows an elevation view of the Reactor Building and adjacent Control Building. Note that only the portion of the Control Building below grade is Seismic Category I; the above grade portion is Seismic Category II. Also seen in this view is the location of the Emergency Breathing Air System (EBAS) structure.

The 4 Isolation Condenser (IC) heat exchangers, 6 Passive Containment Cooling System (PCCS) heat exchangers and their interconnected pools are shown in Figure 8-5.

Main Steam and Feedwater piping and part of the steam tunnel connecting the Reactor Building with the Turbine Building can be seen in Figure 8-6. At this elevation the SRVs and MSIVs can be seen on the Main Steam piping. Also showing at this elevation are the Standby Liquid Control System (SLC) accumulators in the Reactor Building, and HVAC equipment in the Fuel Building.

The drywell-wetwell vent piping and SRV quenchers can be seen in Figure 8-7. Also shown at this elevation is the sliding block support system for the RPV.

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

Figure 8-9 shows the Main Control Room (MCR) general layout.

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

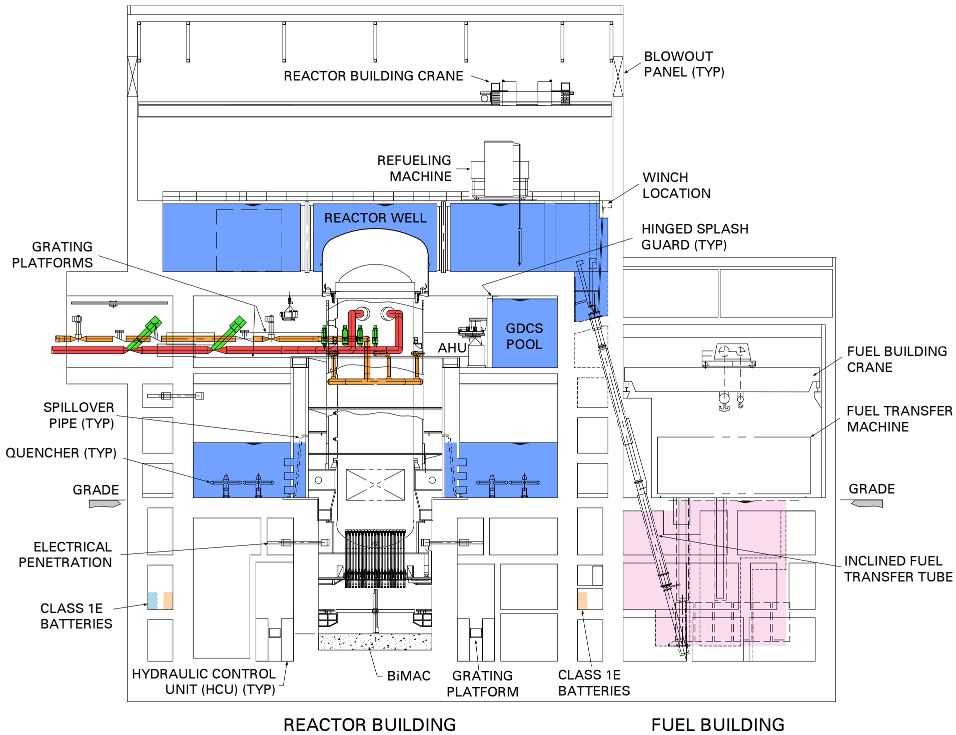


Figure 8-3. ESBWR Reactor and Fuel Building Section AA

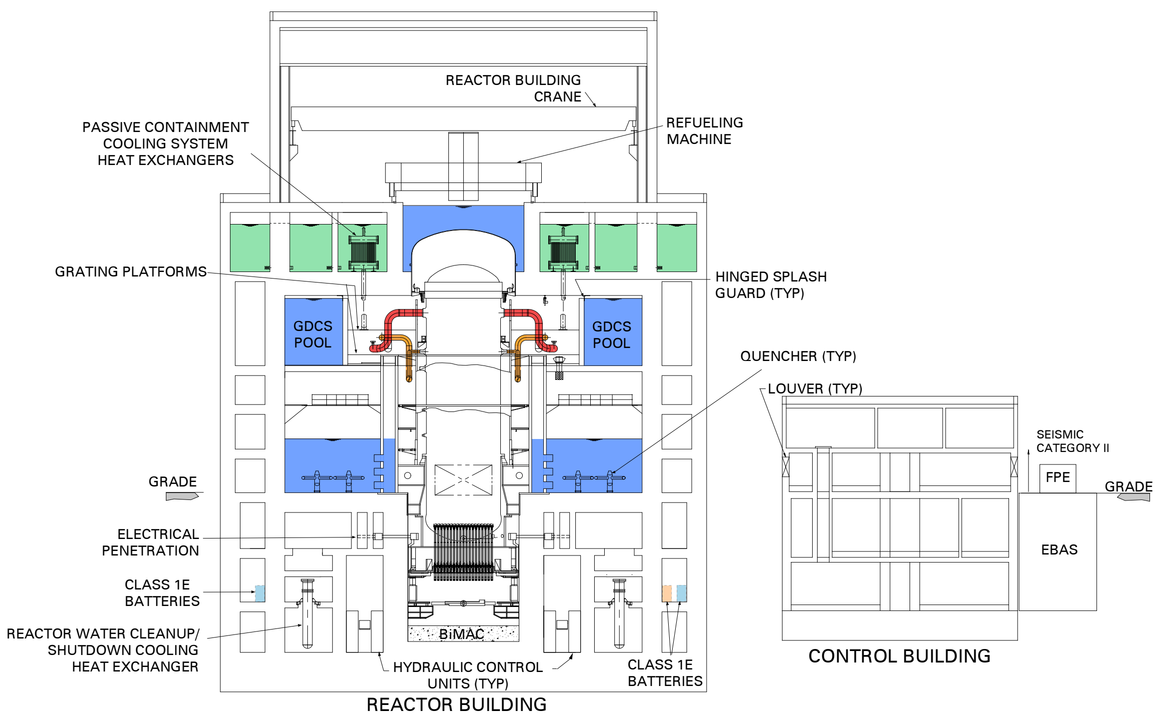


Figure 8-4. ESBWR Reactor and Control Building Section BB

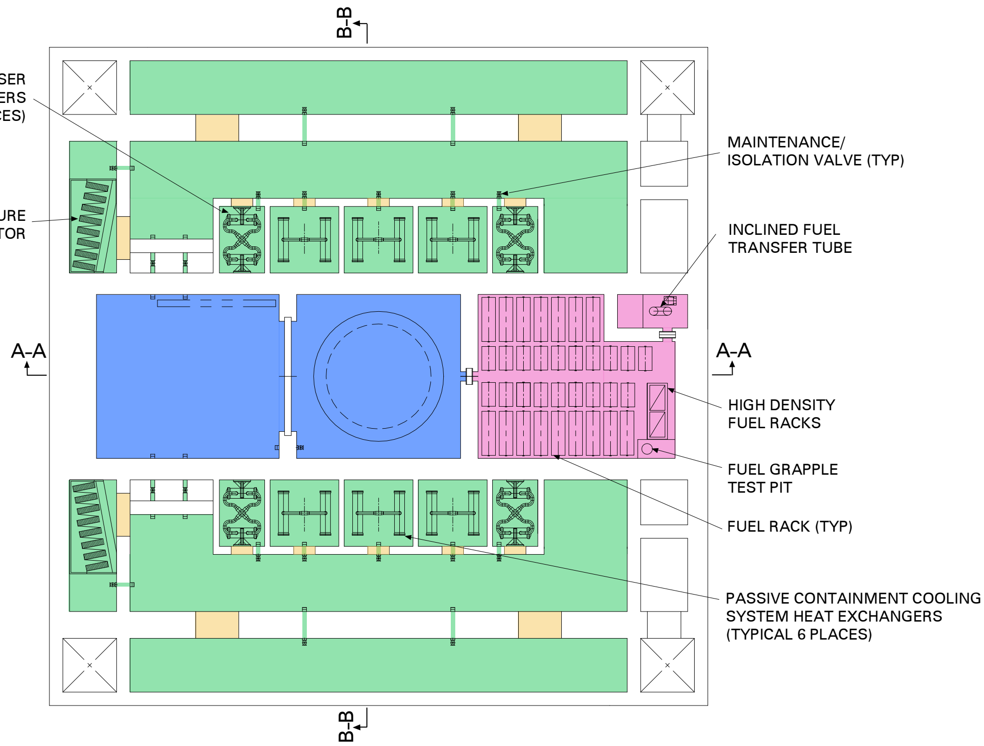


Figure 8-5. ESBWR Reactor Building IC-PCCS Level

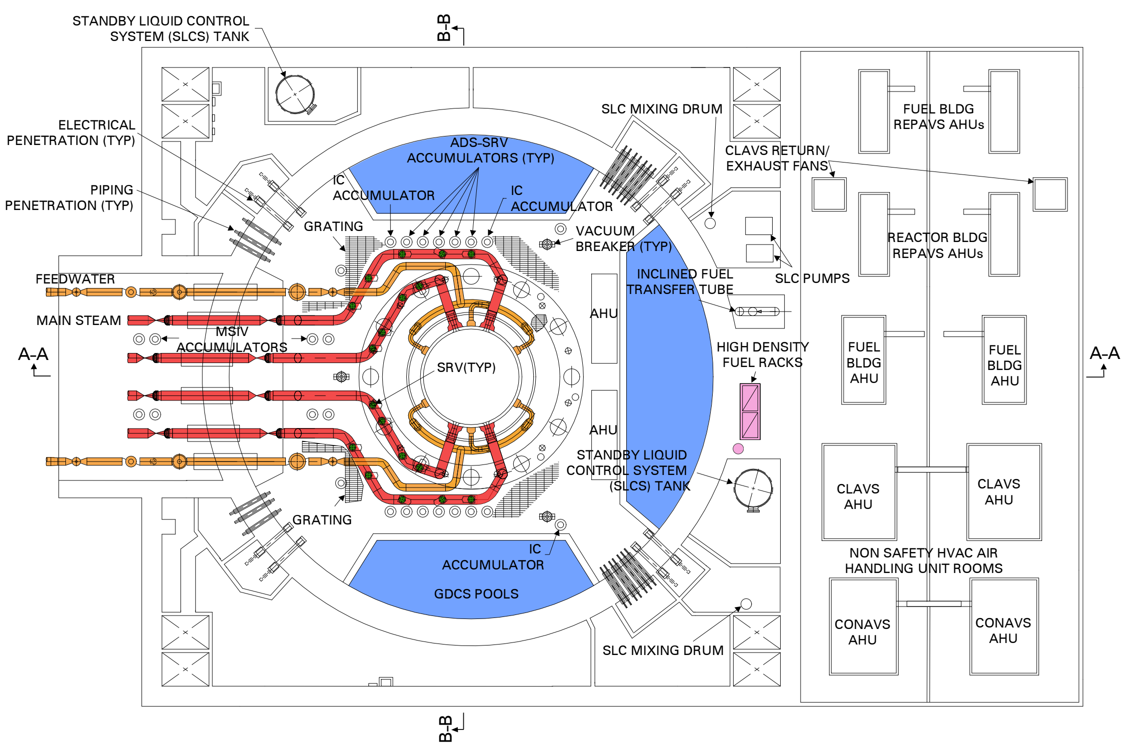


Figure 8-6. ESBWR Reactor and Fuel Building Steam Line Level

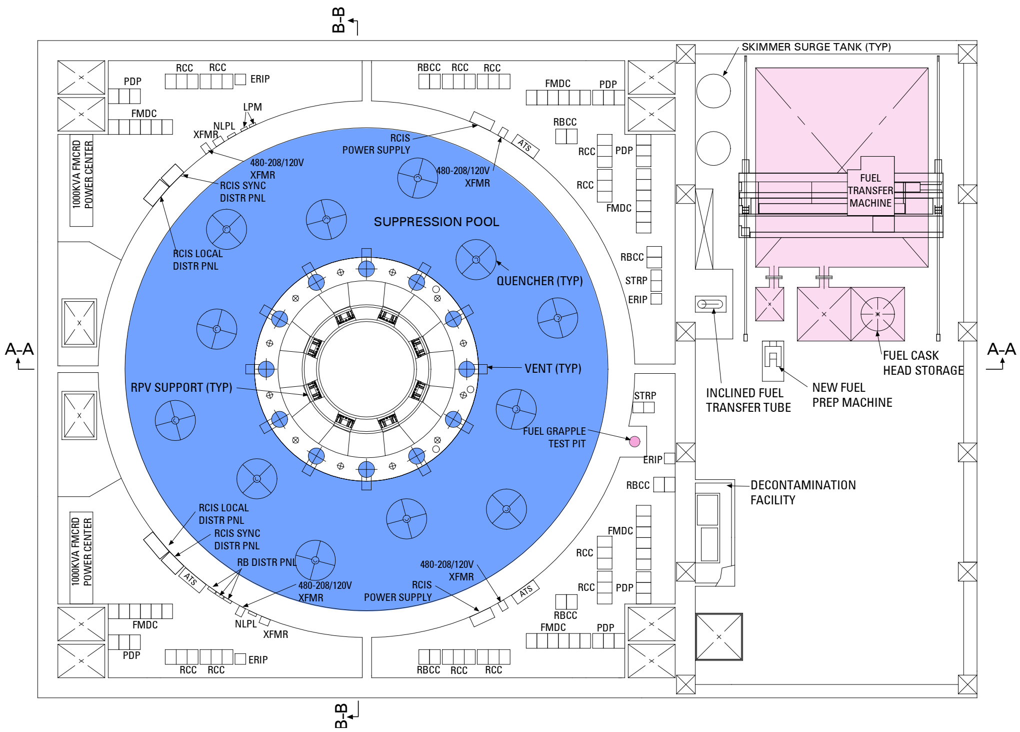


Figure 8-7. ESBWR Reactor and Fuel Building Suppression Pool Level

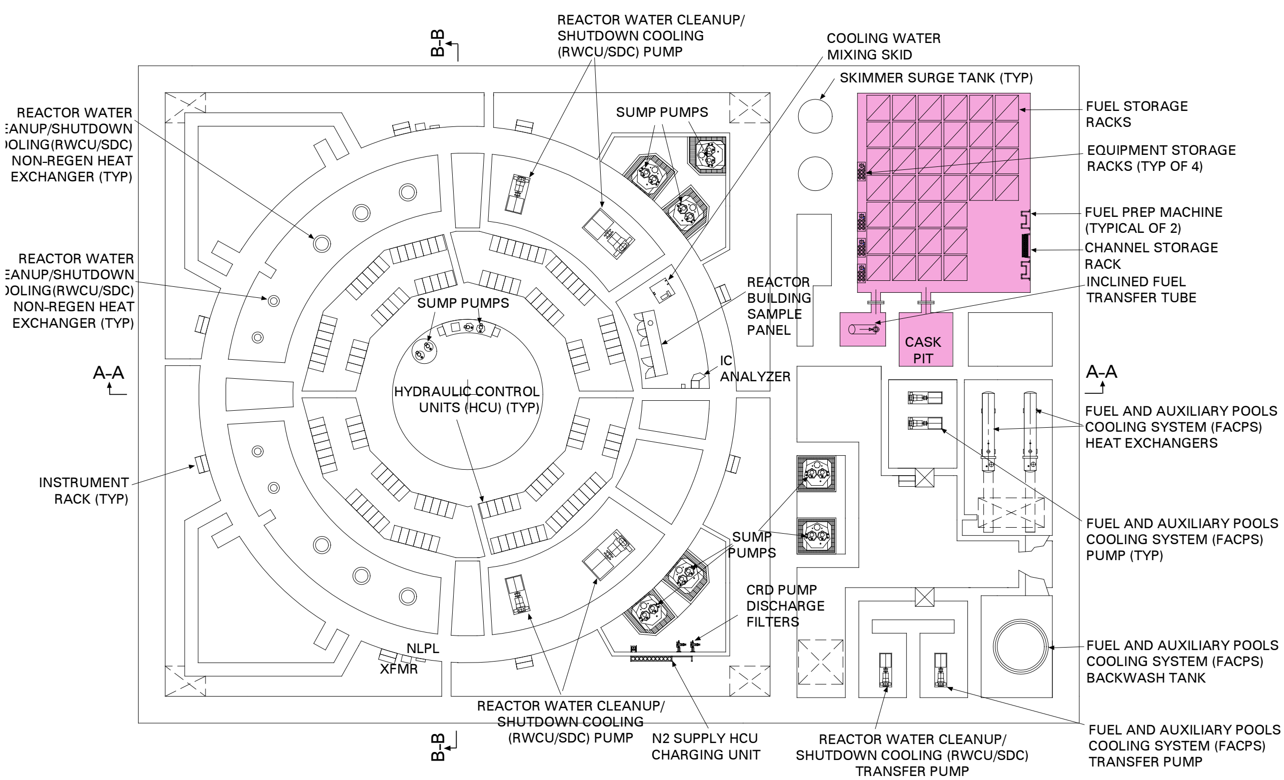


Figure 8-8. ESBWR Reactor and Fuel Building Basemat Level

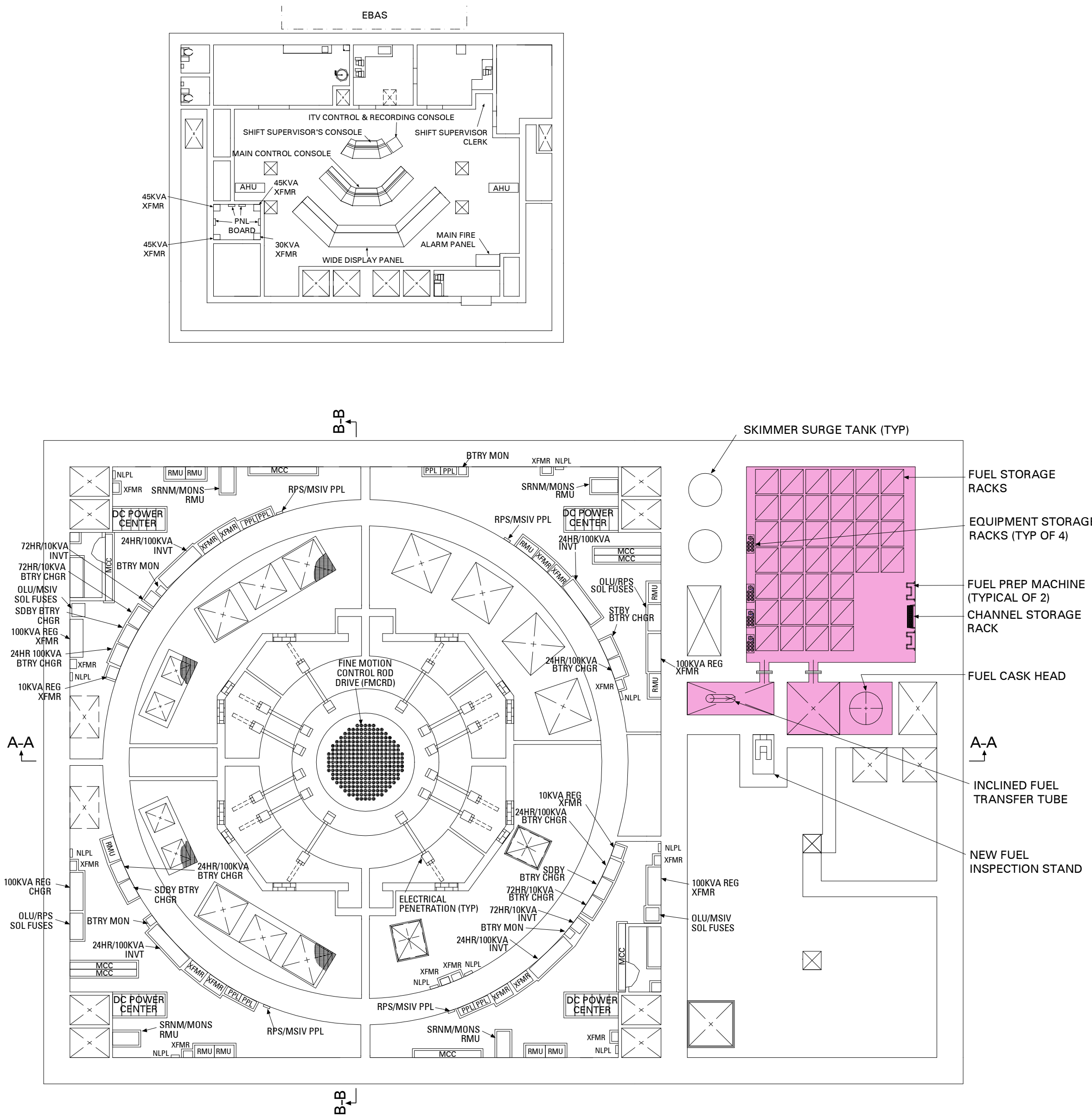


Figure 8-9. ESBWR Reactor, Fuel and Control Buildings MCR Level

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

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

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

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

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

Analyses of the radiological dose consequences for design basis accidents, based on an assumed containment leak rate of 0.5% per day, show that the offsite doses after an accident are about 5 Rem TEDE for the standard US site (see Chapter 11).

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

- Elimination of the recirculation system, which reduces the containment volume associated with high construction costs.
- Reduced building volume which reduces material costs and construction schedule.
- Design with conventional structural shapes to improve constructability which reduces capital costs and construction schedule.
- Improved personnel and equipment access for enhanced operability and maintainability.

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

The volume of the ESBWR Safety Buildings is reduced to approximately 200,000 m^3, a 20% reduction compared to ABWR. Since this reduced volume was obtained by simplification of the reactor supporting systems and optimization of their arrangement with improved access (rather than simply by compaction), it provides attractive material cost savings over previous BWRs and helps reduce the construction schedule without adversely impacting maintenance.

Inclined Fuel Transfer System

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

The ESBWR is equipped with a non-safety grade, but Seismic Category I, Inclined Fuel Transfer System (IFTS). In general the arrangement of the IFTS (refer to Figure 8-10) consists of a terminus at the upper end in the Reactor Building buffer pool that allows the fuel to be placed in a fuel transport carriage and tilted from a vertical position to an inclined position prior to transport to the Spent Fuel Pool. There is a means to lower the fuel transport carriage, means to seal off the top end of the transfer

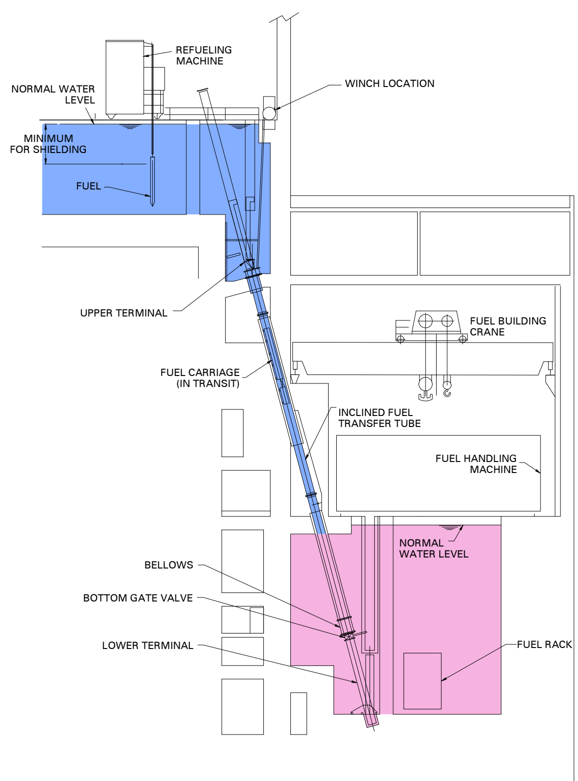


Figure 8-10. ESBWR Inclined Fuel Transfer System

tube, and a control system to effect transfer. It has a lower terminus in the fuel building storage pool, and a means to tilt the fuel transport carriage into a vertical position allowing it to be removed from the transport device. There are controls contained in local control panels to monitor the transfer, opening and closing valves, and raising or lowering the fuel transport carriage. There is a means to seal off the upper and lower end of the tube while allowing filling and venting of the tube.

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

The IFTS terminates in a separate pit in the fuel storage pool. The lower terminus of the IFTS allows for thermal expansion (axial movement relative to the anchor point in the Reactor Building). The lower terminus allows for differential movement between the anchor point in the Reactor Building and the fuel pool terminus, and allows it to have rotational movement at the end of the tube relative to the anchor point in the Reactor Building. The lower end interfaces with the fuel storage pool with a bellows to seal between the transfer tube and the Spent Fuel Pool wall.

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

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

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

Primary Containment System

The ESBWR containment is a low-leakage reinforced concrete structure with an internal steel liner in the drywell and suppression chamber to serve as a leak-tight membrane. The containment is a cylindrical shell structure, which consists of the reactor pressure vessel (RPV) pedestal, the containment cylindrical wall, the top slab, the suppression pool slab and the foundation mat. The containment is divided by the diaphragm floor and the vent wall into a drywell chamber (DW) and a suppression chamber, or wetwell (WW). The top slab of the containment is an integral part of the Isolation Condenser/Passive Containment Cooling (IC/PCC) pools and the services pools for storage of Dryer/Separator and other uses. The pool girders, which serve as barriers of the pools, rigidly connect the top slab and the Reactor Building (RB) walls. The RB floors that surround the containment walls and walls that are under the suppression pool floor slab are also integrated structurally with the concrete containment. The containment foundation mat is continuous with the RB foundation mat, and the Fuel Building (FB) as well. The containment and the structures integrated with the containment are constructed of cast-in-place, reinforced concrete. The containment system is designed to have the following functional capabilities (Figure 8-11):

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

- The containment has capability for rapid closure or isolation of all pipes and ducts that penetrate the containment boundary in order to maintain leak tightness within acceptable limits.
- The containment structure and isolation, with concurrent operation of other accident mitigation systems, is designed to limit fission product leakage during and following the postulated DBA to values less than leakage rates that could result in offsite radiation doses greater than those set forth in 10CFR50.67.
- 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 Design Basis Accident (DBA).
- The containment structure design provides means to channel the flow from postulated pipe ruptures in the DW to the suppression pool.

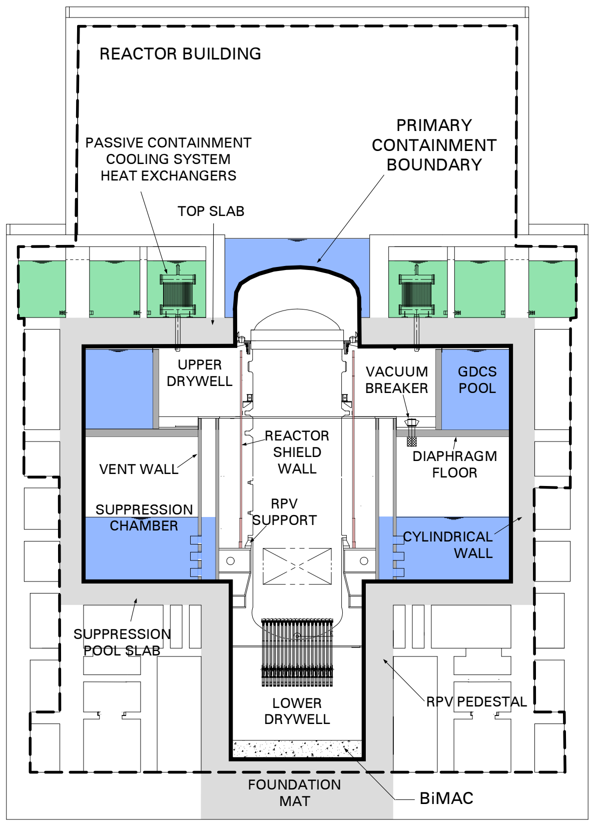


Figure 8-11. ESBWR Reactor Building and Containment

Drywell Structure

The drywell (DW) is comprised of two volumes:

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

The upper DW is a cylindrical, reinforced concrete structure with a removable steel head and a diaphragm floor constructed of steel girders with concrete fill. The RPV sliding block support system supports the RPV while accommodating relative thermal expansion and has openings for communication between the UD and LD. Refer to Figure 8-12. There are eight sliding block supports. One end of each sliding support is fastened to a circumferential RPV flange segment that is forged integral to the vessel shell ring at that RPV elevation. The other end of each sliding block is restrained by sets of steel guide blocks that are attached to the reactor pedestal support brackets. Under this configuration, each sliding support is relatively free to expand in the radial direction but is restrained in the vertical and vessel tangential directions.

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

Wetwell Structure

The wetwell (WW) is comprised of a gas volume and suppression pool filled with water to rapidly condense steam from a reactor vessel blowdown via the SRVs or from a break in a major pipe inside the drywell through the vent system. The WW is connected to the DW by a vent system comprising twelve (12) vertical/horizontal vent modules. Each module consists of a vertical flow steel pipe, with three horizontal vent pipes extending into the suppression pool water. Each vent module is built into the vent wall, which separates the DW from the WW. The cylindrical vent wall is supported off the RPV pedestal. The WW boundary is the annular

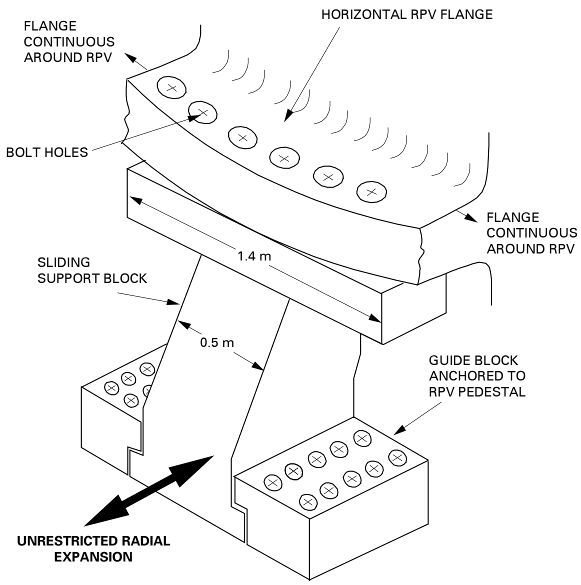


Figure 8-12. RPV Sliding Block Support

region between the vent wall and the cylindrical containment wall and is bounded above by the DW diaphragm floor. All normally wetted surfaces of the liner in the WW are stainless steel and the rest are carbon steel.

Containment Structure

The containment structure includes a steel liner to reduce fission product leakage. All normally wetted surfaces of the liner in the suppression pool are made of stainless steel. Penetrations through the liner for the drywell head, equipment hatches, personnel locks, piping, and electrical and instrumentation lines are provided with seals and leak-tight connections. The allowable leakage is 0.5% per day from all sources, excluding main steam isolation valve (MSIV) leakage.

Containment System

The DW is designed to withstand the pressure and temperature transients associated with the rupture of any primary system pipe inside the DW, and also the negative differential pressures associated with containment depressurization events, when the steam in the DW is condensed by the PCCS, the GDCS, the Fuel and Auxiliary Pools Cooling System (FAPCS), and cold water cascading from the break following post-LOCA flooding of the RPV.

In the event of a pipe break within the DW, the increased pressure inside the DW forces a mixture of noncondensable gases, steam and water through either the PCCS or the vertical/horizontal vent pipes and into the suppression pool where the steam is rapidly condensed. The noncondensable gases transported with the steam and water are contained in the free gas space volume of the WW. The design pressure of the containment is 310 kPa(g) (45 psig).

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

To help manage the water within containment there are 12 spillover pipes connecting the lower part of the UD with the SP. The actual water levels in the RPV and containment compartments will depend on the LOCA break location. For example, Figure 8-13 shows the water levels in the GDCS, RPV, DW and SP in the long term after a FW pipe break LOCA, and Figure 8-14 shows the water levels after an RPV bottom drain line break LOCA.

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

Control of potential hydrogen generation in an accident is handled by maintaining an inerted containment atmosphere via the Containment Inerting System (CIS) which inerts the containment atmosphere with nitrogen to maintain < 3% oxygen- see Chapter 5.

Containment Heat Removal

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

Isolation transients do not present a heat removal

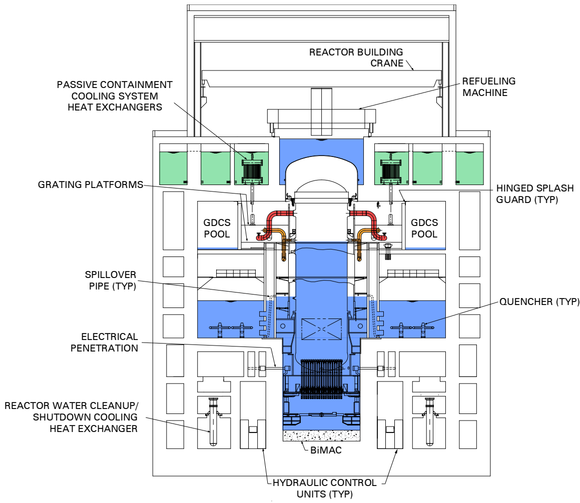


Figure 8-13. Water Levels After a FW Line Break

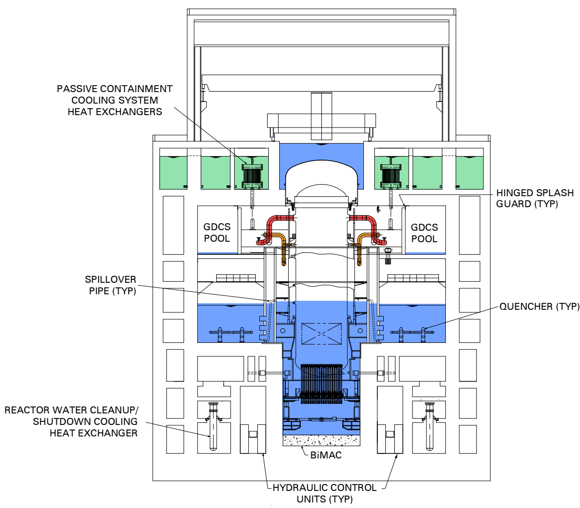


Figure 8-14. Water Levels After a Bottom Drain Line Break

challenge to the ESBWR containment, due to the use of the Isolation Condenser System (ICS). See Chapter 3 for more details.

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

After an accident, the non-safety related Fuel and Auxiliary Pools Cooling System (FAPCS) may be available in the suppression pool cooling mode and/or containment spray mode to control the containment pressure and temperature conditions. Heat is removed via the FAPCS heat exchanger(s) to the Reactor Component Cooling Water System (RCCWS) and finally to the Plant Service Water System (PSWS). These systems are described in Chapter 5.

Vacuum Breakers

A vacuum breaker system has been provided between the DW and WW. The purpose of the DW-to-WW vacuum breaker system is to protect the integrity of the diaphragm floor slab and vent wall between the DW and the WW, and the DW structure and liner, and to prevent backflooding of the suppression pool water into the DW. Refer to Figure 8-15.

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

Each vacuum breaker is provided with redundant proximity sensors to detect its closed position. On the upstream side of the vacuum breaker, a DC solenoid operated butterfly valve designed to fail-close is provided. During a LOCA, when the vacuum breaker opens to equalize the DW and WW pressure and subsequently does not completely close as detected by the proximity sensors, a control signal will close the upstream butterfly valve to prevent extra bypass leakage due to the opening created by the vacuum breaker and therefore maintain the pressure suppression capability of the containment. Plant operators can also manually close the backup valve. Redundant vacuum breaker systems are provided to protect against a single failure.

The ESBWR vacuum breaker has undergone engineering development testing using a full-scale prototype to demonstrate the proper operability, reliability, and leak-tightness of the design. Figure 8-16 shows one of the modules tested.

Severe Accident Mitigation

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

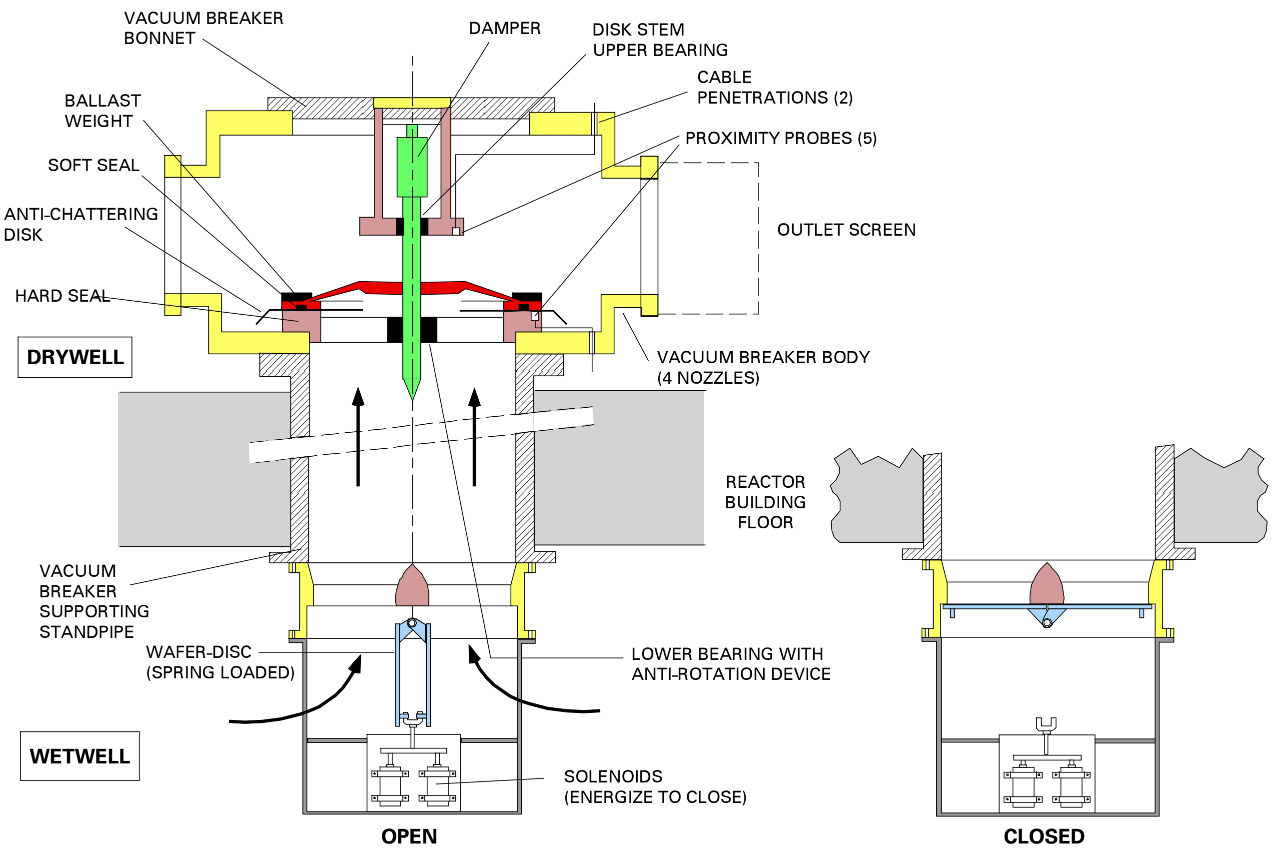


Figure 8-15. ESBWR Wetwell-to-Drywell Vacuum Breaker with Backup Closure Valve

Figure 8-16. Prototype Vacuum Breaker

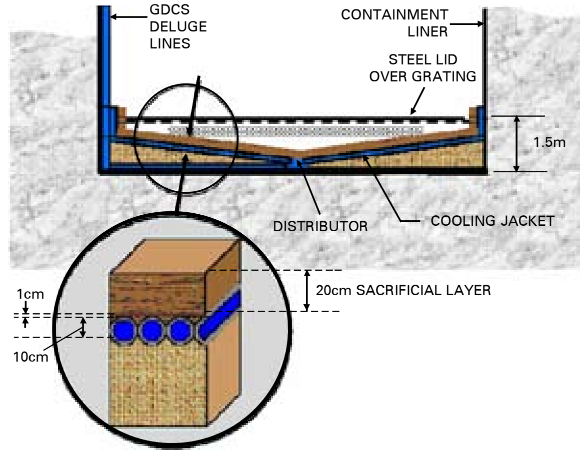


Figure 8-17. BiMAC Concept

basemat and would be called into action only in the event that some or all of the core debris on top is non-coolable.

The device, called Basemat Internal Melt Arrest and Coolability device (BiMAC), is illustrated in Figure 8-17. Important considerations in implementation of this concept are as follows:

Pipe inclination angle. An inclination in the range of 10º promotes excellent heat removal.

Sacrificial refractory layer. A refractory material is laid on top of the BiMAC pipes so as to protect against melt impingement during the initial (main) relocation event, and to allow some adequately short time for diagnosing that conditions are appropriate for flooding. This is to minimize the chance of inadvertent, early flooding. The material is selected to have high structural integrity, and high resistance to melting such as ceramic Zirconia.

Cover plate. As shown in Figure 8-17, a supported steel plate covers the BiMAC. On the one hand this allows that the top is a normal floor as needed for operations, and that the BiMAC is basically "out of the way" until its function is ever needed. On the other hand the so-created cavity, with a total capacity of ~100 m^3, is there to receive and trap the melt in a hypothetical ex-vessel severe accident evolution, including a high pressure melt ejection. For this purpose the top plate is stainless steel of thickness such as to be essentially instantaneously penetrable by a high-velocity melt jet. The plate is made to sit on top of normal floor grating, which itself is supported by steel columns as indicated schematically in Figure 8-16. Between the plate and the grating there is a layer of refractory material, such as a mat of zirconium oxide, so as to protect the steel material from thermal loads from during the ~ 40 seconds steam blowdown period, yet not able to provide any structural resistance to melt penetration as needed for the trapping function noted above. For low pressure severe accident sequences, this whole cover structure has no bearing on the outcome.

The BiMAC cavity. The space available below the BiMAC plate is sufficient to accommodate the full-core debris, and the entire coolable volume, up to the height of the vertical segments of the BiMAC pipes is ~400% of the full-core debris. Thus there is no possibility for the melt to contact the LDW liner. Similarly, the two sumps needed for detecting leakage flow during normal operation, are positioned and protected, as is the rest of the LDW liner, from being subject to melt attack.

The LDW deluge system. This system consists of three main lines that feed off the three independent GDCS pools, respectively, each separating into a pair of lines that connect to the BiMAC main header.

Turbine Building

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

The TB is a Seismic Category II, non-safety building. Figures 8-18 through 8-22 show various elevation and plan views.

Electrical Building

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

Radwaste Building

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

Other Principal Buildings

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

Fire Protection

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

Plant Arrangement

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

Divisional Separation

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

The remote shutdown panels provide redundant control of the safe shutdown function from outside the control room in case the control room becomes uninhabitable.

Fire Containment

Fire containment is achieved through the use of concrete fire barrier floors, ceilings and walls designed to contain a fire for a duration of three hours without structural failure. Fire dampers

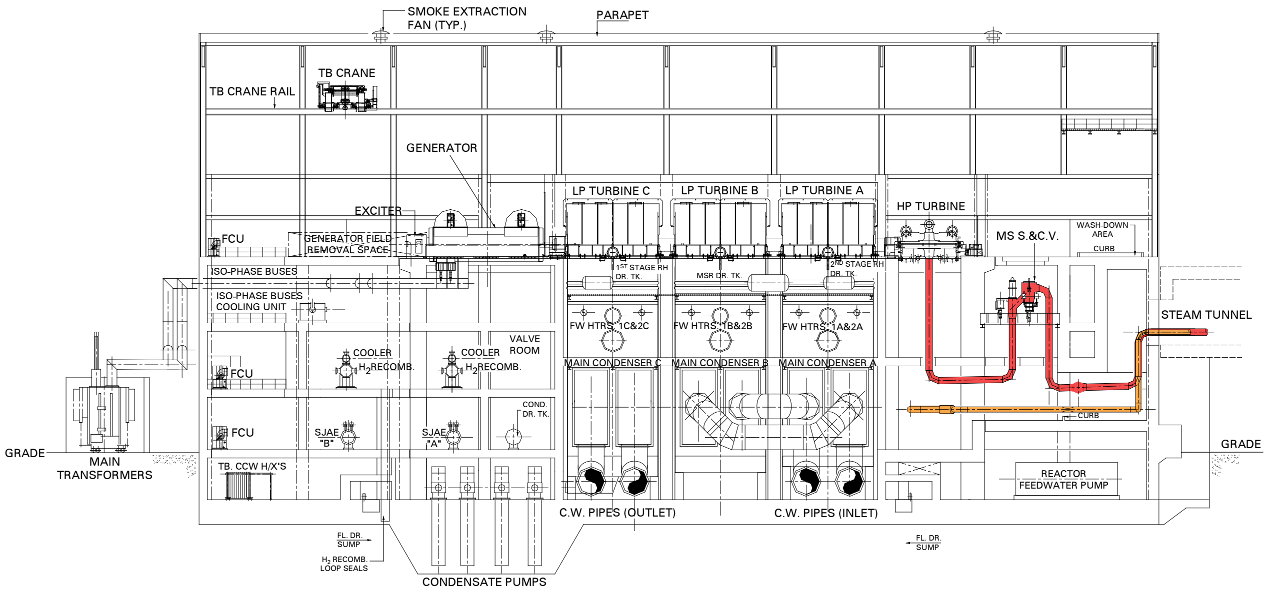


Figure 8-18. ESBWR Turbine Building Section BB

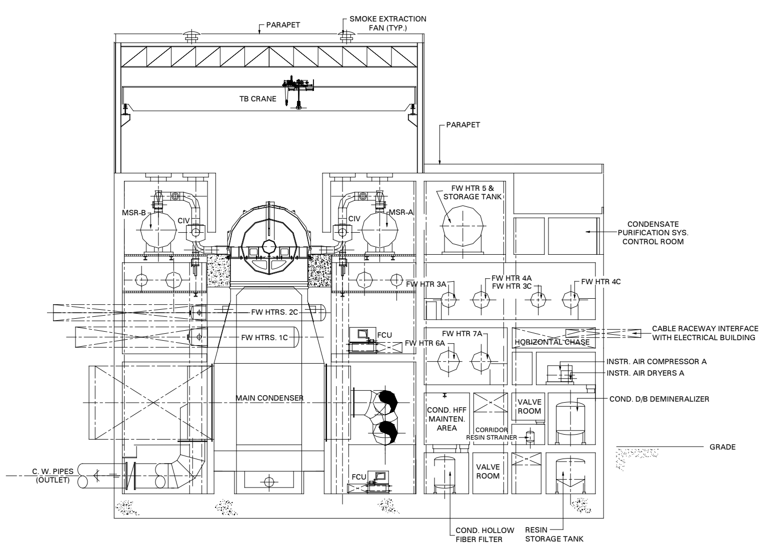


Figure 8-19. ESBWR Turbine Building Section AA

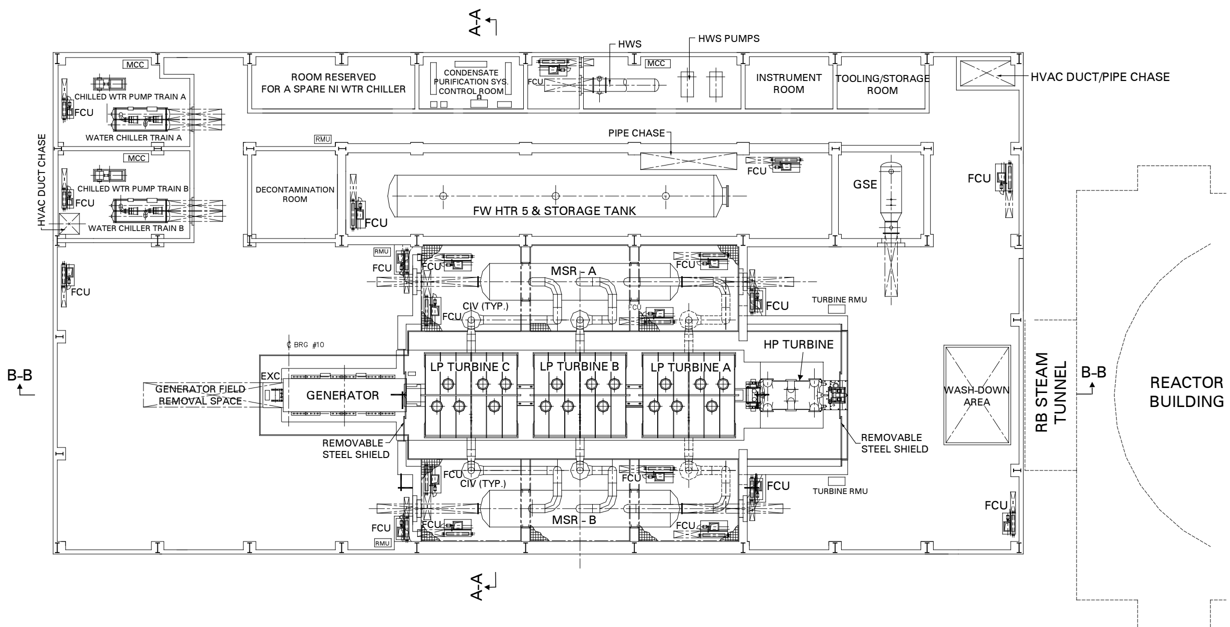


Figure 8-20. ESBWR Turbine Building Operating Floor

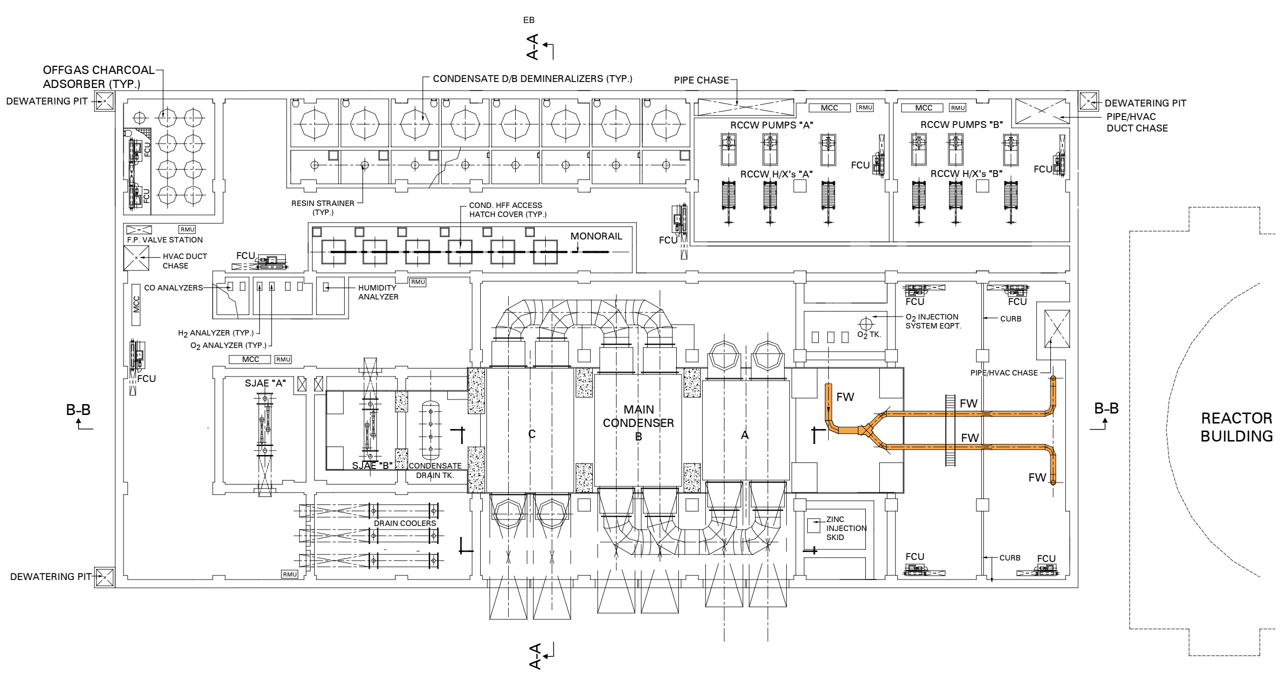


Figure 8-21. ESBWR Turbine Building Grade Elevation

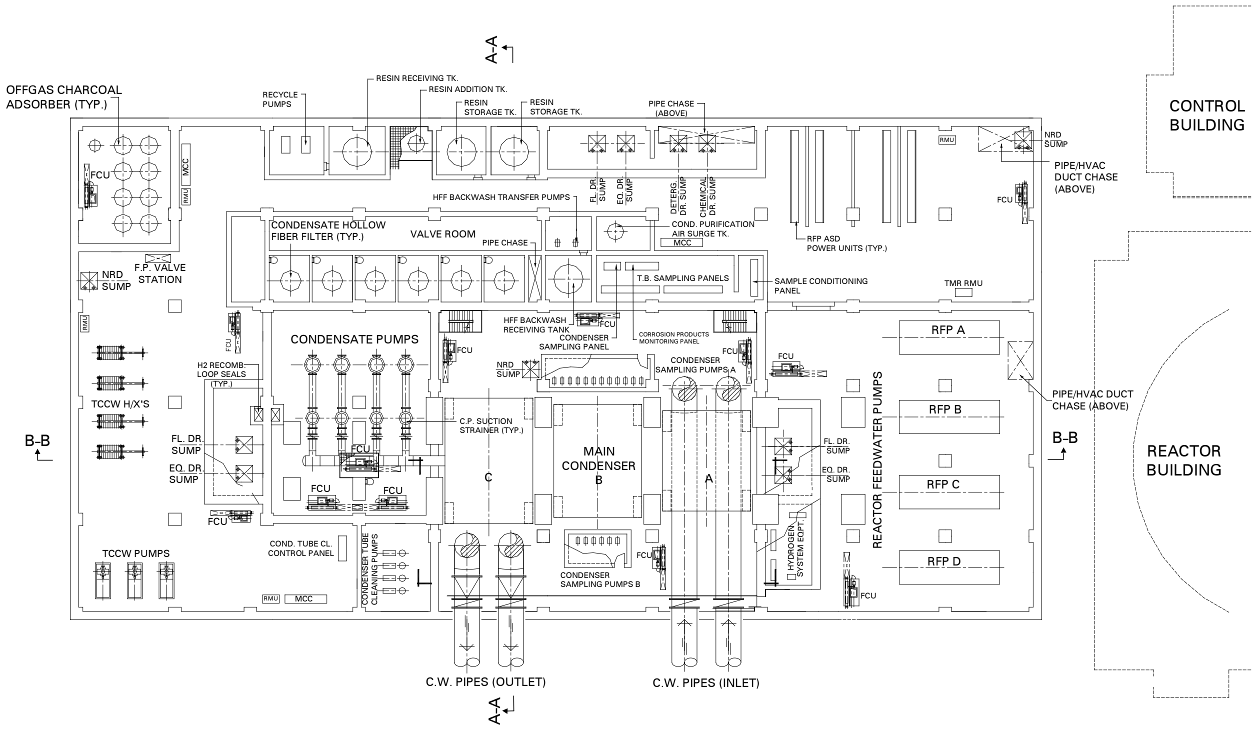


Figure 8-22. ESBWR Turbine Building Basement Level

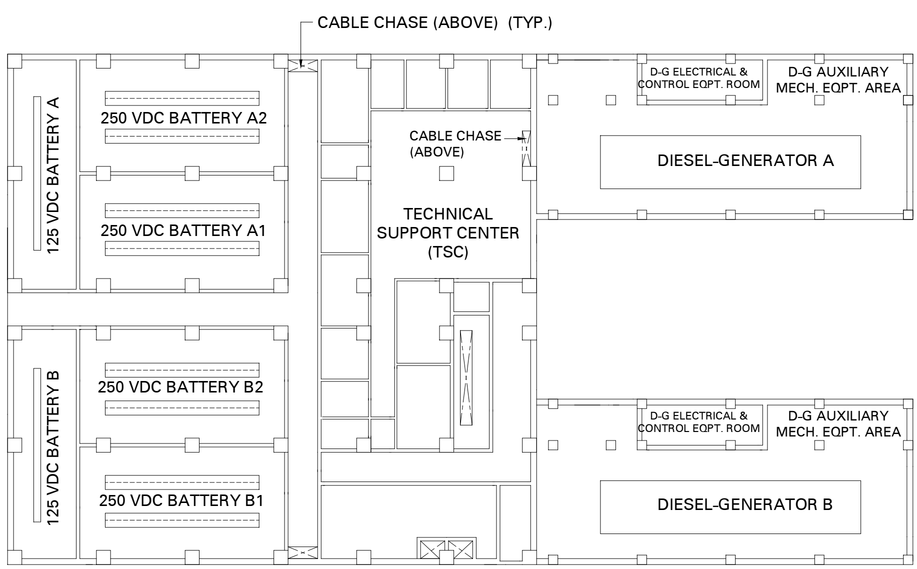


Figure 8-23. ESBWR Electrical Building Grade Level

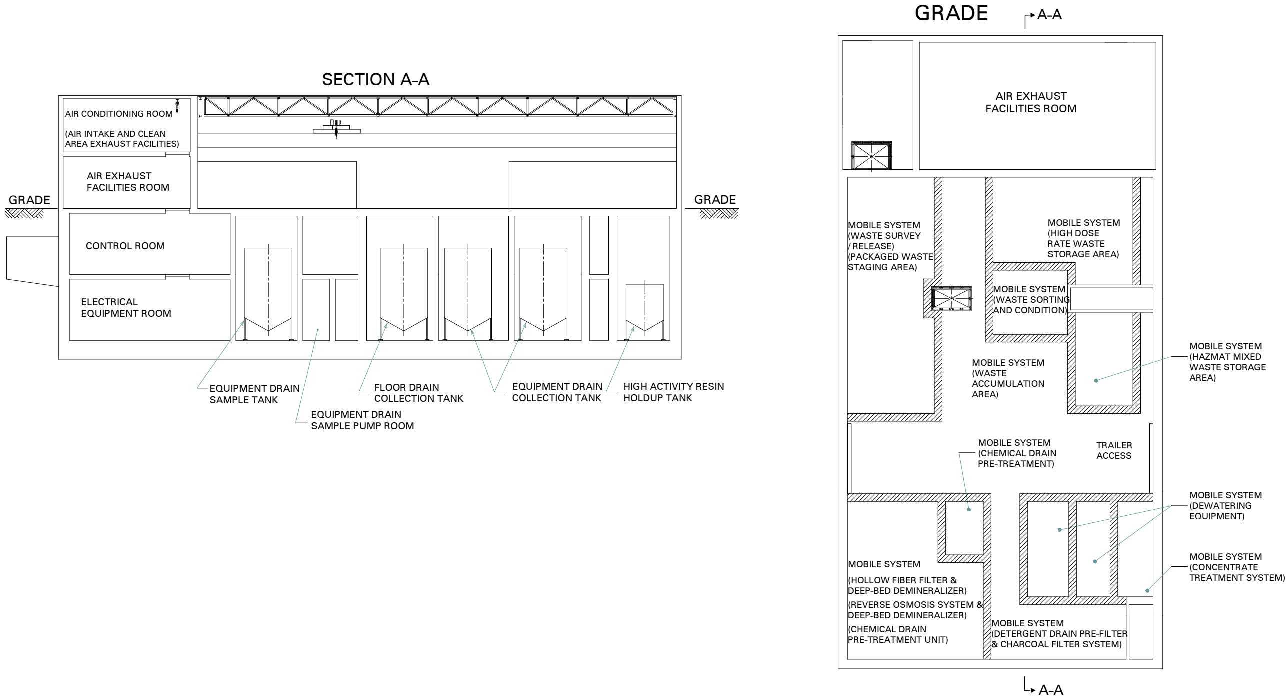


Figure 8-24. ESBWR Radwaste Building Section and Grade Elevation

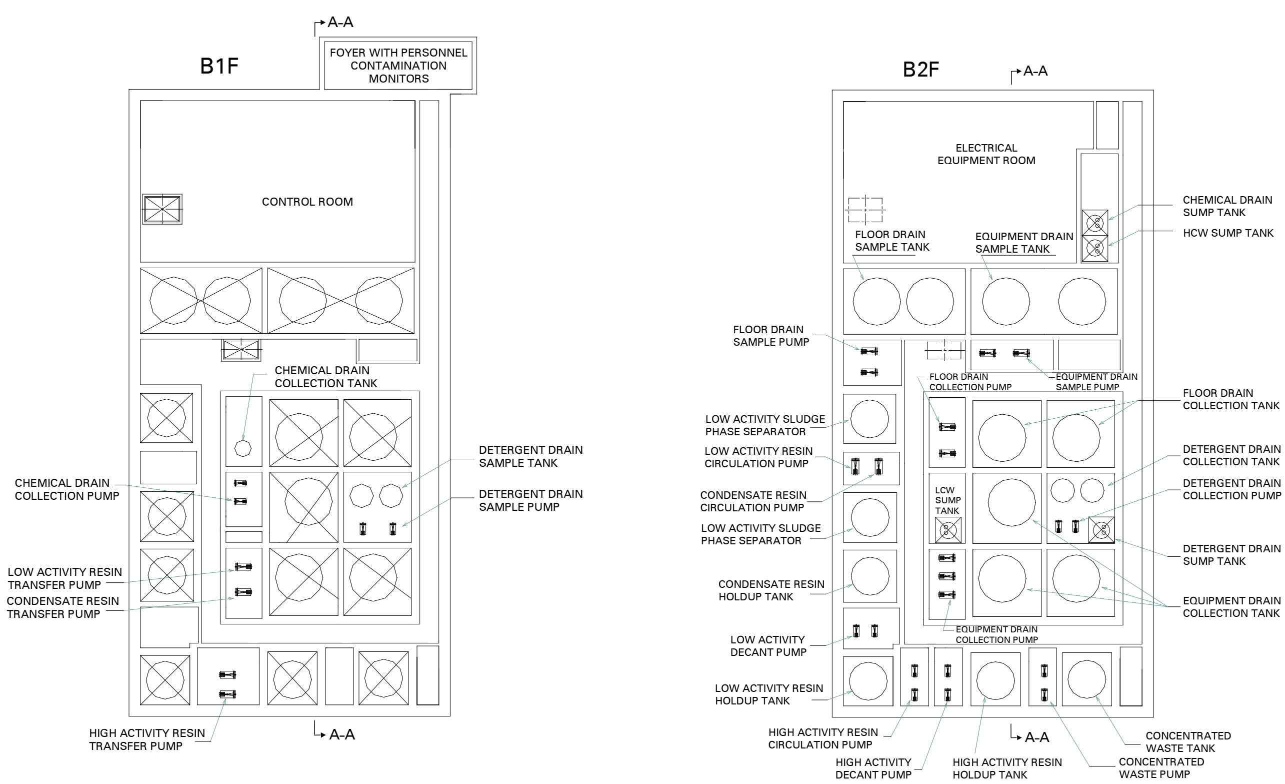


Figure 8-25. ESBWR Radwaste Building Basement Levels

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

Flood Protection

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

Flood Protection from External Sources

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

Flood Protection from Internal Component Failures

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

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

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

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 (UHS) location and temperature, and the offsite power distribution system which can play a significant role in the selected configuration. What follows, therefore, is an example configuration, showing one possible implementation. Changes in this part of the plant will not have any significant impact on the Nuclear Island design or operation.

Steam and Power Conversion System

The Turbine Building houses all equipment associated with the main turbine generator and other auxiliary equipment. The turbine employs a conventional regenerative cycle with condenser deaeration and condensate demineralization. The turbine-generator is equipped with an electrohydraulic control system and supervisory instruments to monitor performance. The gross electrical output of the turbine-generator is approximately 1550 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 purification system, and the condensate and feedwater pumping and heating system. The heat rejected to the main condenser is removed by a circulating water system and discharged to the power cycle heat sink.

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

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

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

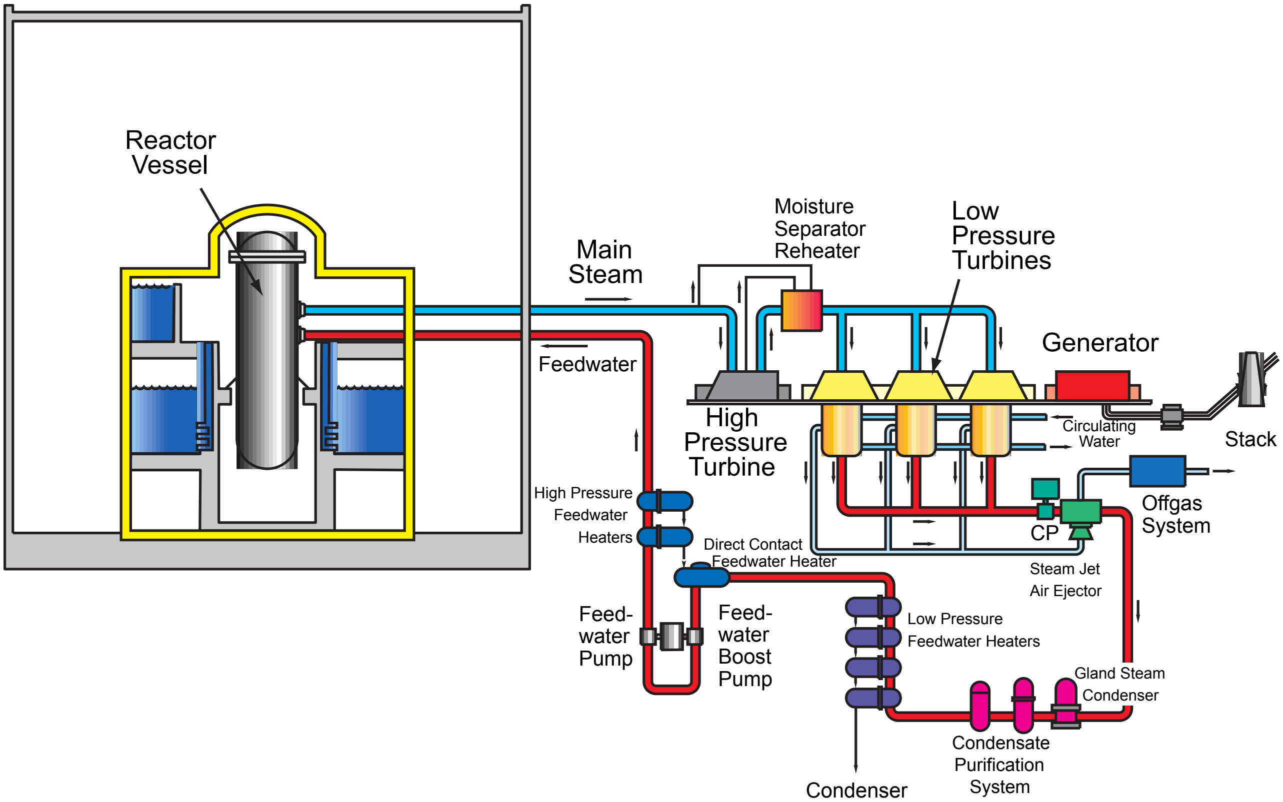


Figure 9-1. ESBWR Steam and Power Conversion System

site conditions.

Normally, the turbine power heat cycle utilizes all the steam being generated by the reactor; however, an automatic pressure-controlled turbine bypass system designed for 100% of the rated steam flow is provided to discharge excess steam directly to the condenser. This allows a loss of full load with the ability to drop to house load without a turbine overspeed trip or a reactor scram.

Turbine Main Steam Systems

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

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

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

The drains from the steamlines inside containment are connected to the steamlines outside the containment to permit equalizing pressure across the MSIVs during startup and following a steamline isolation.

Main Turbine/Generator

The turbine-generator consists of an 1800 rpm turbine, moisture separator/reheaters, generator, exciter, controls, and associated subsystems.

The turbine for the ESBWR reference plant consists of a double-flow, high-pressure unit, and three double flow low-pressure units in tandem. The high-pressure turbine has two stages of steam extraction.

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

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

The generator is a direct driven, three-phase, 60 Hz, 1800 rpm synchronous generator with a water-cooled stator and hydrogen cooled rotor.

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

TG accessories include the bearing lubrication oil system, turbine control system (TCS) , turning gear, hydrogen and CO_2 system, seal oil system, stator cooling water system, exhaust hood spray system, turbine gland sealing system, and turbine supervisory instrument system.

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

The gross electrical output of the turbine generator is approximately 1550 MWe. For utilities generating 50 Hz power, the turbine shaft speed is 1500 rpm.

Main Condenser

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

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

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

The condenser shells are located below the Turbine Building operating floor and are supported on the Turbine Building basemat (see Chapter 8). Failure of or leakage from a condenser hotwell during plant shutdown only results in a minimum water level in the Turbine Building condenser area.

Expansion joints are provided between each turbine exhaust opening and the steam inlet connections of the condenser shell. Water seals and their level indication, if required, are provided around the entire outside periphery to prevent leakage through the expansion joints. Level indication provides detection of leakage through the expansion joint. Two low-pressure feedwater heaters are located in the steam dome of each shell. Piping is installed for hotwell level control and condensate sampling.

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

Main Condenser Evacuation System

The Main Condenser Evacuation System (MCES) removes the 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, Steam Jet Air Ejector (SJAE) units (complete with intercondenser) for power plant operation where one SJAE unit is normally in operation and the other is on standby, as well as two 50% capacity mechanical vacuum pumps for use during startup. The last stage of the SJAE is a non-condensing 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 pumps establish a vacuum in the Main Condenser and other parts of the power cycle. The discharge from the vacuum pumps is then routed to the Turbine Building Compartment Exhaust (TBCE) system, since there is then little or no effluent radioactivity present. Radiation detectors in the TBCE system and plant vent stack alarm in the Main Control Room (MCR) if abnormal radioactivity is detected. Radiation monitors are provided on the main 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 operation, the SJAEs are normally driven by main steam, with the auxiliary steam supply system on automatic standby. The main steam supply, however, is normally used during startup and low load operation, and auxiliary steam is available for normal use of the SJAEs during early startup, as an alternative to the main steam or should the mechanical vacuum pumps prove to be unavailable.

Turbine Gland Steam System

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

Turbine Bypass System

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

The TBS consists of twelve Turbine Bypass Valves (TBV) mounted on four chests (three valves per chest) connected to the TMSS Main Steam Line equalizer. The outlets of TBVs are connected to the Main Condenser via pressure reducers.

The system is designed to bypass at least 110%

of the rated main steam flow directly to the condenser. The TBS, in combination with the reactor systems, provides the capability to shed 100% of the T-G rated load without reactor trip and without the operation of SRVs.

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

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

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

A demineralizer system bypass valve is also provided which is manually or automatically controlled from the main control room. Pressure downstream of the demineralizer or high demineralizer differential pressure is indicated and is alarmed in the main control room to alert the operator. The bypass is used only in emergency and for short periods of time until the CPS flow is returned to normal or the plant is brought to an orderly shutdown. To prevent unpolished condensate through the bypass, the bypass valve control scheme is redundant.

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

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

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

Condensate and Feedwater System

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

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

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

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

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

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

Circulating Water System

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

The Circulating Water System (CIRC) consists of the following components:

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

The cooling water is circulated by four, fixed speed, motor-driven pumps. The pumps are arranged in parallel and discharge lines combine into two parallel circulating water supply lines to the main condenser. Each circulating water supply line connects to a low pressure condenser shell inlet water box. An interconnecting line fitted with a butterfly valve is provided to connect both circulating water supply lines. The discharge of each pump is fitted with a fast actuated motor-operated or electro-hydraulically operated butterfly valve. This arrangement permits isolation and maintenance of any one pump while the others remain in operation and minimize the backward flow through a tripped pump.

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

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

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

Other Turbine Auxiliary Systems

Turbine Component Cooling Water System

The Turbine Component Cooling Water System (TCCWS) is a closed-loop cooling water system that supplies cooling water through the TCCW heat exchangers to Turbine Island equipment coolers and rejects heat to the Plant Service Water System (PSWS, see Chapter 5). It operates at a higher pressure than the PSWS, so that any intersystem leakage will not affect Turbine Building equipment. The system consists of a single loop with 2-100% capacity pumps and 2-100% capacity heat exchangers.

Station Electrical Power

Offsite Power System

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

The system includes the plant switchyard, the high voltage tie lines, the unit auxiliary transformers, the reserve auxiliary transformers the generator breaker, the isolated phase bus, and the 13.8 kV and 6.9 kV bus ducts from the unit and reserve auxiliary transformers to the unit auxiliary and PIP switchgears.

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

- “Normal Preferred” source through the unit auxiliary transformers (UAT); and
- “Alternate Preferred” source through the reserve auxiliary transformers (RAT).

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

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

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

Onsite AC Power Distribution

General

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

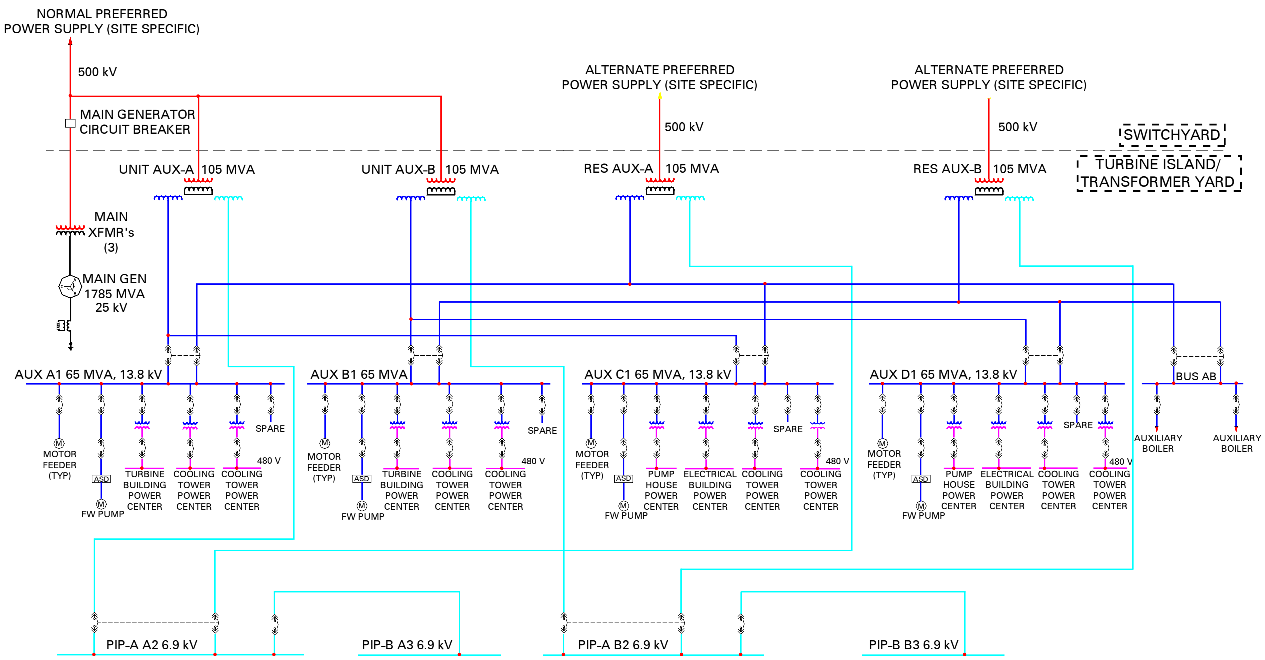


Figure 9-2. ESBWR Electrical One-Line

- **Power Generation (PG) non safety-related buses** - are those buses that are not directly backed by standby onsite AC power sources and have connections to the main or second offsite source through the unit or reserve auxiliary transformers, respectively. Backfeed from the plant investment protection (PIP) non safety-related buses is prevented by reverse power relaying. The PG non safety-related buses are the 13.8 kV unit auxiliary switchgear and associated lower voltage load buses.
- **Plant Investment Protection (PIP) non safety-related buses** - are those buses that are backed by the Standby Onsite AC Power Supply System and have connections to the normal preferred and alternate preferred offsite sources through the unit and auxiliary reserve transformers, respectively. The PIP non safety-related buses are the 6.9 kV PIP buses and associated lower voltage load buses exclusive of the safety-related buses.

The PG non safety-related buses feed non safety-related loads that are required exclusively for unit operation and are normally powered from the normal preferred power source through the unit auxiliary transformers (UATs). These buses are also capable of being powered from the alternate preferred power source through the reserve auxiliary transformers (RATs) in the event that the normal preferred power source is unavailable.

The PIP non safety-related buses feed non safety-related loads that, because of specific functions, are generally required to remain operational at all times or when the unit is shut down. In addition, the PIP non safety-related buses supply AC power to the safety-related buses. The PIP non safety-related buses are backed up by a separate standby onsite AC power supply system in each power load group. These buses are also connected to the normal preferred and alternate preferred power sources through the unit and reserve auxiliary transformers, respectively. Refer to Figure 9-3 and 9-4.

Medium Voltage AC Power Distribution System

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

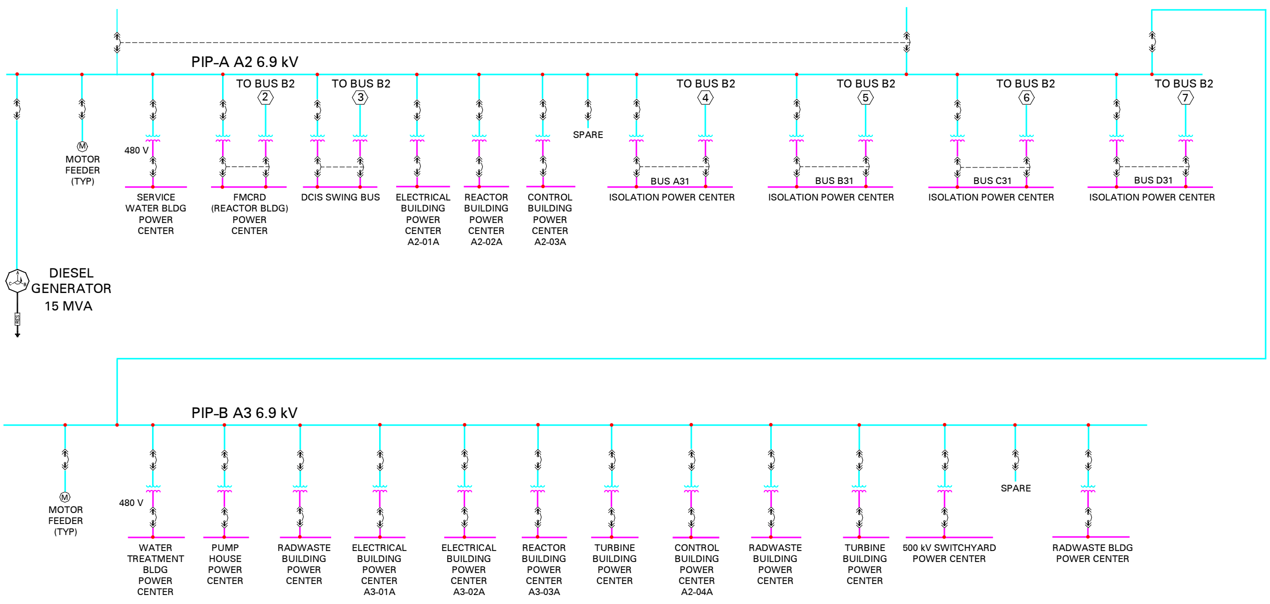


Figure 9-3. ESBWR PIP Bus - 1

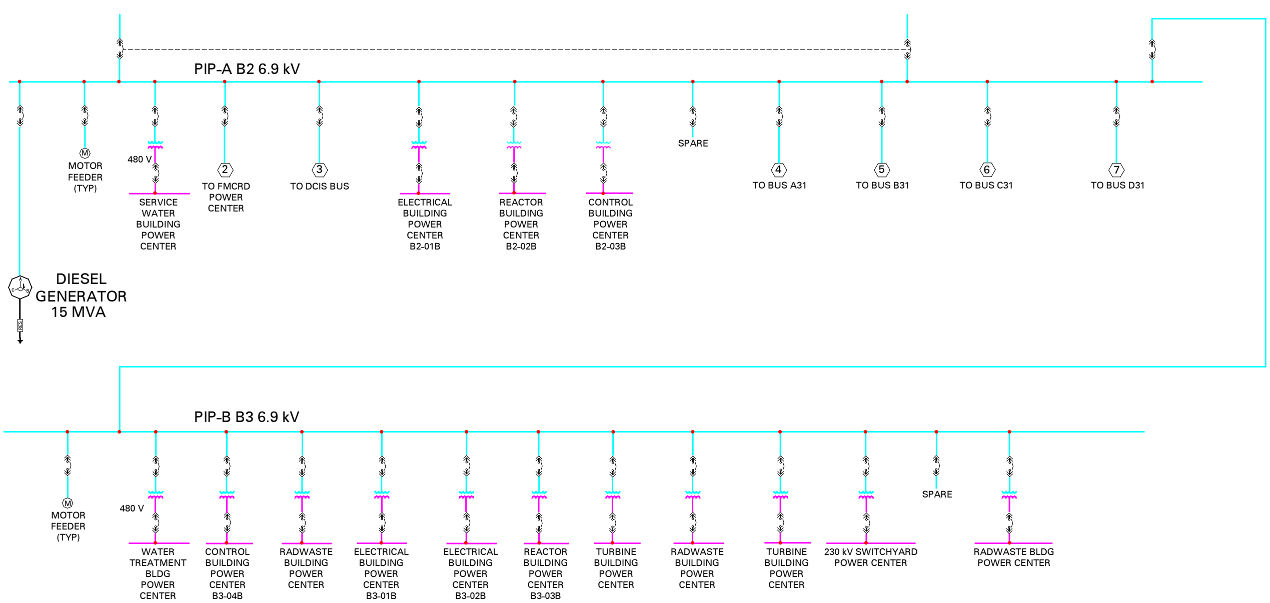


Figure 9-4. ESBWR PIP Bus - 2

proximately 250 kW and larger and power center transformers.

Four 6.9 kV PIP buses (two per load group) provide power for the non-Class 1E PIP loads. Each two of these four buses is assigned to one of the two power load groups and is backed by the standby onsite AC power supply system. Each PIP bus is normally powered from the normal preferred power source through the UAT of the same load group. Additionally, in the event of unavailability of the normal preferred power source, each PIP bus has connections to and can be powered from the alternate preferred power source through the RAT of the same load group. The source breakers of the normal and alternate preferred power sources are electrically interlocked to prevent paralleling of the two power supplies. The PIP buses distribute power at 6.9 kV to motor loads of approximately 250 kW and larger and to power center transformers.

Standby AC power for the PIP non-Class 1E buses is supplied by standby diesel generators at 6.9 kV and distributed by the non-Class 1E power distribution system. The 6.9 kV PIP buses are automatically transferred to the standby diesel generators when the normal and alternate preferred power supplies to these buses are lost. The startup time for the standby diesel generators is much less critical than in previous BWRs, due to the passive ECCS – one minute to start and 10 minutes to fully load.

Low Voltage AC Power Distribution System

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

Power is supplied from the power center transformers to the 480V power centers. The power centers supply power to motor loads of approximately 100 kW through 249 kW, and to the 480V MCCs. The power centers are of the single-fed or double-ended type depending on the redundancy requirements of the loads powered by a given power center. The power supplies to the double-ended power center transformers of the PIP non safety-related buses are supplied from different power load groups. Each double-ended power center is normally powered by its normal power source through its normal source main breaker, with the alternate source main breaker open. The power center normal and alternate source main breakers are electrically interlocked to prevent simultaneous powering of the power center by normal and alternate sources.

Isolation Power Centers

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

The isolation power centers supply power to safety-related loads of their respective division. These loads consist of the Class 1E battery chargers, Class 1E inverters and Class 1E regulating transformers. In addition, there is no Class 1E lighting that operates directly from the 480 VAC (or higher voltage) in the ESBWR design. There are no Class 1E actuators (pumps, valves, etc.), that operate directly from 480VAC (or higher) in the ESBWR design.

In addition, each isolation power center has provisions for connecting a transportable AC generator via plug-in connections, capable of supplying certain Class 1E loads while recharging the Class 1E batteries. The emergency power main circuit breaker provided in connection with these provisions is normally locked open. An interlock is provided so that only one main breaker may be closed at any time. The plug-in connections are located in a locked box. The position of the emergency main breaker and the doors to the plug-in connections are alarmed in the control room when not in the normal position. All keys are under administrative control.

Motor Control Centers

MCCs supply 99 kW and smaller motors,

control power transformers, process heaters, motor operated valves and other small electrically operated auxiliaries, including 480 to 208/120V and 480 to 240/120V transformers. MCCs are assigned to the same load group as the power center that supplies their power. Starters for the control of 480V motors smaller than 100 kW are MCC-mounted, across-the-line magnetically operated, air break type. MCC circuits feeding loads within the containment have a backup protective device in series with the primary overcurrent protective device.

Class 1E Uninterruptible AC Power Supply System

Figure 9-5 shows the overall Class 1E Uninterruptible AC Power Supply (UPS) system. The Class 1E UPS for each of the four divisions is supplied from a 480V isolation power center in the same division. The isolation power centers are connected to PIP non safety-related buses, which are backed by standby diesel generators.

Divisions I and II each have two inverters. One inverter receives 480 VAC normal power from the isolation power center of that division and has a Class 1E 72-hour battery of that division supplying 250 VDC emergency power, with an inverter output at 120 VAC single phase. The second inverter receives 480 VAC normal power from the isolation power center of that division and has a Class 1E 24-hour battery of that division supplying 250 VDC emergency power, with an inverter output at 480 VAC three phase.

Divisions III and IV each have one inverter which receives 480 VAC normal power from the Isolation Power Center of that division and has a Class 1E 24-hour battery of that division supplying 250 VDC emergency power, with an inverter output at 480 VAC three phase.

Power is distributed to the individual loads from distribution transformers and associated distribution panels, and to logic level circuits through the control room logic panels.

Non-Class 1E Uninterruptible Power Supply System

Figure 9-6 shows the overall non-Class 1E UPS. The non-Class 1E UPS for each of the two plant power distribution load groups is supplied from a 480V power center in the same load group, with standby onsite AC power of the same load group providing backup power should a failure of the normal supply occur. Emergency power of the same load group from 250 VDC batteries is provided should loss of normal and standby onsite AC power sources occur.

A third non-Class 1E UPS load group is provided to supply the non-Class 1E DCIS loads. This load group's non-Class 1E UPS is normally powered from a 480 VAC double-ended power center, which can receive power from either of the two power load groups. The power center normal and alternate source main breakers are electrically interlocked to prevent the normal and alternate sources

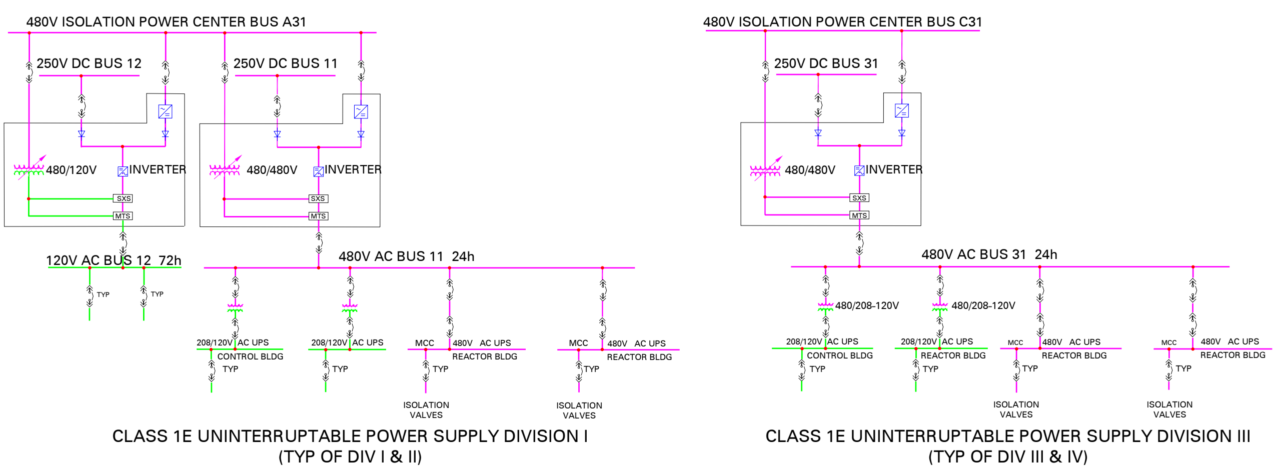


Figure 9-5. ESBWR Class 1E Uninterruptible AC Power

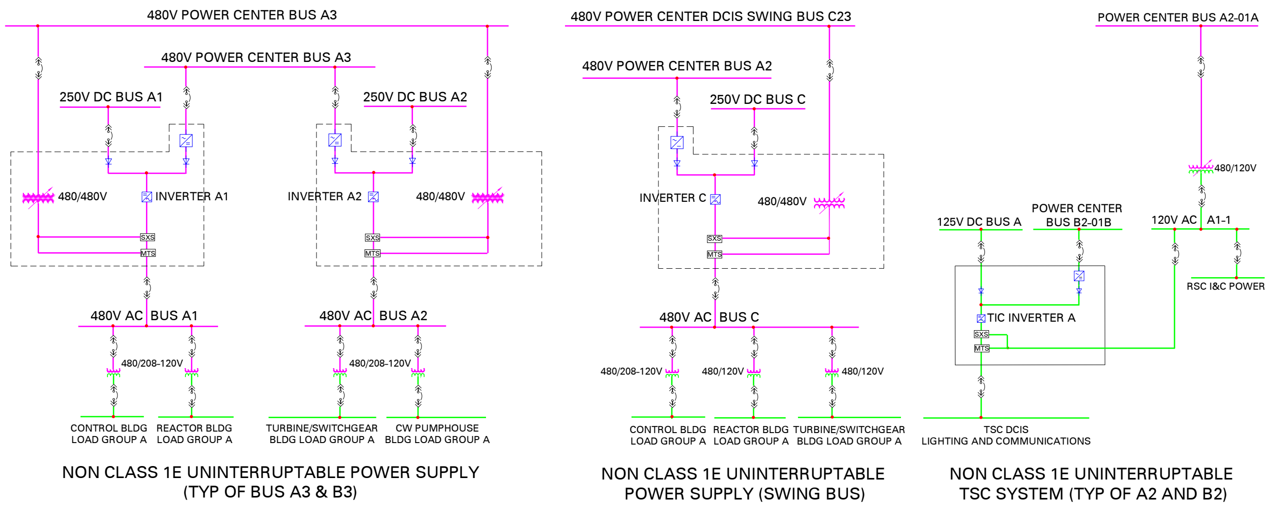


Figure 9-6. ESBWR Non Class 1E Uninterruptable AC Power

from simultaneously providing power to the power center. Additionally, standby onsite AC power from either of the two load groups provides backup power should a failure of the normal and alternate supplies occur. Emergency power of the same load group from 250 VDC batteries is provided should loss of normal, alternate, and standby onsite AC power sources occur.

Two dedicated non-Class 1E UPS are provided for the Technical Support Center (TSC), also in a two-load group configuration. Power for each TSC non-Class 1E UPS is normally supplied from a 480 VAC power center in the same load group, with standby onsite AC power of the same load group providing backup power should a failure of the normal supply occur. Emergency power of the same load group from 125 VDC batteries is provided should loss of normal and standby onsite AC power sources occur.

The non-Class 1E UPS provides reliable, uninterruptible AC power for important non safety-related equipment required for continuity of power plant operation. Each non-Class 1E UPS load group includes a solid-state inverter, solid-state transfer switch, manual transfer switch, and distribution transformers with associated distribution panels.

Instrumentation and Control Power Supply System

Regulating step-down transformers provide 208/120VAC power to I&C loads not requiring uninterruptible power. The I&C buses are each supplied independently from separate 480VAC power centers.

Four of these I&C buses power Class 1E I&C loads, and they are supplied independently from isolation power centers A31, B31, C31 and D31. These provide an alternative power source to certain Class 1E Distributed Control and Information System (E-DCIS) equipment.

Other instrumentation and control buses are supplied from the DCIS Swing Bus power center to supply non-Class 1E I&C loads. This system supplies AC loads of the Non-Essential Distributed Control and Information System (NE-DCIS), solenoid valves and other I&C loads.

The non-safety Instrumentation and Control Power Supply System does not perform any safety function.

DC Power Distribution

General

Completely independent Class 1E (i.e., safety-related) and non-Class 1E (i.e., non safety-related) DC power systems are provided. The Class 1E DC system is shown in Figure 9-7. The non- Class 1E DC system is shown in Figure 9-8.

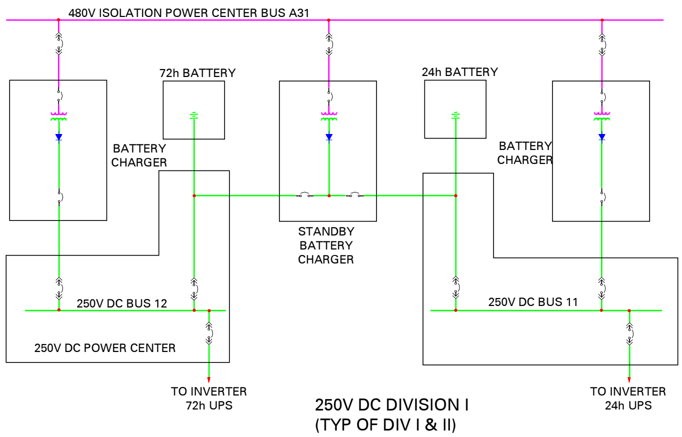


Figure 9-7. ESBWR Class 1E DC Power

Four independent Class 1E 250 VDC systems are provided, one each for Divisions 1, 2, 3 and 4. They supply normal and emergency DC power for station emergency auxiliaries and for control and switching during all modes of operation.

Five independent non-Class 1E DC systems are provided consisting of three 250VDC systems and two 125 VDC system. The non-Class 1E DC systems supply power for control and switching, switchgear control, TSC, instrumentation, and station auxiliaries.

Class 1E Station Batteries and Battery Chargers
250V Class 1E DC Systems Configuration

Figure 9-7 shows the overall 250 VDC system provided for Class 1E Divisions 1, 2, 3 and 4. Divisions 1 and 2 consists of two separate battery sets for each division. One set supplies power to selected safety loads for at least 72 hours following a licensing basis event, and the other set supplies power to other loads for a period of at least 24 hours without load shedding. Divisions 3 and 4 each have one battery set which can supply loads for at least 24 hours without load shedding. The DC systems are operated ungrounded for increased reliability. Each of the Class 1E battery systems has a 250 VDC battery, a battery charger, a main distribution panel and a ground detection panel. One divisional battery charger is used to supply each group DC distribution panel bus and its associated battery. The divisional battery charger is normally fed from its divisional 480V Isolation Power Center. The main DC distribution bus feeds the local DC distribution panels, UPS inverter, and DC motor control center. Each division has a standby charger to equalize the battery charging of that division.

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

The plant design and circuit layout of the DC systems provide physical separation of the equipment, cabling, and instrumentation essential to plant safety. Each 250 VDC battery is separately housed in a ventilated room apart from its charger, distribution, and ground detection panels. Equipment of each division of the DC distribution system is located in an area separated physically from the other divisions. All the components of Class 1E 250 VDC systems are housed in Seismic Category I structures.

Class 1E Batteries

In divisions 1 and 2, there are two separate 250 volt Class 1E batteries rated for 72-hour and 24- hour station blackout conditions, respectively. In divisions 3 and 4, there are only 24-hour batteries. The DC system minimum battery terminal voltage at the end of the discharge period is 210 volts.

The Class 1E batteries have sufficient stored capacity without their chargers to independently supply the safety-related loads continuously for the time

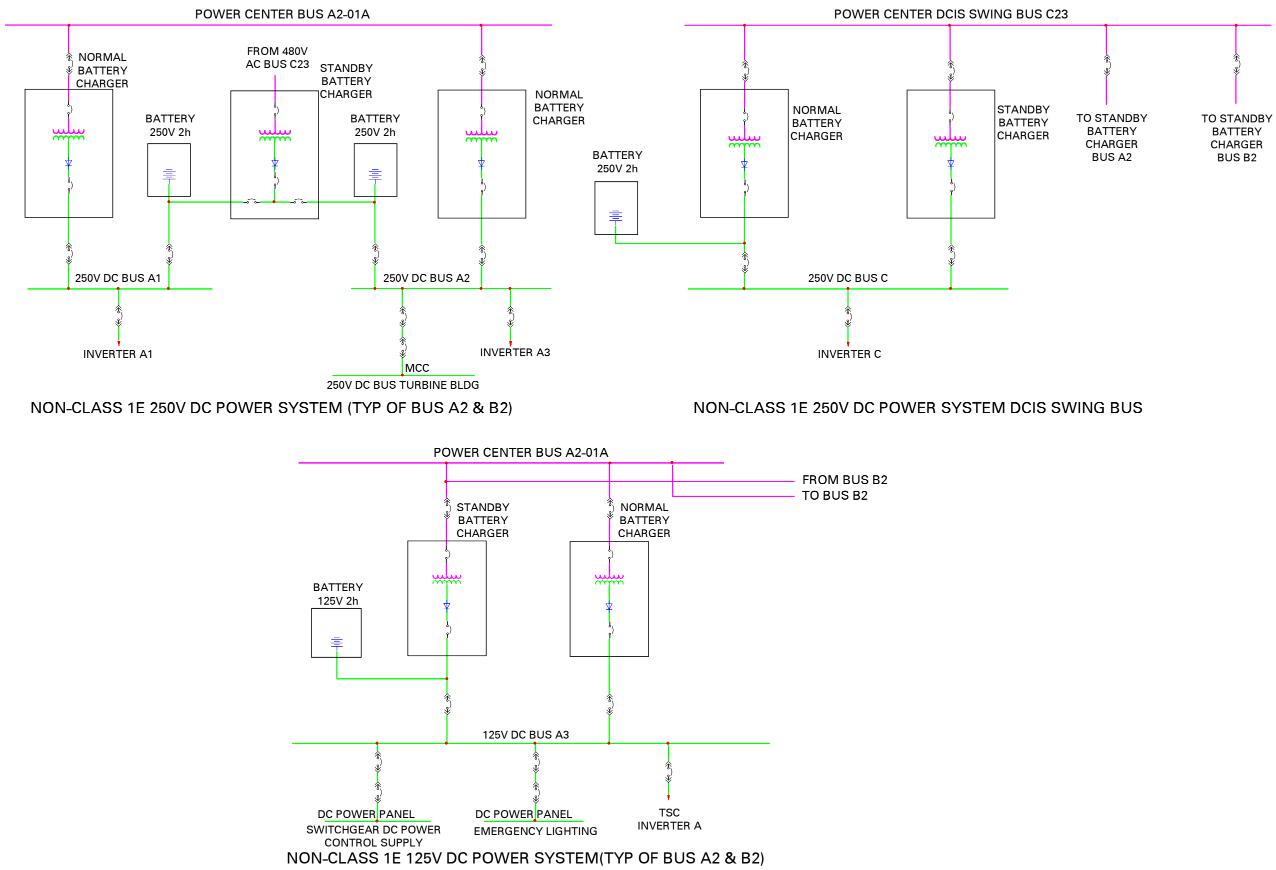


Figure 9-8. ESBWR Non-Class 1E DC Power

periods stated above. The battery banks are designed to permit the replacement of individual cells.

Class 1E Battery Chargers

The Class 1E battery chargers are full wave, silicon-controlled rectifiers. The chargers are suitable for continuously float charging the batteries. The chargers operate from a 460 volt, 3 phase, 60 Hz supply. The power for each divisional battery charger is supplied by that division's dedicated Isolation Power Center. While the standby battery charger is used to equalize its associated battery off-line, the normal charger associated with that battery is utilized to provide power to its associated DC bus.

Standby chargers are supplied from the same Isolation Power Center as the normal charger. Each battery charger is capable of recharging its battery from the design minimum charge to 95% of fully charged condition within 12 hours.

The battery chargers are the constant voltage type, adjustable between 240 and 290 volts, with the capability of operating as battery eliminators. The battery eliminator feature is incorporated as a precautionary measure to protect against the effects of inadvertent disconnection of the battery.

The battery charger's output is of a current limiting design. The battery chargers are designed to prevent their AC source from becoming a load on the batteries because of power feedback from loss of AC power.

Non-Class 1E Station Batteries and Battery Chargers 125V and 250V Non-Class 1E DC Systems Configuration

Figure 9-8 shows the overall 125V and 250V non-Class 1E DC systems. The DC systems are operated ungrounded for increased reliability. Each of the DC systems has battery, a battery charger, main

DC distribution panel, and ground detection panel. The main DC distribution buses feed the local DC distribution panels, UPS inverter and/or DC motor control center.

The plant design and circuit layout of the non-Class 1E DC systems provide physical separation of the equipment, cabling and instrumentation associated with the load groups of non-Class 1E equipment. Each 125V and 250 VDC battery is separately housed in a ventilated room apart from its charger, distribution, and ground detection panels. Equipment of each load group of the DC distribution system is located in an area separated physically from the other load groups.

The non-Class 1E DC power is required for standby lighting, control and switching functions such as the control of 6.9 kV and 480V switchgear, DC motors, control relays, meters and indicators.

Non-Class 1E Batteries

The 125 volt non-Class 1E batteries are sized for 2-hour duty cycles at a discharge rate of 2 hours. The DC system minimum battery terminal voltage at the end of the discharge period is 105 volts. The maximum equalizing charge voltage for 125V batteries is 140 VDC.

The 250 volt non-Class 1E batteries are sized for 2-hour duty cycles at a discharge rate of 2 hours. The DC system minimum battery terminal voltage at the end of the discharge period is 210 volts. The maximum equalizing charge voltage for 250V batteries is 280 VDC.

The non-Class 1E batteries have sufficient stored capacity without their chargers to independently supply their loads continuously for at least 2 hours. Each distribution circuit is capable of transmitting sufficient energy to start and operate all required loads in that circuit. The battery banks are designed to permit replacement of individual cells without loss of availability or capability.

Non-Class 1E Battery Chargers

The non-Class 1E battery chargers are full wave, silicon-controlled rectifiers. The chargers are suitable for float charging the batteries. The chargers operate from a 460 volt, 3 phase, 60 Hz supply. Each train charger is supplied from a separate power center, which is backed by the standby diesel generator.

Standby chargers are used to equalize battery charging. Standby chargers are supplied from a different power center than the main charger.

Each battery charger is capable of recharging its battery from the design minimum charge to 95% of fully charged condition within 12 hours.

The battery chargers are the constant voltage type, with the 125 VDC system chargers having a voltage adjustable between 120 and 145 volts and the 250 VDC system chargers having a voltage adjustable between 240 and 290 VDC, with the capability of operating as battery eliminators. The battery eliminator feature is incorporated as a precautionary measure to protect against the effects of inadvertent disconnection of the battery. The battery charger's output is of a current limiting design. The battery chargers are designed to prevent their AC source from becoming a load on the batteries caused by power feedback from a loss of AC power.

Chapter 10
Radioactive Waste Systems

Overview

The radwaste facility has been significantly improved compared to past designs. The use of mobile reprocessing technologies for both liquid and solid radwaste processing improves the efficiency of the process and allows new mobile technologies to be readily adapted to the existing ESBWR radwaste system. The radwaste building is designed to be very flexible. The only permanently installed equipment are the collection and sample tanks and the support systems required by the mobile reprocessing skids. The liquid radwaste system is designed for 100% recycle with no offsite release.

The impact of these improvements in the ESBWR design gives assurance that dewatered or powdered solid waste volume is less than 70 m^3/year and dry solid waste volume is less than 370 m^3/yr, reducing the radwaste volume significantly compared to current U.S. operating plants. Annual releases to the environment from the ESBWR 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.

The Radwaste systems include the Liquid Waste Management System (LWMS) System, the Offgas System (OGS) System, and the Solid Waste Management System (SWMS).

Liquid Radwaste Management System

The ESBWR Liquid Waste Management System (LWMS) is designed to control, collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences. All potentially radioactive liquid wastes are collected in sumps or drain tanks at various locations in the plant and transferred to collection tanks in the radwaste facility.

System components are designed and arranged in shielded enclosures to minimize exposure to plant personnel during operation, inspection, and maintenance. Tanks, processing equipment, pumps, valves, and instruments that may contain radioactivity are located in controlled access areas.

The LWMS normally operates on a batch basis. Provisions for sampling at important process points are included. Protection against accidental discharge is provided by detection and alarm of abnormal conditions and by administrative controls.

The LWMS is divided into several subsystems, so that the liquid wastes from various sources can be segregated and processed separately, based on the most economical and efficient process for each specific type of impurity and chemical content. Cross-connections between subsystems provide additional flexibility in processing the wastes by alternate methods and provide redundancy if one subsystem is inoperative.

The LWMS is housed in the radwaste building and consists of the following four subsystems:

- Equipment (low conductivity) drain subsystem;
- Floor (high conductivity) drain subsystem;
- Chemical drain subsystem;

- Detergent drain subsystem

These process subsystems ensure that liquid waste from various sources can be segregated and treated separately, based on the most economical and efficient process for each specific type of impurity and chemical content. LWMS has been designed to recycle 100% of the ESBWR's liquid waste.

Equipment (Low Conductivity) Drain Subsystem

The subsystem block flow diagram is shown in Figure 10-1. The equipment drain collection tanks receive low conductivity inputs from various sources within the plant. These waste inputs have a high chemical purity and are processed on a batch basis. The equipment drain subsystem consists of three collection tanks and collection pumps, a mobile based Hollow Fiber Filter (HFF) and Deep-Bed Ion Exchanger system including organic material pre-treatment equipment, an intermediate tank/pump, and two sample tanks and sample pumps. One collection tank is normally used as a surge tank that can collect waste from the low conductivity waste and/or high conductivity waste. Cross-connections with the floor drain subsystem allows processing through the mobile system for floor drain treatment.

Mobile based chemical addition equipment is provided to add chemical agent(s) for recovering the performance of the HFF System. The equipment provides for addition of chemical agent(s) for the Reverse Osmosis System (RO) in the floor drain subsystem.

A strainer or filter is provided downstream of the last ion exchanger in series to collect any crud and resin fines that may be present.

The process effluents are collected in one of the two sample tanks for chemical and radioactivity analysis. If acceptable, the tank contents are returned to the condensate storage tank for plant reuse. A recycle line from the sample tanks allows the sampled effluents that do not meet water quality requirements to be pumped back to an Equipment (Low Conductivity) Drain Collection Tank or Floor

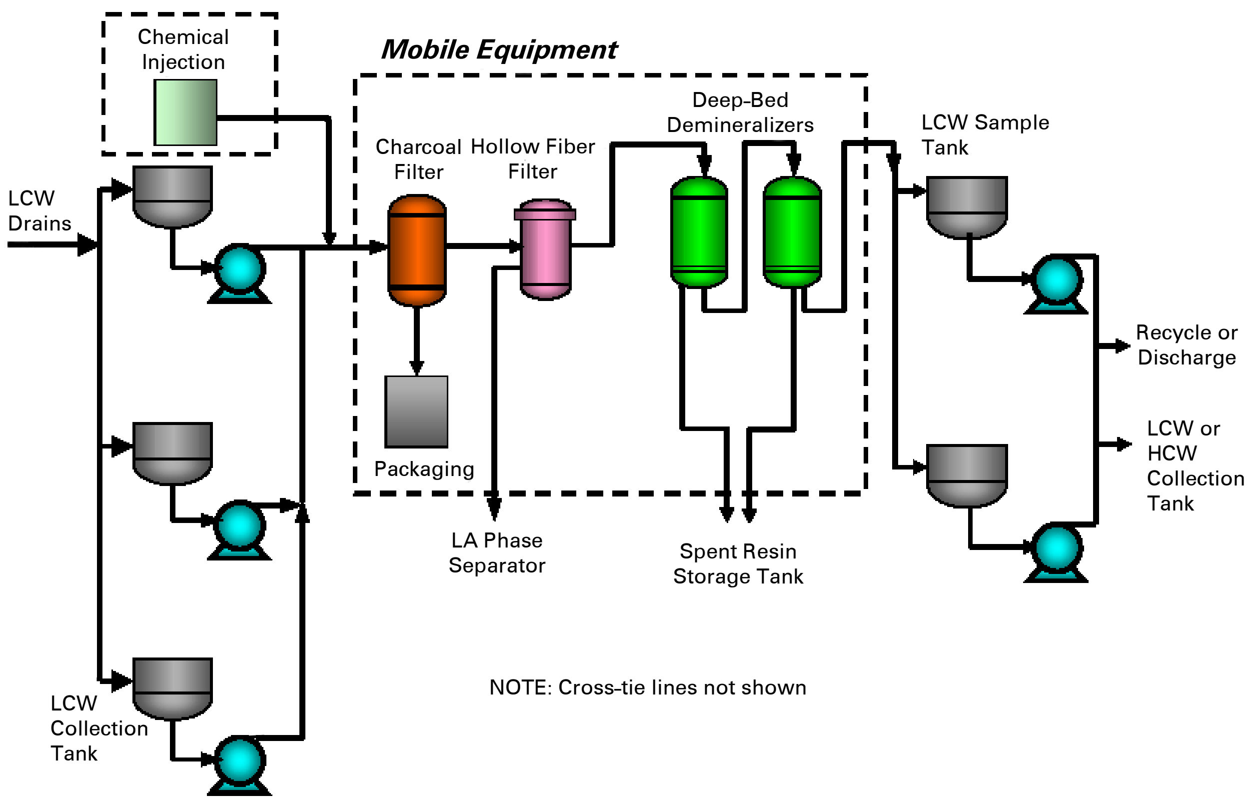


Figure 10-1. Equipment Drain Subsystem Schematic

(High Conductivity) Drain Collection Tank for additional processing. If the plant condensate inventory is high, the sampled process effluent may be discharged.

The HFF is backwashed periodically to maintain filtration capacity. Backwash waste is discharged to a low activity phase separator. Spent deep-bed ion exchanger resin is discharged to a low activity spent resin holdup tank as a slurry.

Floor (High Conductivity) Drain Subsystem

The subsystem block flow diagram is shown in Figure 10-2. The floor drain collection tanks receive high conductivity waste inputs from various floor drain sumps in the reactor building, turbine building, and radwaste building. The floor drain collection tanks also receive waste input from chemical drain collection tank.

The floor drain subsystem consists of two floor drain collection tanks and collection pumps, a mobile based Reverse Osmosis (RO) and Deep-Bed Ion Exchanger System including suspended solid pre-treatment equipment, an intermediate tank/pump and two sample tanks and sample pumps. The waste collected in the floor drain collection tanks are processed on a batch basis. Cross-connections with the equipment drain subsystem also allow for processing through that subsystem. Additional collection capacity is also provided by one additional equipment drain collection tank that is shared with the equipment drain subsystem.

A strainer or filter is provided downstream of the last ion exchanger in series to collect any crud and resin fines that may be present.

The floor drain sample tanks collect the process effluent, so that a sample may be taken for chemical and radioactivity analysis before discharging or recycling. The discharge path depends on the water quality, dilution stream availability and plant water inventory. Off-standard quality effluent can be recycled to floor drain collection tanks or equipment drain collection tanks. If the treatment effluent meets

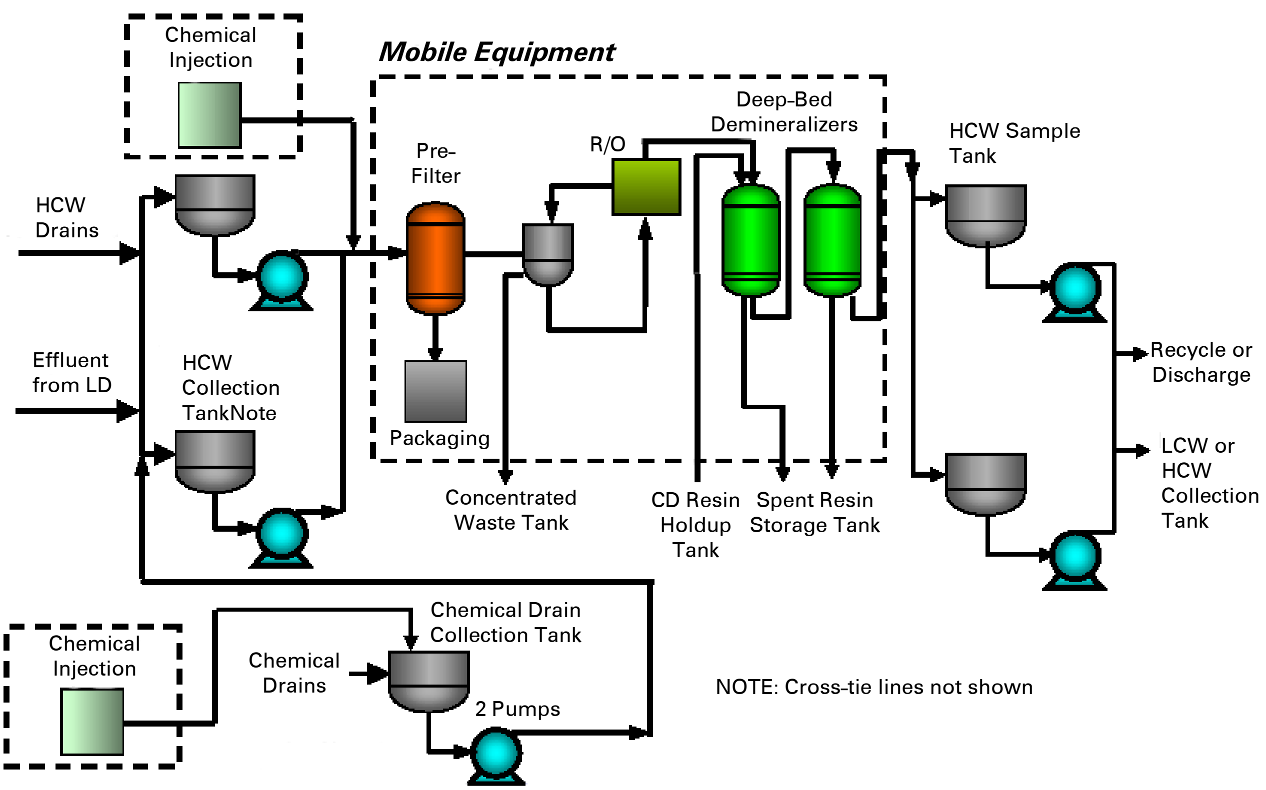


Figure 10-2. Floor Drain Subsystem Schematic

water quality standards and if the water inventory permits it to be recycled, the processed floor drain effluent can be recycled to the condensate storage tank or discharged offsite.

The liquid waste is concentrated in the RO system and is periodically discharged to a concentrated waste tank. Spent deep-bed ion exchanger resin is discharged to a low activity spent resin holdup tank as a slurry.

The capability exists to accept used condensate polishing resin in a Condensate Resin Holdup Tank. The used condensate polishing resin from Condensate Purification System is transferred to the Condensate Resin Holdup Tank prior to use in the pre-treatment deep-bed ion exchanger in the floor drain subsystem.

Chemical Drain Subsystem

The subsystem block flow diagram is shown in Figure 10-2. The chemical waste collected in the chemical drain collection tank consists of laboratory wastes and decontamination solutions. After accumulating in the chemical drain collection tank, chemical agents may be added to the chemical drain by mobile-based chemical pre-treatment equipment if necessary and the pre-treated chemical drain is transferred to floor drain collection tanks for further processing. Chemical pre-treatment operation is typically a neutralization process. A sample is then taken and if discharge standards are met, then the waste may be discharged offsite. A cross-connection with the detergent drain subsystem is also provided.

Detergent Drain Subsystem

The subsystem block flow diagram is shown in Figure 10-3. Waste water containing detergent from the controlled laundry and personnel decontamination facilities and decontamination waste water from the reactor building or turbine building throughout the plant is collected in the detergent drain collection tanks. The detergent drain subsystem consists of two detergent drain collection tanks and collection pumps, a mobile-based detergent drain filter and charcoal filter system including organic material pretreatment equipment, an intermediate tank/pump, and two sample tanks and sample pumps. The detergent wastes are processed through a suspended solid removal process and organic material removal process and collected in sample tanks. A sample is then taken and if discharge standards are met, then the waste is discharged offsite. Off-standard quality water can either be recycled for further processing to the detergent collection tank or to the floor drain collection tank. A cross-connection with the chemical drain collection subsystem is also provided.

Mobile Subsystems

For Equipment Drain Processing—The equipment drain mobile system utilizes non-precoat type

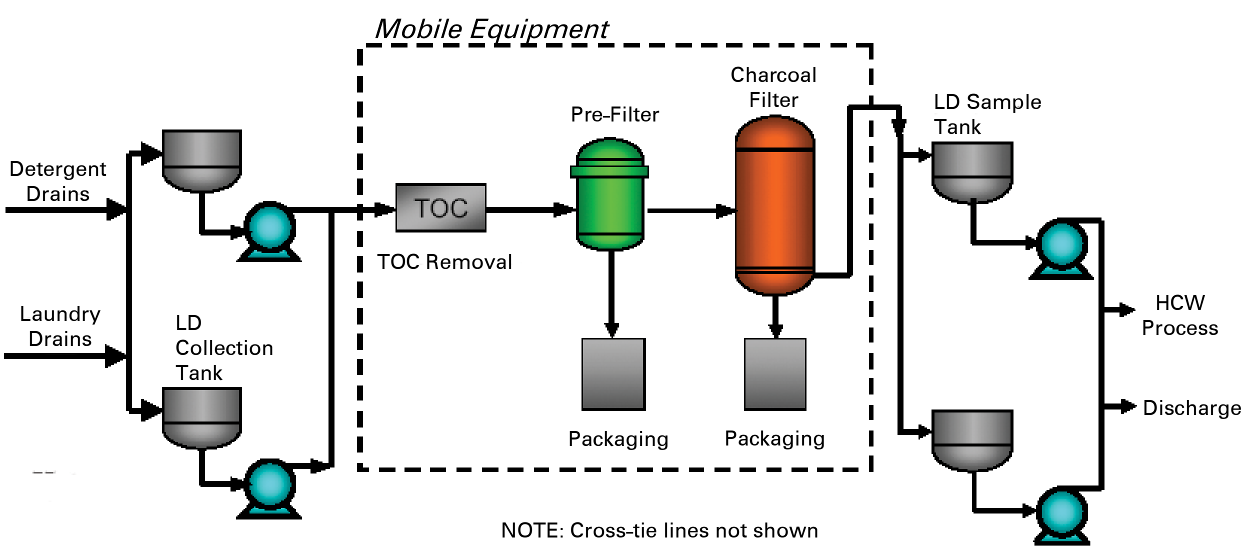


Figure 10-3. Detergent Drain Subsystem Schematic

Hollow Fiber Filter (HFF) for removing suspended solid and radioactive particulate material and charcoal filters for organic material removal. Backwash operation for HFF is performed when the differential pressure of HFF exceeds a preset limit. HFF backwash waste is discharged to a low activity phase separator. A charcoal filter is located upstream of HFF for the purpose of removing organic material, which may cause fouling of the HFF. Spent charcoal is packaged directly into the container when the differential pressure exceeds a preset limit or waste quality of the effluent from the charcoal filter exceeds a preset value. The equipment drain ion exchangers following the HFF are of the mixed-bed type. Exhausted resins are sluiced to the low activity spent resin holdup tank when some chosen effluent purity parameter (such as conductivity) exceeds a preset limit or upon high differential pressure. Fine mesh strainers with backwashing connections are provided in the ion exchange vessel discharge and in the downstream piping to prevent resin fines from being carried over to the sampling tanks. The mobile system is skid-mounted and is designed and configured for relatively easy installation and process reconfiguration. In-plant supply and return connections from permanently installed equipment to the mobile system are provided to keep operational flexibility.

For Floor and Chemical Drain Processing—Floor drain and chemical drain wastes are more complex to process than equipment drains. The floor drain mobile system utilizes pre-filtration equipment for removing suspended solids and organic impurities, Reverse Osmosis System (RO) for removing ionic impurities, and finally deep-bed ion exchangers for polishing. The pre-filtrated liquid waste is collected into the feed tank of the RO System. The feed tank serves as a front-end supply tank to the process. The liquid waste is transferred to the RO unit via a booster pump. The RO unit uses membrane tubes that are made of a semi-permeable material. When pressure is applied to the feed side of the membrane, the solution passes through the membrane (permeates) and the solids and high molecular wastes are rejected. The rejected solids and ionic impurities are returned to the feed tank and the final permeate is polished by deep-bed ion exchangers. The floor drain ion exchangers following the RO are of the mixed-bed type. Exhausted resins are sluiced to the spent resin tank when some chosen effluent purity parameter (such as conductivity) exceeds a preset limit or upon high differential pressure. Fine mesh strainers with backwashing connections are provided in the ion exchange vessel discharge and in the downstream piping to prevent resin fines from being carried over to the sampling tanks. The chemical drain pre-treatment unit performs a pre-conditioning of chemical waste, such as pH adjustment, prior to processing in the RO system. The mobile system is of a skid-mounted design and configured for relatively easy installation and process reconfiguration. In-plant supply and return connections from permanently installed equipment to the mobile system are provided to ensure operational flexibility.

For Detergent Drain Processing—The detergent drain mobile system typically utilizes a charcoal filter to remove organics and a cartridge type filter to remove suspended solids. When the deferential pressure of the filter exceeds a preset value, the filter media is exchanged and the spent filter media is packaged as active solid waste. The mobile system is of a skid-mounted design and configured for relatively easy installation and process reconfiguration. In-plant supply and return connections from permanently installed equipment to the mobile system are provided to ensure operational flexibility. The mobile systems are located in the Liquid Waste Treatment System bay to allow truck access and mobile system skid loading and unloading. Modular shield walls are provided in the Radwaste Building to allow shield walls to be constructed to minimize exposure to personnel during operation and routine maintenance.

Offgas System

The Gaseous Waste Management or Offgas System (OGS) processes and controls the release of gaseous radioactive effluents to the site environs so as to maintain the exposure of persons outside the controlled area and personnel working near the system components to as low as reasonably achievable. The OGS process flow diagram is shown in Figure 10-4.

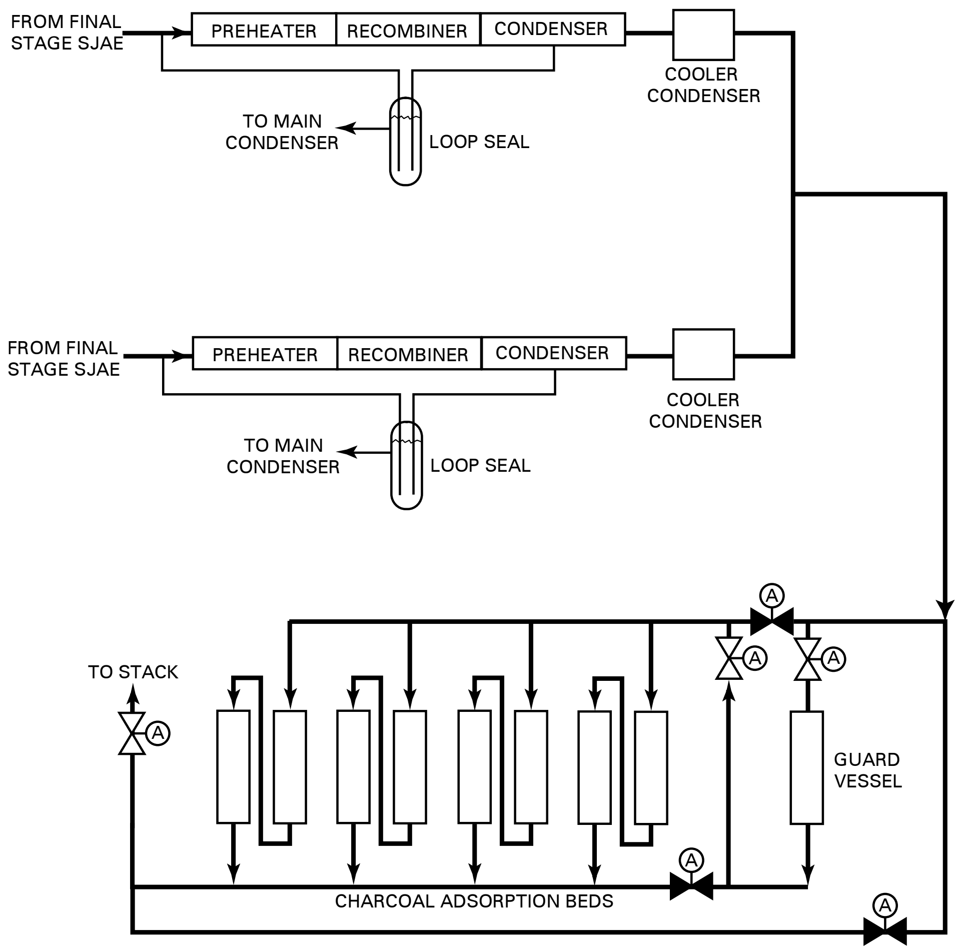


Figure 10-4. Offgas System Schematic

The OGS is an all-welded leak-tight system with redundant active components. The OGS process equipment is housed in a reinforced-concrete structure to provide adequate shielding. Charcoal adsorbers are installed in a temperature monitored and controlled vault. The facility is located in the turbine building to minimize piping.

The main condenser evacuation system removes the noncondensable gases from the main condenser and discharges them to the OGS. The evacuation system consists of two 100% capacity, multiple-element, two stage steam jet air ejectors (SJAEs) with intercondensers, for normal station operation, and mechanical vacuum pumps for use during startup. The OGS receives air and noncondensable gases from the SJAEs and processes the effluent for the decay and/or removal of gaseous and particulate radioactive isotopes.

The OGS also reduces the possibility of an explosion from the buildup of radiolytic hydrogen and oxygen. This is accomplished by the recombination of the radiolytic hydrogen and oxygen under controlled conditions within a catalytic recombiner. This process strips the condensables and reduces the volume of gases being processed. Recombiner preheaters preheat gases to provide for efficient catalytic recombiner operation and to ensure the absence of liquid water that suppresses the activity of the recombiner catalyst.

Each recombiner is part of an integrated preheater-recombiner-condenser pressure vessel assembly. The preheater section uses nuclear quality steam to heat the offgas process stream gases before they reach the catalyst in the recombiner section. The recombined hydrogen and oxygen, in the form of superheated steam, which leaves the recombiner section is then condensed (by power cycle condensate) to liquid water in the condenser section of the assembly, while the noncondensable gases are cooled. The condensed water in the condenser section is drained to a loop seal that is connected to the main condenser hotwell. Condensed preheater section steam is also drained to the above loop seal that is connected to the hotwell. No flow paths exist whereby unrecombined offgas can bypass the recombiners.

The gaseous waste stream is then further cooled by chilled water in the cooler condenser. The cooler condenser is designed to remove any condensed moisture by draining it to the offgas condenser.

The remaining noncondensables (principally air with traces of krypton and xenon) are passed through activated charcoal beds which are operated at an ambient temperature and provide a holdup volume to allow time for the krypton and xenon to decay. In order to insure enough noncondensable flow, a small quantity of air is deliberately introduced into the system. After processing, the gaseous effluent is monitored and released to the environs through the plant stack.

The OGS process equipment is housed in a reinforced-concrete structure to provide adequate shielding. Charcoal adsorbers are installed in a temperature monitored and controlled vault. The facility is located in the turbine building to minimize piping.

Solid Radwaste Management System

The Solid Waste Management System (SWMS) is designed to control, collect, handle, process, package, and temporarily store wet and dry solid radioactive waste prior to shipment. This waste is generated as a result of normal operation and anticipated operational occurrences. The SWMS is located in the radwaste building. It consists of the following four subsystems:

- Wet solid waste collection subsystem;
- Mobile wet solid waste processing subsystem;
- Dry solid waste accumulation and conditioning subsystem;
- Container storage subsystem

Wet Solid Waste Collection Subsystem

The wet solid waste collection subsystem collects spent bead resin slurry, filter and tank sludge slurry and concentrated waste into the one of the five tanks in accordance with the waste characteristics (see Figure 10-5).

Spent bead resin sluiced from the RWCU, FAPCS, Condensate Purification System and LWMS are transferred to three spent resin tanks for radioactive decay and storage. Spent resin tanks are categorized as follows:

- High Activity Resin Holdup Tank for receiving RWCU and FAPCS spent bead resin,
- Low Activity Resin Holdup Tank for receiving LWMS spent bead resin,
- Condensate Resin Holdup Tank for receiving Condensate Purification System spent bead resin.

The capability exists to keep the higher activity resins, the lower activity resins and condensate resins in separate tanks. Excess water from a holdup tank is sent to the equipment drain collection tank or floor drain collection tank by a decant pump.

When sufficient bead resins have been collected in the high/low activity resin holdup tank, they are mixed via the high/low activity resin circulation pump and sent to the mobile wet solid waste processing subsystem via the high/low activity resin transfer pump. When sufficient bead resins have been collected in the condensate resin holdup tank, they are mixed via the condensate resin circulation pump and sent to the LWMS pre-treatment ion-exchanger for reuse or the mobile wet solid waste processing subsystem via the condensate resin transfer pump.

Two Low Activity Phase Separators receive suspended solid slurries from the Condensate Purification System, mobile filtration system of the

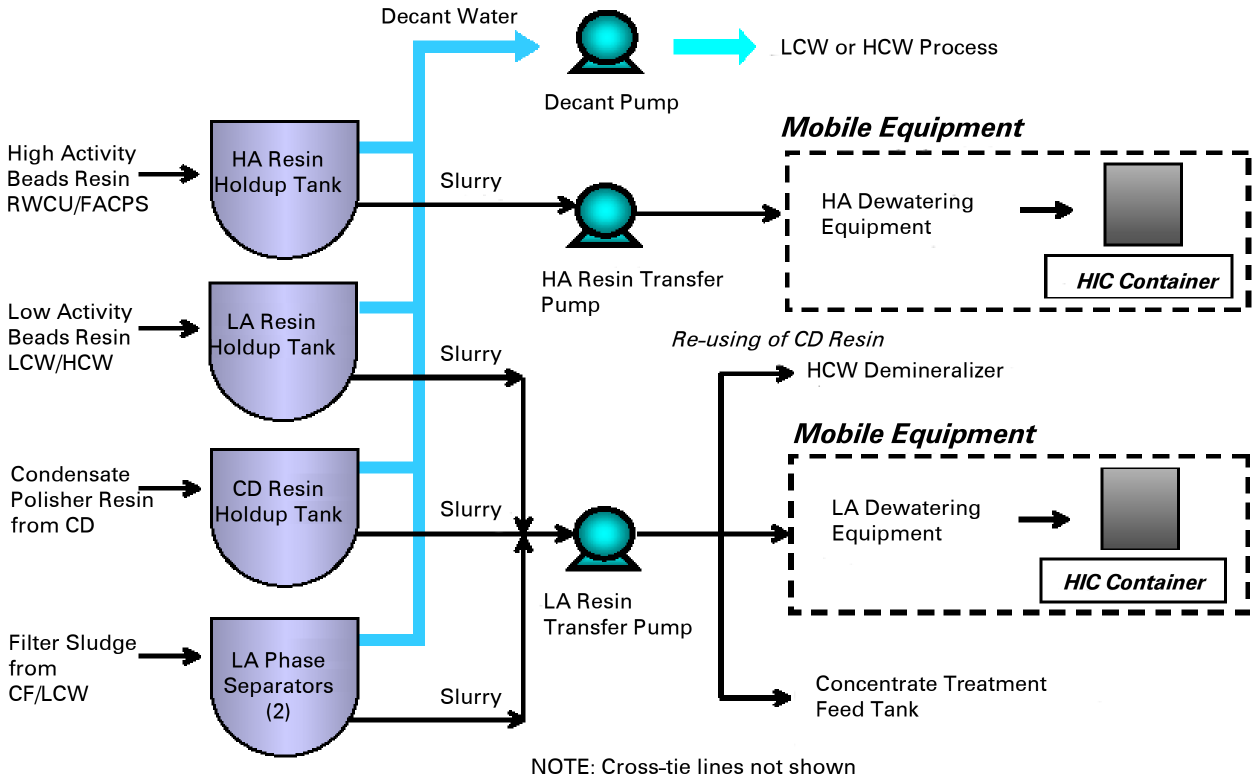


Figure 10-5. Wet Solid Waste Collection Subsystem Schematic

LWMS and high integrity containers (HIC) effluent. The suspended solids are allowed to settle and the residual water is transferred by the low activity decant pump to the equipment drain collection tanks or floor drain collection tanks for further processing. When sufficient sludges have been collected in the tank, the sludges are mixed by the low activity resin circulation pump and sent to the mobile wet solid waste processing subsystem by the low activity resin transfer pump.

During transfer operations of the spent bead resins and the sludges, the suspended solids are kept suspended by a circulation pump to prevent them from agglomerating and possibly clogging lines.

One Concentrated Waste Tank receives concentrated waste from the mobile reverse osmosis system of the LWMS. When sufficient concentrated waste has been collected in the tank, the concentrated waste is sent to the mobile wet waste processing subsystem by a mixing/transfer pump.

Mobile Wet Solid Waste Processing Subsystem

The mobile wet solid waste processing subsystem consists of a dewatering station for high activity spent resin, a dewatering station for low activity spent resin and sludge and a dewatering station (or dryer) for concentrated waste (see Figures 10-5 and 10-6). An empty HIC is lifted off of a transport trailer and placed in each empty dewatering station. The tractor/trailer may then be released. The HIC closure lid is removed and placed in a laydown area. Spent cartridge filters may be placed in the HIC at this point, if not shipped in separate containers.

Next, the fill head is positioned over the HIC using a crane. The fill head includes a closed circuit television camera for remote viewing of the fill operation. The HIC is then filled with each kind of wet solid waste. The capability to obtain samples during the fill operation is provided.

Excess water is removed from the HIC and sent by a resin pump to the high activity resin holdup tank or a low activity phase separator that is in the

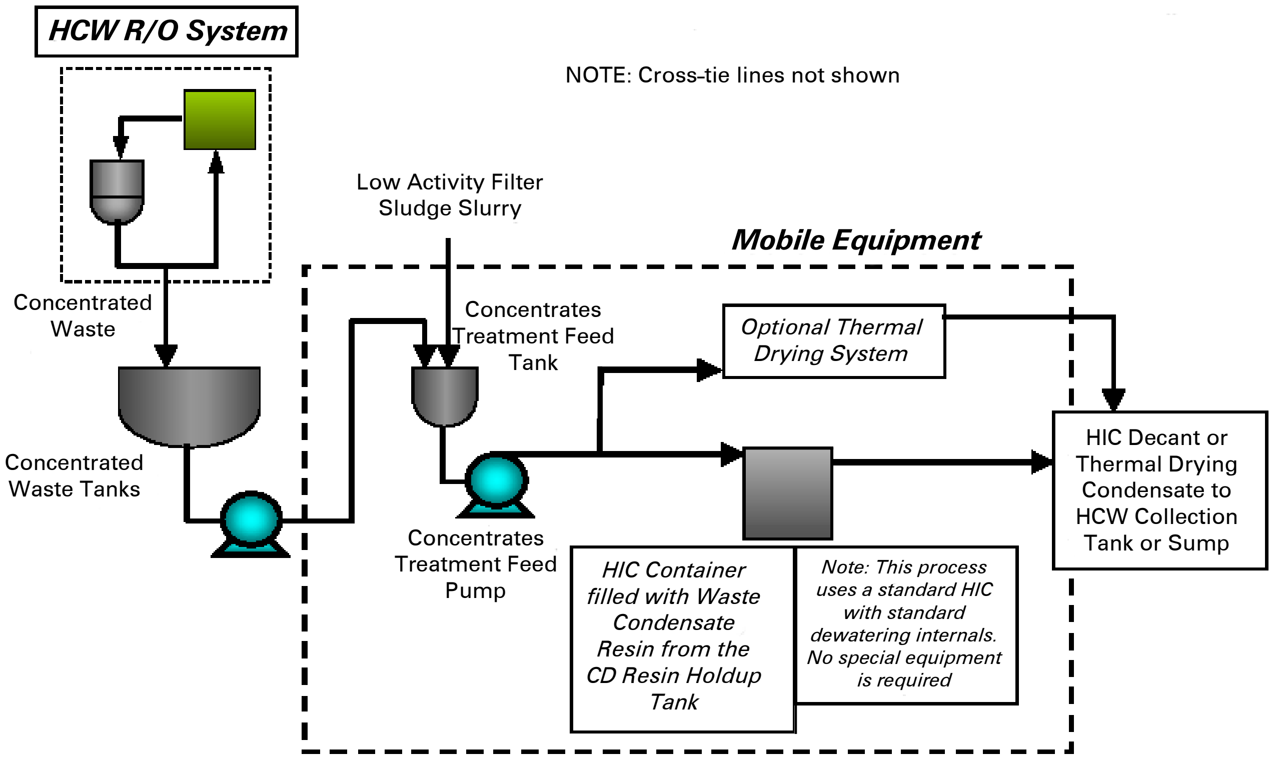


Figure 10-6. Mobile Wet Solid Waste Subsystem Schematic

receiving mode by a resin pump. Sufficient water is removed to ensure there is very little or no free standing water left in the HIC. Drying of the HIC contents may also be performed with heated air.

The fill head is then removed and placed in a laydown area. The closure head is then placed on the HIC. The HIC is vented just prior to being shipped offsite for disposal to ensure that pressure is not building up. Radiation shielding is provided around the HIC stations.

Dry Solid Waste Accumulation and Conditioning Subsystem

Dry solid wastes consist of air filters, miscellaneous paper, rags, etc., from contaminated areas; contaminated clothing, tools, and equipment parts that cannot be effectively decontaminated; and solid laboratory wastes. The activity of much of this waste is low enough to permit handling by contact. These wastes are collected in containers located in appropriate areas throughout the plant, as dictated by the volume of wastes generated during operation and maintenance. The filled containers are sealed and moved to controlled-access enclosed areas for temporary storage (see Figure 10-7).

Most dry waste is expected to be sufficiently low in activity to permit temporary storage in unshielded, cordoned-off areas. Dry active waste will be sorted and packaged in a suitably sized container that meets DOT requirements for shipment to either an offsite processor or for ultimate disposal. The dry active waste is separated into three categories: non-contaminated wastes (clean), contaminated metal wastes, and the other wastes, i.e., clothing, plastics, HEPA filters, components, etc.

In some cases, large pieces of miscellaneous waste are packed into larger boxes. Because of its low activity, this waste can be stored until enough is accumulated to permit economical transportation to an offsite burial ground for final disposal. The capability exists to bring a shipping container into the truck bay during periods of peak waste generation. Bagged dry active waste can be directly loaded into the shipping container for burial or processing in offsite facilities. A truck scale is provided to ensure optimum shipping/disposal weight of the shipping container.

Cartridge filters that are not placed in HICs are placed in suitability-sized containers meeting DOT requirements.

Container Storage Subsystem

Onsite storage space for 6-months volume of packaged waste is provided. Packaged waste includes HICs, shielded filter containers and 55-gallon (200-liter) drums as necessary.

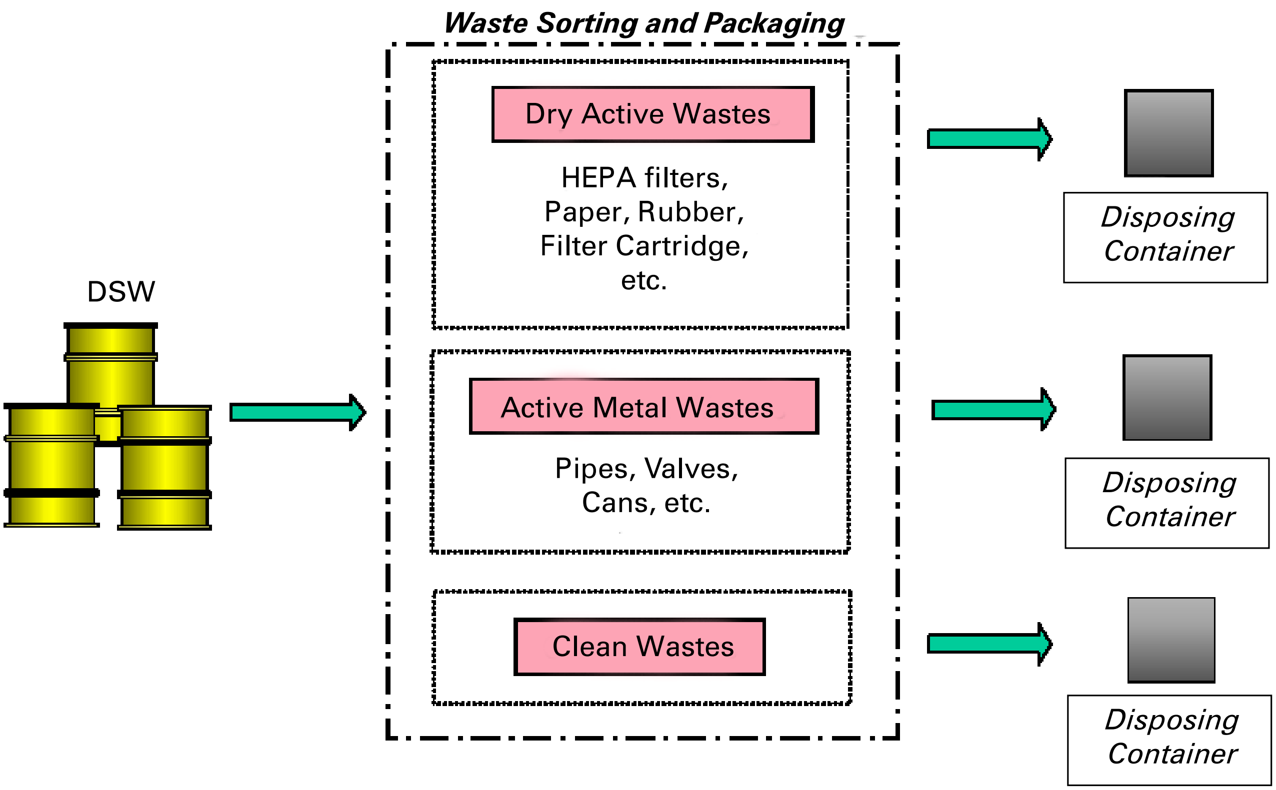


Figure 10-7. Dry Solid Waste Subsystem Schematic

Chapter 11 Safety Evaluations

Overview

The ESBWR represents a new approach to reactor safety. The use of natural circulation and passive ECCS requires a reactor vessel with a relatively large steam volume at power and a relatively large water volume when shutdown. These features permit a reactor design with a more gentle response to design basis transients and accidents. The use of passive containment cooling together with passive drywell flooding and a lower drywell core catcher lead to a containment virtually certain to survive severe accidents

Transient Performance

Transient performance, in the safety sense, becomes translated into fuel performance and operating margins. The primary BWR measures are minimum critical power ratio (MCPR), and maximum linear heat generation rate (MLHGR). These design parameters vary, depending on the specific fuel design being used (e.g., 9x9 or 10x10). However, the ESBWR was designed to assure flexibility of use of advancing fuel technologies while maintaining significant operating margins to fuel limits (20% or more up to 18 month cycles and 15% or more for 24 month cycles).

Pressurization transients historically have limited BWR transient performance. With the combination of a large steam volume in the RPV (for transient pressure rise) and the use of isolation condensers (for longer term heat removal), the pressure rise is less than SRV setpoints, so there is no relief valve lift even for isolation transients. In

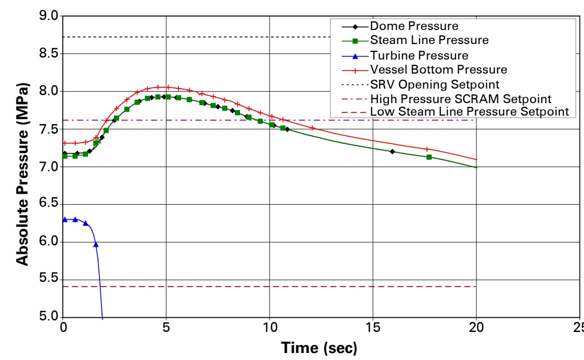


Figure 11-1. MSIV Closure Transient

addition, the transient change in MCPR (ΔMCPR) is no longer limiting. Figure 11-1 shows the ESBWR pressure response to an MSIV closure event.

All transients proceed at a slower pace than in previous BWRs. In fact, the most limiting transient for ΔMCPR is a loss of feedwater heating event which (assuming no operator action) takes place over several hundred seconds, and the use of Select Control Rod Run-In (SCRRI) establishes a lower power operating point with CPR margin restored - see Figure 11-2.

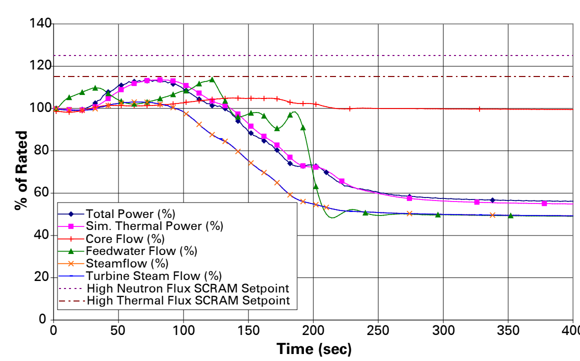


Figure 11-2. Loss of Feedwater Heating Transient

One other characteristic of a large natural circulation reactor is a larger level swing upon scram due to collapsing voids in the tall chimney. The most limiting of these is the Loss of Feedwater event, shown in Figure 11-3. Even though there is approximately an 8 m level drop upon loss of feedwater, the minimum level is still 5 m above the top of active fuel (TAF).

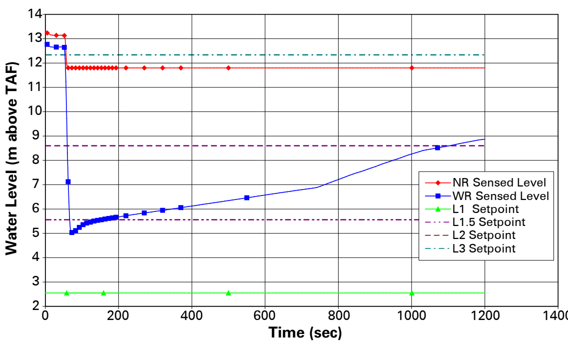


Figure 11-3. Loss of Feedwater Transient

Accident Performance

The ESBWR uses passive systems (GDCS, PCCS, IC) to mitigate loss-of-coolant accidents (LOCA). More information about these systems can be found in Chapters 3 and 4. Another design feature is provided by the raised suppression pool in the containment and sufficient in containment water to assure long term core coverage. Finally, there are no large pipes attached to the RPV below core elevation. The combination of these features assures the ESBWR has no core uncovery even for the most limiting design basis loss-of-coolant accident (DBA LOCA). Figure 11-4 shows core water level in the

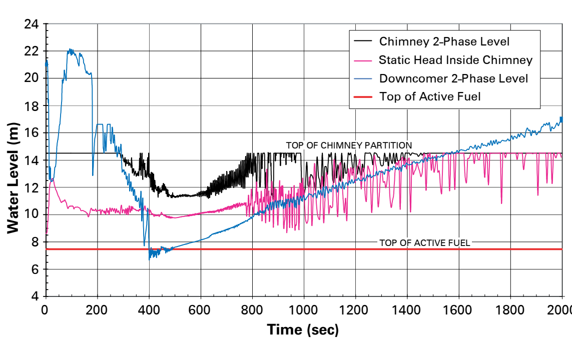


Figure 11-4. Bottom Drain Line LOCA - Water Levels

short term after a bottom drain line break. This event has the lowest short-term water level, but it is still well above the top of active fuel. Longer term, the water level in the RPV depends on the break location, but it will be at least as high as the spillover vent in the drywell (see Chapter 8).

The ability of the PCCS to remove decay heat and maintain containment pressure well within design limits is shown in Figure 11-5.

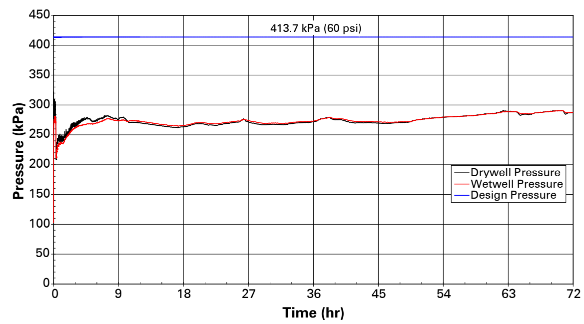


Figure 11-5. Feedwater Line LOCA - Containment Pressures

Calculated doses for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) were done using USNRC R.G. 1.183 with conservative meteorology, to bound potential sites. For example, the X/Q used for EAB was 1×10^{-3} s/m^3 which should allow even poor meteorological sites to establish the EAB at 800 m. The calculated doses of ~ 5 Rem (TEDE) are well within the regulatory limits.

Special Event Performance

Special events are those that are required by regulation to be analyzed regardless of expected frequency of occurrence. Two of the most challenging events are discussed here - Station Blackout (SBO), and Anticipated Transients Without Scram (ATWS).

Station Blackout events have historically been the most demanding for BWRs to cope with, and have usually been the dominant sequence for Severe

Accidents. However, the ESBWR passive features - IC, GDCS, PCCS - provide means for heat removal and inventory control. Due to loss of drywell cooling in an SBO, the combination of low water level in the RPV and high drywell pressure eventually leads to a blowdown, followed by GDCS injection and PCCS operation. The safe endpoint could be considered to the same as a "no-break" LOCA. Therefore the ESBWR coping time of at least 72 hours (without operator action) far exceeds the regulatory requirements.

With the adoption of FMCRDs and automation of actions needed to mitigate ATWS events, the probability of such events leading to significant consequences has been greatly reduced for ESBWR. Nonetheless, analyses have been performed to demonstrate meeting ATWS acceptance criteria. One of the most limiting of this class of transients is the MSIV Closure ATWS, because it challenges peak reactor power, minimum water level, RPV pressure, and suppression pool temperature. The response of ESBWR to an MSIV Closure ATWS is shown in Figures 11-6 to 11-8. The ESBWR safely mitigates this event.

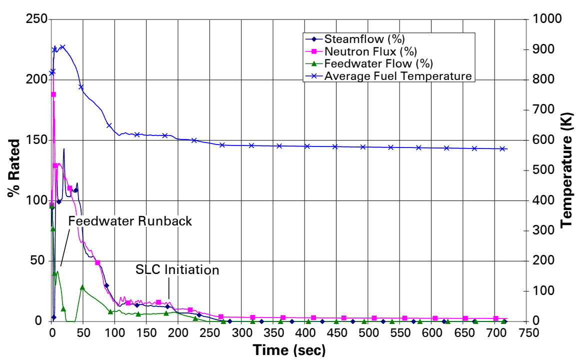


Figure 11-6. MSIV Closure ATWS - Power

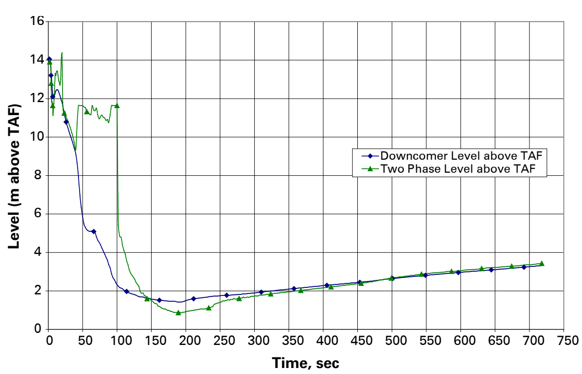


Figure 11-7. MSIV Closure ATWS - Water Level

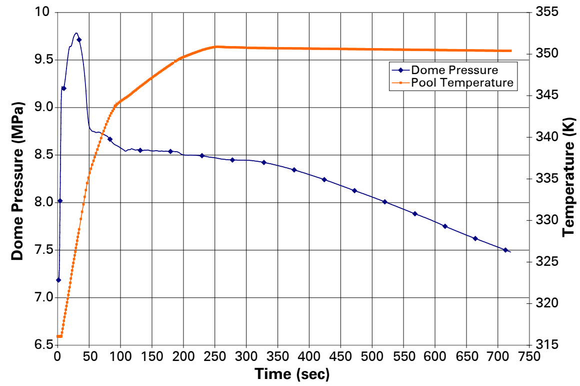


Figure 11-8. MSIV Closure ATWS - RPV Pressure, SP Temperature

Severe Accident Performance

Although demonstration of performance for the traditional set of design basis transients and accidents is important, in recent years regulatory emphasis has shifted toward performance for beyond design basis events, classified as "severe accidents". The ESBWR's 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 and development process. These evaluations influenced the design choices and certain design features in the final product. The final evaluation indicates that events resulting in damage to the reactor core are extremely unlikely, but that if such events were postulated to occur, passive accident mitigation features would limit the offsite dose such that the effect on the public and surrounding land would be insignificant.

In the ESBWR, GE has provided passive severe accident mitigation features to protect the containment from overpressurization and to limit the consequences to the public.

ESBWR Probabilistic Risk Assessment (PRA)

PRA studies played a major role in improving the overall plant design. For example, early PRAs were used in deciding to use diverse valves in the IC. Insights gained from the PRAs were used to improve plant technical specifications, emergency procedure guidelines, and the control room interface. The important insights from the PRA were also collected to provide input into the integrated reliability assurance program. These insights will be used throughout the lifetime of the plant to ensure that plant operations maintain a high level of safety.

A comparison of the internal events PRA for the ESBWR to PRAs performed for other reactors clearly demonstrates the overall improvement in safety (Figure 11-9). The USNRC risk goal for the frequency of core damage events in new reactors is 1×10^{-4}/ry, and the Utility requirement for ALWR plants is 1×10^{-5}/ry. The core damage frequency (CDF) for the ESBWR was found to be approximately 3×10^{-8}/ry. This represents a factor of 5 improvement compared to ABWR, and a factor of 100 or more improvement compared to most operating light water reactors.

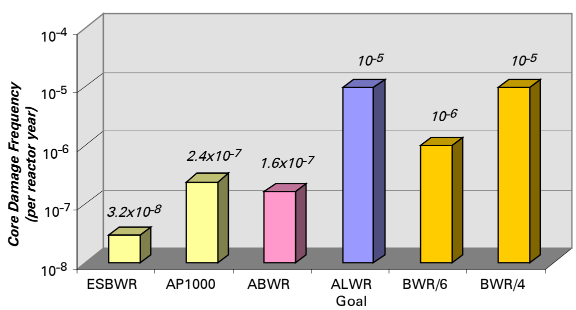


Figure 11-9. Comparison of Internal Event PRAs

Even with such a low CDF, due to the passive PCCS for containment heat removal and BiMAC core catcher (see Chapter 8), the Contingent Containment Failure Probability (CCFP) is 2.5%, meaning that the probability of a large release is exceedingly low. The key reasons for this are explained below.

The contribution to core damage by event is shown in Figure 11-10. As can be seen, transients (Loss of Feedwater, Loss of Offsite Power) dominate the risk. Another perspective is given in Figure 11-11, which shows the breakdown by Accident Class. It can be seen that low pressure core melt sequences dominate.

The fact that transients which lead to low pres-

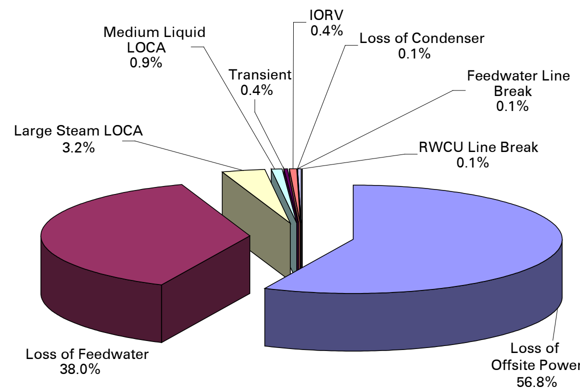


Figure 11-10. Core Damage Risk by Event

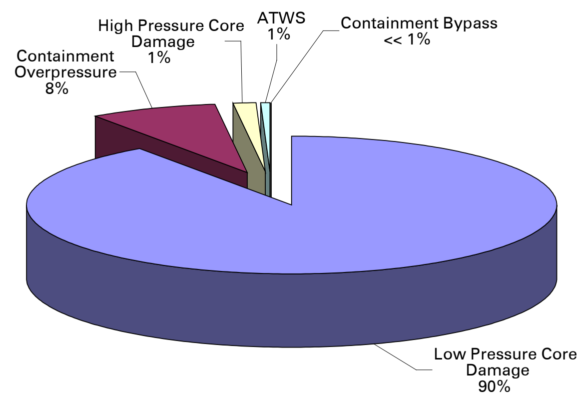


Figure 11-11. Core Damage Risk by Accident Class

sure core melt dominate core melt is the major reason why the CCFP is so low.

In addition events at power, the risk of core damage during shutdown was also evaluated. This can occur primarily because of operator errors or small line breaks which might drain the RPV while the lower containment hatches are open. The mitigating factor in these scenarios is the significant amount of time available to correct the situation. The quantitative evaluation of the risk while in shutdown mode is 4×10^{-9}/ry, a small fraction of the internal events risk.

Probabilistic methods were also applied to events initiated externally (e.g. tornado, flood, fire and earthquake). The important design features to ensure plant safety for each of these events were identified in a manner similar to that for the internal events PRA.

Tornado risk: The CDF due to a tornado was found to be extremely low because all safety components are located in the concrete Reactor Building and the internal events PRA already evaluates the ESBWR probabalistically for loss-of-offsite power due to other causes.

Flood risk: The objective of the ESBWR internal probabilistic flood analysis is to identify and provide a quantitative assessment of the core damage frequency due to internal flood events. Internal floods may be caused by large leaks due to rupture or cracking of pipes, piping components, or water containers such as storage tanks. Other possible flooding causes are the operation of fire protection equipment and human errors during maintenance.

The results of the ESBWR bounding analysis show that the CDF for internal flooding is considerably less than the total plant CDF. The risk from internal flooding is acceptably low. The following insights concerning the flooding mitigation capability of the ESBWR are identified:

- Safety system redundancy and physical separation for flooding by large water sources along with alternate safe shutdown features in buildings separated from flooding of safety systems give the ESBWR significant flooding mitigation capability.

- A small number of location-specific design features must be relied on to mitigate all potential flood sources. The flood specific features are: watertight doors on the Control and Reactor Buildings, floor drains in the Reactor and Control Buildings, Circulating Water System (CIRC) pump trip and valve closure on high water level in the condenser pit.

- While timely operator action can limit potential flood damage, all postulated floods can be adequately mitigated (from a risk perspective) without operator action.

Fire risk: The evaluation of fires was based on the Fire-Induced Vulnerability Evaluation methodology developed by the Electric Power Research Institute (EPRI). This conservative methodology provides procedures for performing quantitative screening analyses of fire risk. The results from this conservative screening analysis show that all the screening cases analyzed have a CDF much lower than the internal events CDF and therefore do not require a further detailed fire analysis. The following insights are provided on the fire mitigation capability of the ESBWR:

- Safety system redundancy and physical separation by fire barriers ensure that one fire limits damage to one safety system division. PIP system commonality is limited and is only affected by a few fire areas.

- Fires in the control room have the capacity to affect the execution of human actions. One feature relevant to the design is that a fire in the control room does not affect the automatic actuations of the safety systems. The remote shutdown panels allow the mitigation of any accident condition as if the main control room.

Seismic Risk: The risk of seismic events was evaluated using the seismic margins method. The ESBWR was designed for a Safe Shutdown Earthquake (SSE) of at least 0.3g. In the margins method, the margins implicit in the system designs are evaluated to determine a somewhat conservative estimate of the actual capacity of each system. Then, using fault trees and event trees similar to those developed for the internal events analysis, the system capacities are combined to determine the overall plant capacity. The ESBWR was shown to have a factor of margin to 0.3g SSE of more than two. This ensures that there is very little possibility of a core damage event as a result of an earthquake.

ESBWR Features to Mitigate Severe Accidents

In the event of a core damage accident, the ESBWR containment has been designed with specific mitigating capabilities. These capabilities not only mitigate the consequences of a severe accident but also address uncertainties in severe accident phenomena. The capabilities are listed below.

Isolation Condenser System (ICS): Although the primary purpose of the IC is to prevent lifts of SRVs during reactor isolation transients (see Chapter 3), the IC pool has been sized to provide capability for approximately 72 hours. This will provide a heat sink outside of containment during SBO events.

Passive Containment Cooling System (PCCS): The PCCS heat exchangers are located directly above the containment in water pools and form part of the containment boundary. There are no valves in the system and they act totally passively to remove heat added to the containment after any accident.

AC-Independent Water Addition: The Fire Protection System (FPS) and Fuel and Auxiliary Pools Cooling System (FAPCS) not only play an important role in preventing core damage through common lines but they are the backup source of water for flooding the lower drywell should the core become damaged and relocate into the containment (primary source is the deluge subsystem pipes of Gravity Driven Cooling System). The primary point of injection for these systems is the LPCI injection, through feedwater pipeline, to the reactor pressure vessel. Flow can also be delivered through the drywell spray header to the drywell. The drywell spray mode of this system not only provides for debris cooling, but it is capable of directly cooling the upper drywell atmosphere and scrubbing airborne fission products.

Three fire protection system pumps are provided on the ESBWR: two pumps are powered by AC power, the other is driven directly by a diesel engine. A fire truck can provide a backup water source.

Inerted Containment: The ESBWR containment is normally inerted with nitrogen containing < 3% oxygen (see discussion of the Containment Inerting System in Chapter 5). Therefore, any potential for hydrogen burning or detonation after a severe accident is avoided.

Basemat Internal Melt Arrest and Coolability Device: The ESBWR design uses a passively-cooled boundary that is impenetrable by the core debris in whatever configuration it could possibly exist on the lower drywell (LDW) floor. For ex-vessel implementation, this boundary is provided by a series of side-by-side inclined pipes, forming a jacket which can be effectively and passively cooled by natural circulation when subjected to thermal loading on any portion(s) of it. Water is supplied to this device from the GDCS pools via a set of squib-valve-activated deluge lines. The timing and flows are such that (a) cooling becomes available immediately upon actuation, and (b) the chance of flooding the LDW prematurely, to the extent that this opens up a vulnerability to steam explosions, is very remote. The jacket is buried inside the concrete basemat and would be called into action only in the event that some or all of the core debris on top is non-coolable. More details can be found in Chapter 8.

Analyses have shown that the containment will not fail by Basemat melt-through or by overpressurization as long as the BiMAC functions.

Manual Containment Overpressure Protection: 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 opening the wetwell atmosphere to the plant stack via a remote manual valve. Providing a relief path from the wetwell airspace precludes an uncontrolled containment failure. Directing the flow to the stack provides a monitored, elevated release. Relieving pressure from the wetwell, as opposed to the drywell, takes advantage of the decontamination factor provided by the suppression pool. This function of the Containment Inerting System is called the Manual Containment Overpressure Protection System (MCOPS); see Chapter 5.

Protection of the Public

The low core damage frequency combined with low failure probability of the containment leads to very low offsite doses, even after severe accidents. Figure 11-12 shows the offsite dose at 800 m (0.5 mile) as a function of probability for a nominal US site. It can be seen that large releases do not occur even at a one-in-a-billion probability per year.

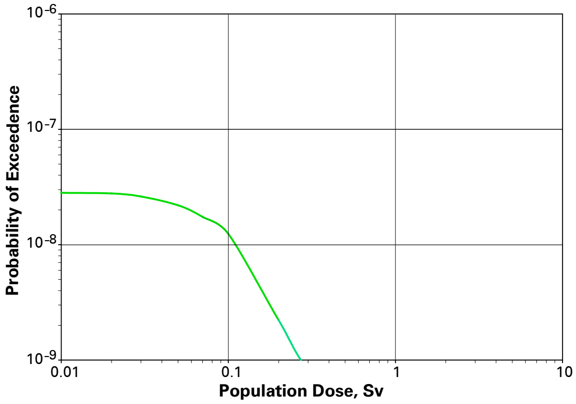


Figure 11-12. ESBWR TEDE Dose at 800m

Chapter 12 Plant Economics and Project Schedule

Introduction

A nuclear power plant makes an ideal baseload generator because its cost of electricity (COE) is mostly fixed. Of the total cost of electricity produced by an advanced nuclear plant, the capital costs are about 70%, O&M costs are 15% and fuel costs are also about 15%. The experience of recent years is that O&M costs are essentially fixed. Moreover, nuclear fuel costs have been remarkably stable over time. Thus, approximately 85% of the COE of a nuclear plant is fixed for the lifetime of the plant. Contrast this with a combined cycle plant for which the COE depends heavily on prices of natural gas (Table 12-1).

	Nuclear	Natural Gas
Fixed Costs	85%	20%
Variable Costs	15%	80%
Fuel Costs	stable	volatile

Table 12-1. Fixed and Variable Costs

A stable, low cost source of electricity is important to countries with growing economies. It supports economic growth and it makes the cost of products produced for export more competitive in world markets. Moreover, for countries which import large amounts of fossil fuels, the use of nuclear energy decreases vulnerability to changes in the supply and price of oil and gas.

During the development of the ESBWR, a good deal of emphasis was placed on ensuring that it would be economically competitive. Table 12-2 provides a comparison between key economic parameters of the ESBWR and those of existing U.S. BWRs.

Capital Costs

The capital cost of any nuclear plant depends upon a number of key variables. For example, the cost of labor, equipment, and commodities in the

	ESBWR	U.S. BWRs	
		Average	Best of Class
Capital Costs, $/kW	1400	3000 to 5000	2000
Construction schedule, years	3[1]	10-15	5
Capacity Factor, %	>95[2]	80	>90
Production costs[3], c/kWhr	1.0~1.5	2.0	1.5
Outage length, days	<14[4]	50	< 30
Staffing, people per unit	~300	800-1000	500

1. 36 months from first structural concrete to nuclear fuel load; duration of startup test program to commercial operation approximately 6 months
2. Refuel/maintenance outage every two years and assuming maximum forced outage duration of 5 days per year
3. O&M and fuel costs (projected)
4. Two year cycle, refuel only; shorter cycle, minimum shuffle refuel or colder cooling water may shorten by a few days

Table 12-2. Economic Comparison

host country—and the amount of local content—are important determinants of overall costs. So, too, are financing costs and escalation rates, which vary not only from country to country but depend upon how the project itself is structured.

To understand a cost estimate, therefore, it is important to know what assumptions have been made (as opposed to a firm price quotation in which all costs are known). Table 12-2 assumes the following:

Host Country:	United States
ESBWR Design:	Design Certified or U.S. version
No. of units:	Single, stand-alone unit on existing site
Rating:	1600 MWe gross/ 1535 MWe net*
Costs:	U.S. labor rates and productivity figures U.S. material and equipment costs
Design engineering costs	Standardized design developed in DoE supported NP-2010 program
Not included:	Owner's cost, escalation and financing costs

*Typical - dependent on site conditions and BOP/site interface design

This last set of assumptions, namely, what is not included, is in keeping with industry practice. In particular, because financing and escalation are so project dependent, the industry prefers to report capital costs in so-called "overnight costs"; that is, as if the plant could be built instantaneously. Potential owners of an ESBWR use the overnight cost to determine the as-constructed cost by applying escalation and financing costs appropriate for their situation.

As the direct result of the experience with two ABWR projects, contingency costs associated with the ESBWR are minimal.

Achieving a Competitive Capital Cost

The design of a nuclear power plant is a key factor in determining its capital cost. Design simplification, in particular, pays large dividends. The design effort, however, is certainly not the only source of cost reductions and maybe not even the most important. Managing capital costs is an on-going effort that spans the entire life of the design and construction of a plant. Table 12-3 summarizes all of the opportunities for reducing the capital costs.

Several of these items will be discussed.

Design Features

Design simplification and the use of new technology has reduced the amount of equipment and construction quantities in the ESBWR compared to the previous generation of BWRs. For example, the ESBWR eliminates the reactor recirculation system by employing natural circulation.

This eliminates the large external recirculation loops found in previous BWRs, and has many cost benefits. The large recirculation pumps, flow control valves, jet pumps, piping and pipe supports have all been eliminated. Also, the containment and Reactor Building are more compact, thereby reducing the amount of material quantities needed to construct them. Finally, because there are now no large nozzles below the top of the core, the safety systems can keep the core covered with water with less capacity and using gravity-driven cooling.

The design of the control rod systems has also been simplified. Fifty percent of the hydraulic control units (HCUs) in the control rod drive systems have been eliminated. Because the new Fine Motion Control Rod Drives (FMCRDs) discharge water directly into the reactor during a scram, the scram discharge volume and the accompanying piping have also been eliminated.

The use of new technology further reduces the amount of plant equipment and construction quantities. The use of fiber optic networks, which carry substantially more information, instead of copper

Opportunities to Reduce Costs	Sources of ESBWR Cost Reductions
Design features	Simplification, compact buildings, less equipment and quantities
Complete engineering before start of construction	Standardized design developed in DoE supported NP-2010 program
Shorter, predictable schedule	Construction experience, advanced information management system (IMS)
Institutional changes	One-step licensing
Construction techniques	Modularization, productivity, lessons learned
Learning curve	Transfer experience from ABWR projects
Standardization	Complete project engineering, advanced IMS
Global sourcing	Competitive equipment and component costs

Table 12-3. ESBWR Capital Cost Reductions

cabling, has eliminated 1.3 million feet of cabling and 135,000 cubic feet of cable trays. Use of microprocessors and solid-state devices in the control networks has reduced the number of safety system cabinets in the control room.

The ESBWR containment is a Reinforced Concrete Containment Vessel (RCCV). This technology was first introduced in a limited number in Mark III containments. The advantage of reintroducing this technology is that the containment can be made more compact, especially in comparison with the free-standing steel version of the Mark III design. The ESBWR containment volume is over 50% less compared to that design.

Shorter, Predictable Construction Schedule

Use of the RCCV has another important advantage – it reduces the construction schedule. Use of this containment and modular construction techniques reduces the overall construction schedule by an impressive seven months.

In constructing steel containments, the containment vessel is completed first, then the outer biological shield is erected, and, finally, the Reactor Building is constructed. For the RCCV, however, the construction of the containment vessel can take place concurrently with the construction of the floors and walls of the Reactor Building so that the entire construction schedule of the whole plant can be shortened. Also, RCCVs can be built in any shape. In the case of the ESBWR, this is generally a right circular cylinder, which was chosen because it is easier to construct.

The use of fiber optic cabling also reduces the construction schedule, in this case by one month, simply because there is less cable to install.

It is perhaps not generally appreciated that the ESBWR has been designed for extensive use of modular construction, in particular large modules. The entire control room (400 tons), the steel lining of the containment, the reactor pedestal, the turbine generator pedestal, and the upper drywell structure with piping and valves are notable examples.

Several studies have shown that the length of the construction period does not materially affect capital costs, provided the schedule is met. It is when construction takes longer than expected that capital costs, namely interests costs, become significantly higher. In other words, spreading the costs over a

4-year instead of a 3-year period does not incur substantially more interests costs if properly managed. Falling behind on the schedule, however, will cause interest costs to soar. Utilities appreciate this too, since it means they will have the electricity when expected and won't have to use, or purchase, expensive replacement energy.

Experience

There is no substitute for experience. The Lungmen ABWRs are being supplied by a team of worldwide 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. The ESBWR will build on ABWR technology and construction experience. This is the basis for the "learning curve" effect, which reduces capital costs by about 10% with each new unit.

Global Sourcing

One of the most important tasks in the delivery of a nuclear plant is the establishment of a dependable, cost competitive source of suppliers. This is by no means an easy task and, once established, this network is an extremely valuable asset for future projects. Currently, ABWR projects are supported by a worldwide network of suppliers. GE and its sub-suppliers work closely together to deliver a cost effective, high quality product, using such collaborative tools as Six Sigma used throughout the General Electric Company.

Production Costs

Once the plant is built, capital costs are basically sunk costs; there are, however, the occasional costs of capital addition that occur during the life of the plant. The on-going costs of producing nuclear electricity are the fuel cycle costs and the costs of plant operation and maintenance (O&M), the sum of which are referred to as the production costs. In theory, production costs are variable with the amount of electricity produced. For a base-loaded nuclear plant, however, the reality is that, in *total dollars*, O&M costs are largely independent of how much electricity is actually generated in given operating cycle.

The ESBWR production cost is projected to be between 1.0 and 1.5 cents per kWhr, depending on uranium supply costs and utility O&M practices, which would place it in the best-of-class category for U.S. operating BWR plants.

A typical figure for the ESBWR fuel cycle cost is about 0.5 to 0.6 cents per kWhr. As such, it represents only about 15% of the total cost of electricity (COE). The overall nuclear electricity cost then is not sensitive to changes in fuel prices. For example, if the cost of uranium were to *double*, the total COE would only increase by 3%.

A bottoms-up estimate of the O&M cost is 0.65 cents per kWhr, although this may vary, depending on utility O&M practices. In the 1990's, the nuclear industry made a concerted effort to contain and then reduce O&M costs, which had risen to a level high enough to threaten the economic viability of operating plants. Many good practices came out of that effort, which demonstrated that capable plant management — people with know-how and skill — are the key to low O&M costs. A nuclear plant designed for ease of maintenance can help but not replace this essential element.

Ease of Maintenance

The ESBWR is designed for ease of maintenance and reduced staff. For example, inside the containment, equipment at all levels is accessible by stairs and platforms which encircle the vessel. Monorails are available to remove the equipment, such as a main steam isolation valve, to a conveniently located service room via an equipment hatch. Removal of the FMCRDs has been automated. Handling devices, which in the case of the FMCRD are operated remotely from outside the containment, engage and remove the equipment. The drive is laid on a transport device and removed through the equipment hatch. Just outside the hatch is a pathway to a dedicated service room in the adjacent Fuel Building, where the equipment can be decontaminated and serviced in a shielded environment. The entire operation is done efficiently and with very little radiation exposure to the personnel.

Some other reasons why the ESBWR design will have less staffing needs are presented in Table 12-4.

Capacity Factor

The ESBWR can operate with a cycle length up to 24 months. ESBWR design improvements would indicate that the ESBWR should experience no unplanned scrams or other types of forced outages. The capacity factor, in this case, will be determined primarily by the length of the refueling and maintenance outage.

The expected capacity factor for the ESBWR is >95% based upon an outage length of 3 weeks or less on a 24-month operating cycle, and a maximum of 5 days per year forced outage.

The design of the ESBWR lends itself to these short outages. The ESBWR has four redundant divisions of passive safety systems which require little maintenance, and four divisions of protection systems which can be maintained on-line. Furthermore, many of the normal power generation systems support on-line condition monitoring and on-line maintenance. Therefore, during operation there will be more preventative maintenance programs, which keep major refurbishments off the critical path of the refueling outage.

Maintenance staff
- Less maintenance requirements
- Maintenance made easier
- Shorter outage lengths
- Fewer major repairs

Technical support staff
- Standardized, pre-licensed design
- Electronic Configuration Management
- Standardized training, operator and maintenance procedures

Operating staff
- Self-diagnosing C&I systems
- Plant automation
- Fewer technical specifications

Table 12-4. Reducing Staffing Levels

Project Schedule

The ESBWR project schedule consists of a 36 month construction schedule, as measured from when first structural concrete is poured to fuel load, followed by a six month startup test program, leading to commercial operation. This is preceded by a period during which the plant is licensed, lead equipment is ordered and the site is prepared for construction, including about a three-month excavation period.

The Kashiwazaki ABWR units were built in record time for a nuclear power plant (part of the reason was the extensive use of large construction modules). From first concrete to fuel loading, construction of the plant took only 36.5 months, all the more impressive because these were first-of-a-kind units. The overall construction schedule was 48 months. This experience provides confidence that the ESBWR can also be constructed in 36 months as there are basic similarities in equipment, systems and structural technologies.

A project schedule, with its key milestones, is given in Figure 12-1.

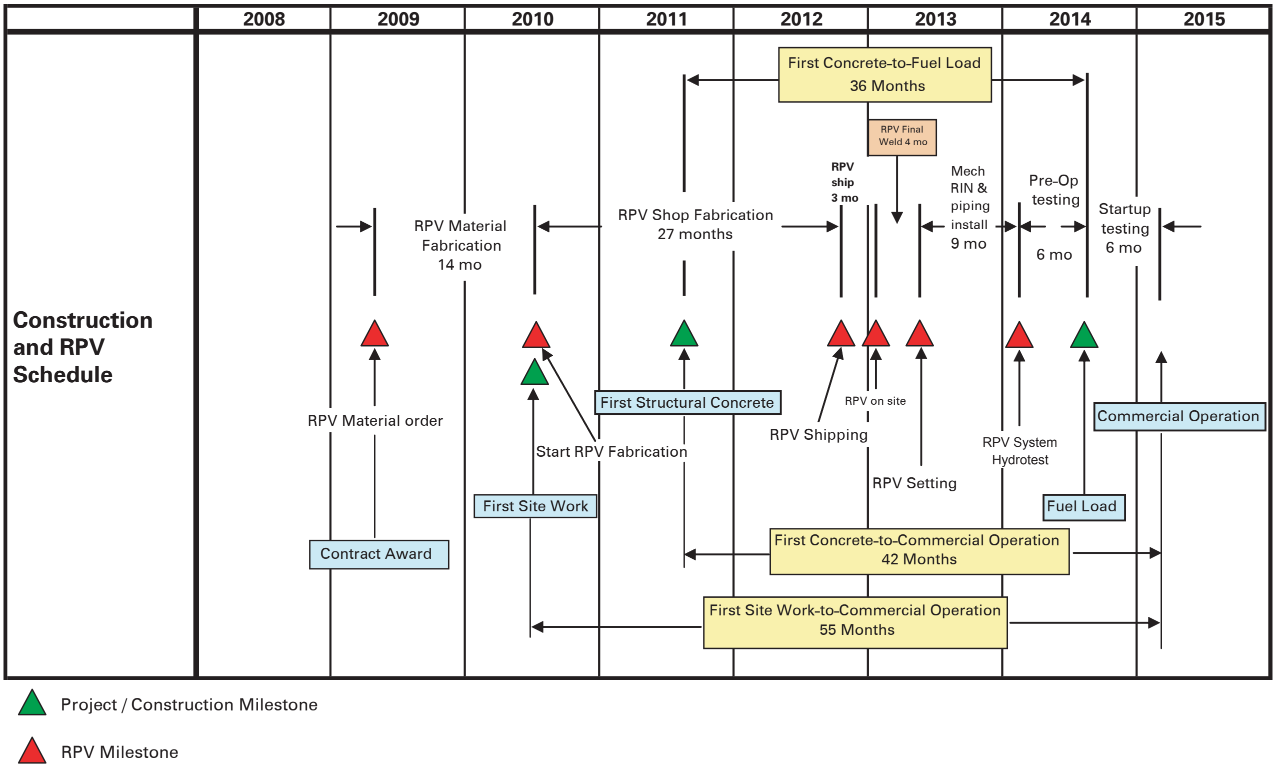


Figure 12-1. ESBWR Project Schedule

Appendix A
Key Design Characteristics

This appendix lists key design characteristics for the ESBWR, using the standard design licensed in the US as a reference. Further details can be obtained from theESBWR Standard Safety Analysis Report, Chapter 1 (26A6642AD).

Overall Design	
Site Envelope	
Safe shutdown earthquake, g	0.3 envelope
Wind design, km/h	225
Maximum tornado, km/h	531
Max dry bulb/wet bulb ambient temperature, °C	46/27
Thermal and Hydraulic	
Rated Power, MWt	4500
Generator Output, MWe	1600
Steam flow rate, Mkg/h	8.76
Core coolant flow rate, Mkg/h	36.0
System operating pressure, MPa	7.17
Average core power density, kW/l	54.3
Maximum linear heat generation rate, kW/m	44.0
Average linear heat generation rate, kW/m	15.1
Minimum critical power ratio (MCPR)	1.4 - 1.5*
Core average exit quality, %	17.0
Feedwater temperature, °C	215.6

* Depending on the cycle length

Core Design	
Fuel Assembly	
Number of fuel assemblies	1132
Fuel rod array	10 x 10
Overall length, cm	379
Weight of UO2 per assembly, kg	144
Number of fuel rods per assembly	92
Rod diameter, cm	1.026
Cladding material	Zircaloy-2
Fuel Channel	
Thickness corner/wall, mm	3.05/1.91
Dimensions, cm	14 X 14
Material	Zircaloy-2
Reactor Control System	
Method of variation of reactor power	Moveable control rods
Number of control rods	269
Shape of control rods	Cruciform
Pitch of control rods, cm	31
Type of control rod drive	Bottom entry electric hydraulic fine motion
Rod step size, mm	36.5
Number of hydraulic accumulators	135
Hydraulic scram speed, sec to 60% insert	1.15
Electric drive speed, mm/sec	30
Type of temporary reactivity control	Burnable poison; gadolinia urania fuel rods
High pressure coolant injection 1/2 pumps, m^3/h	118/235
Incore Neutron Instrumentation	
Total number of LPRM detectors	256
Number of incore LPRM penetrations	76
Number of LPRM detectors per penetration	4
Number of SRNM penetrations	12

Reactor Vessel and Internals	
Reactor Vessel	
Material	Low-alloy steel/ stainless and Ni-Cr-Fe alloy clad
Design pressure, MPag	8.62
Inside diameter, m	7.1
Inside height, m	27.6
Steam Separators and Dryers	
Separator type	AS-2B
Number of separators	379
Dryer type	Chevron
Main Steam	
Number of steam lines	4
Diameter of steam lines, cm	70
Number of safety/relief valves	18
Number of depressurization valves	8
Isolation Condenser	
Number of loops	4
Capacity of each loop, MWt	34
Number of safety/relief valves	18

Emergency Core Cooling	
Gravity Driven Core Cooling	
Number of loops	4
Number of pumps	0
Flow rate, m^3/h	500*
Automatic Depressurization	
Number of relief valves	10
Number of depressurization valves	8
Passive Containment Cooling System	
Number of loops	6
Heat removal duty per loop, MWt	11
Standby Liquid Control	
Number of accumulators	2
B10 enrichment, %	94
Capacity per accumulator, m^3	7.8
Initial flow rate per accumulator, m^3/h	66

* At runout

Containment	
Primary	
Type	Pressure suppression
Construction	Reinforced concrete with steel liner
Drywell	Concrete cylinder
Wetwell	Concrete cylinder
Design pressure, MPa	0.31
Design leak rate, % free volume/day	0.5*
Drywell free volume, m^3	7206
Wetwell free volume, m^3	5467
Suppression pool water volume, m^3	4383
Number of vertical vents	10
Vertical vent diameter, m	1.2
Number of horizontal vents/vertical vent	3
Horizontal vent diameter, m	0.7
Reactor Building	
Type	Low leakage
Construction	Reinforced concrete/steel
Design in leakage rate at 6.4 mm water, %/day	100

* Excluding MSIV leakage

Auxiliary Systems	
Reactor Water Cleanup/Shutdown Cooling	
Number of trains	2
Number of pumps per train high/low capacity	1/1
Type	Canned rotor
Flow rate per train (cleanup mode), m^3/h/ % of feedwater	116/1
No. of regenerative heat exchangers per train	2
No. of non-regenerative heat exchangers per train	3
Return water temperature (cleanup mode), °C	227
Flow rate (shutdown mode), m^3/h	1365
Heat removal duty (shutdown), MWt	55
Fuel and Auxiliary Pools Cooling	
Number of trains	2
Number of pumps/train	1
Flow rate per pump, m^3/h	250
Number of heat exchangers/train	1
Total heat removal capability, MWt	4.0
Backup LPCI flow/train, m^3/h	454
Reactor Component Cooling Water	
Number of trains	2
Capacity of each train, %	100
Number of pumps per train	3
Number of heat exchangers per train	3
Flow rate per loop (normal), m^3/h	1250
Heat removal duty (normal), MWt	31
Flow rate per loop (shutdown), m^3/h	2500
Heat removal duty (shutdown), MWt	86
Plant Service Water	
Number of trains	2
Capacity of each train, %	100
Number of pumps per train	2
Flow rate per loop (normal), m^3/h	9085
Drywell Cooling	
Number of trains	2
Flow rate per train, m^3/h	72800
Number of fans per train	4
Heat removal duty per train, MWt	1.78

Appendix Frequently Asked Questions B

What proof is there that natural circulation works in such a large reactor?

History

Natural circulation in Boiling Water Reactors (BWR) is a proven technology. Some of the early GE BWRs employed natural circulation. These were small plants (e.g. Dodewaard at 183 MWt and Humboldt Bay at 165 MWt), but they clearly demonstrated the feasibility of the BWR and provided valuable operating data and experience. GE moved to forced circulation plants to achieve higher power ratings in a compact pressure vessel. Pressure vessel fabrication capability at the time was a factor in this decision. Now, after several decades, GE is returning to natural circulation for the ESBWR.

Natural circulation provides major simplification by removal of the recirculation pumps and associated piping, heat exchangers and controls. It is also synergistic with two other requirements that GE considered to be important in the design of a new reactor: large safety margins with a very reliable passive Emergency Core Cooling System (ECCS), and avoidance of safety/relief valve (SRV) opening for pressurization transients such as turbine trips or main steam line isolation events. Both of these features need a tall pressure vessel with large water volumes. The tall vessel leads to enhanced natural circulation flow, so the natural circulation capability comes with no additional cost.

Evolutionary Design

ESBWR builds on the design features of operating BWRs. Figure B-1 shows a cutaway of the ESBWR reactor pressure vessel. Most components in the ESBWR are standard BWR components that have been operating in the field for years (steam separators, control rods and guide tubes, core support structure, etc.). The main difference is the taller reactor vessel with the addition of a partitioned chimney above the core and a correspondingly taller downcomer annulus. The fluid in the taller downcomer provides the additional driving head for natural circulation flow through the core, as well as a large water inventory for a Loss of Coolant Accident (LOCA). Steam in the chimney also provides a cushion to dampen void collapse in the core during pressurization transients, leading to a softer response

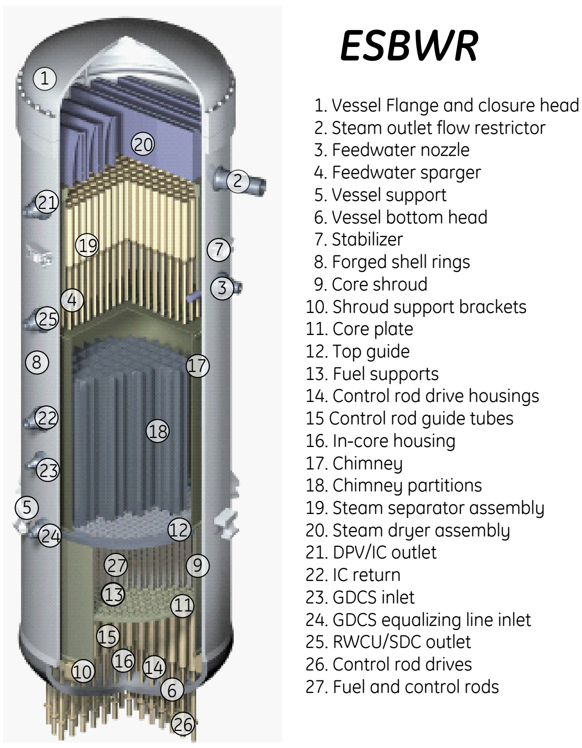


Figure B-1. Cut-away of ESBWR Reactor Assembly with no SRV discharges.

Figure B-2 shows a view of the partitioned chimney and its layout above the core. The core consists of conventional BWR fuel bundles, shortened from 12 ft. to 10 ft. to improve pressure drop and stability characteristics. The absence of hardware in the downcomer (jet pumps or internal pumps) reduces flow losses and further enhances natural circulation. Figure B-3 shows how the taller, open downcomer and reduced core resistance lead to a great enhancement of natural circulation flow in the ESBWR relative to operating BWRs.

Operating Experience

Valuable operating experience was gained from the early natural circulation BWRs. It was demonstrated that BWRs could operate in natural circulation without problems. The plants were extremely stable and presented no unusual characteristics relative to noise in the instrumentation. Power was raised by control rod withdrawal. The ESBWR will also adjust output using control rods, but with electrically driven control rod drives that move slower and have finer positioning capability than the locking piston design for conventional BWRs.

Present-day BWRs can operate at about 50% of rated power in natural circulation; however, stability considerations prevent steady operation in this region. There have been recirculation pump trip events in operating plants, which led to a natural circulation state at around 50% of rated power. The operating conditions in present day BWRs (power, flow, power distribution) in natural circulation following the pump trip were well predicted by the calculational models used for ESBWR performance analysis.

The operating parameters for the ESBWR, such as the power density, steam quality, void fraction and void coefficient are within the range of operating plant data. Figure B-4 shows a comparison of the ESBWR power – flow operating map with those of operating BWRs. This figure is based on the power per bundle and flow per bundle so that a meaningful comparison can be made. The power per bundle and flow per bundle for the ESBWR are both lower than for a modern jet pump plant at rated operating conditions, but the ratio of power to flow is similar to that for an uprated BWR at Maximum Extended Load Line Limit Analysis -Plus (MELLLA+) conditions. This means that the core exit steam quality (ratio of steam flow to core flow) is also similar.

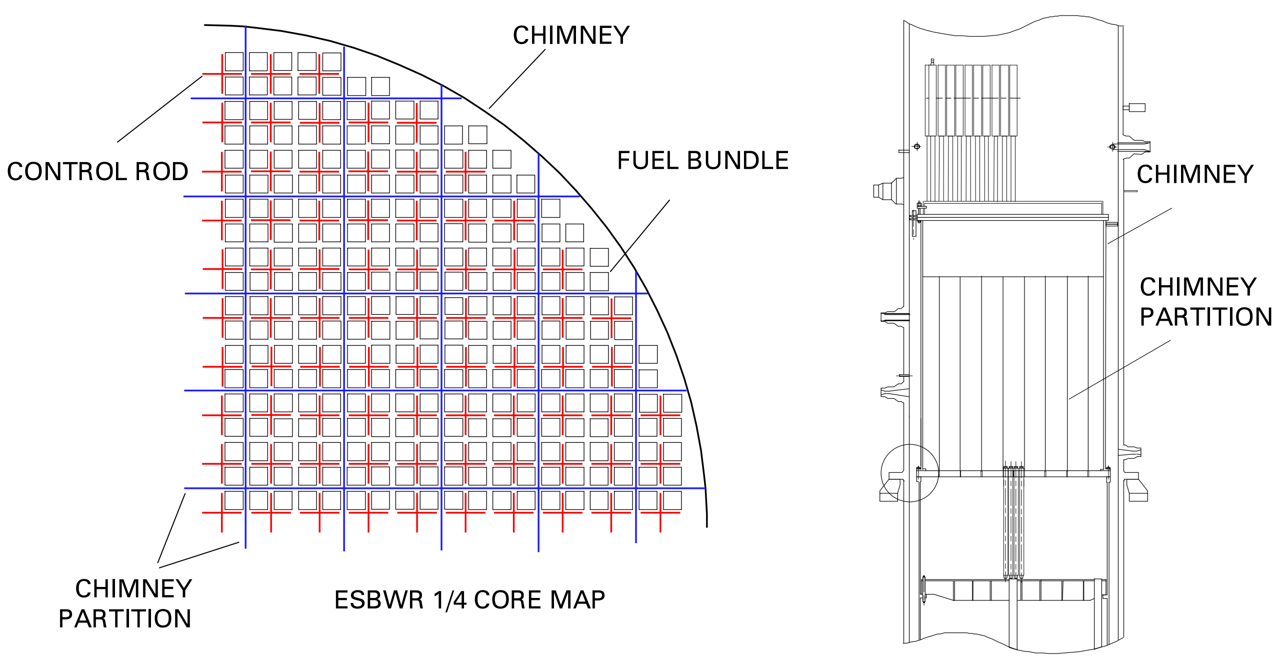


Figure B-2. ESBWR Core and Chimney

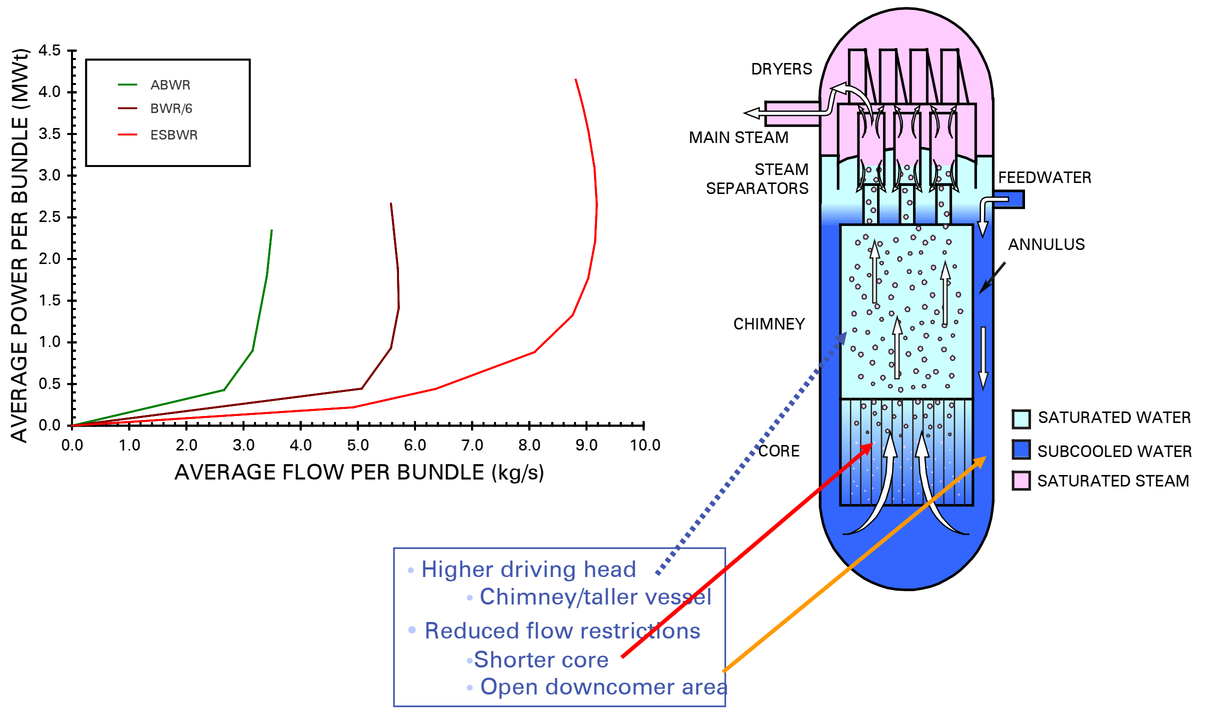


Figure B-3. ESBWR Enhanced Natural Circulation

Test Data and Code Validation

For analysis of the ESBWR, GE uses the state-of-the-art TRACG code. GE has invested over a hundred man-years in the development and validation of this technology, which originated in the National Laboratories. A systematic strategy was adopted, using separate effects tests, component performance tests, integral system tests and BWR data to validate TRACG.

Natural circulation flow in ESBWR is driven by the difference in the static head between the downcomer annulus outside the core shroud and the static head in the core and chimney inside the core shroud. The flow is governed by the flow loop losses, primarily in the core and separators. There is a large data base for the pressure drop in the fuel channels and separators. Extensive data are also available for the void fraction in the fuel bundles.

Additionally, the test data base was augmented for the chimney region by tests performed at Ontario Hydro. These tests measured the void fraction in pipes with diameter and height similar to a cell of the partitioned chimney. These data were obtained at pressures, flow rates and void fractions representative of ESBWR operation. All these test data have been used validate the TRACG code. Figure B-5 shows a schematic of the Ontario Hydro test facility and a comparison of the measured and calculated void fraction in a 0.53m diameter pipe. A chimney cell in the ESBWR has a similar hydraulic diameter (0.6m). The good agreement with data provides confidence in the ability of TRACG to correctly calculate the flow regimes and void fractions inside the partitioned chimney cells.

Uncertainties in the calculation of the natural circulation flow are small, and can readily be accommodated in the design process. The uncertainties in parameters that govern the natural circulation flow, such as the core frictional losses, separator frictional losses and chimney void fraction, were statistically combined through a Monte Carlo process, resulting in an overall uncertainty (1 σ) of approximately 3% in the calculation of core flow. Calculations of natural circulation flow for operating plants following pump trips (Oyster Creek, Hatch, LaSalle) confirm the accuracy of the TRACG predictions. Figure B-6 shows a comparison of measured and calculated

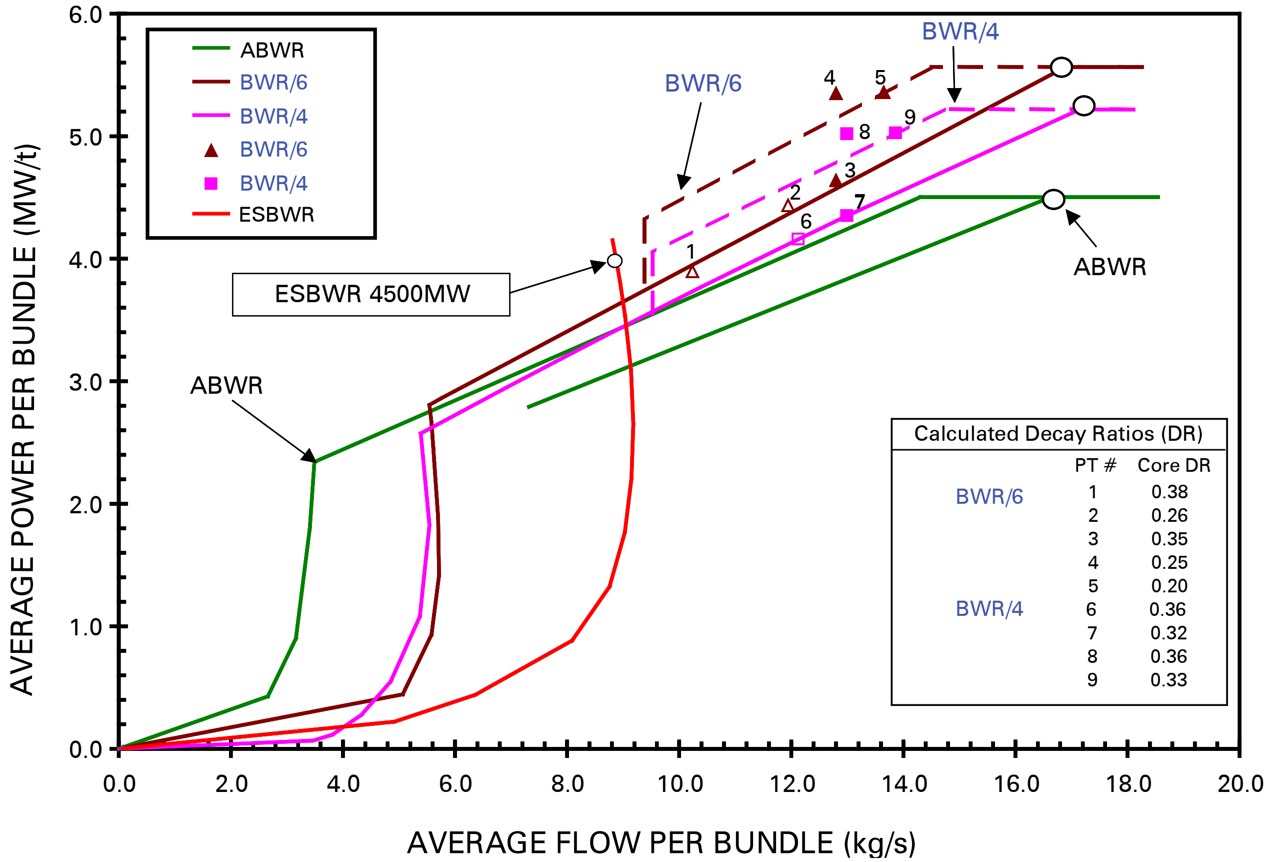


Figure B-4. Comparison of ESBWR Operating Map with Operating BWRs

flows for the LaSalle plant following a trip of the recirculation pumps. The magnitude of the flow rate is seen to be calculated very accurately.

Stability data have been obtained at test facilities and in operating BWRs. The phenomena that govern stability are the same in the ESBWR and operating BWRs. The FRIGG facility in Sweden tested hydrodynamic stability performance; the SIRIUS test facility in Japan was scaled to SBWR and investigated stability under startup conditions as well as higher pressures. Figure B-7 shows a comparison of the measured and calculated oscillations in the test facility at 7.2 MPa and the measured and calculated stability maps (regions of instability). TRACG is able to capture the characteristics of these oscillations as well as the onset of instability. Note that the region of instability is far removed from the ESBWR operating state. Operating plant data from LaSalle, Leibstadt, Forsmark, Cofrentes, Nine Mile Point 2 and Peach Bottom 2 have been used to benchmark TRACG predictions of stability performance. As an example, Figure B-8 shows a comparison of the regional oscillation profile seen in test data from a European BWR with the corresponding TRACG calculation. The agreement is excellent. The shape of the oscillation profile corresponds closely to the shape of the first azimuthal harmonics of the core neutronics.

The NRC and ACRS have reviewed the test data base and TRACG validation, and have concluded that additional tests are not required. The NRC staff has also completed a favorable Safety Evaluation of the applicability of TRACG for analyzing ESBWR stability.

ESBWR Stability

The enhanced natural circulation flow rate in the ESBWR greatly improves stability performance relative to operating BWRs at natural circulation conditions. Figure B-4 shows that the rated oper-

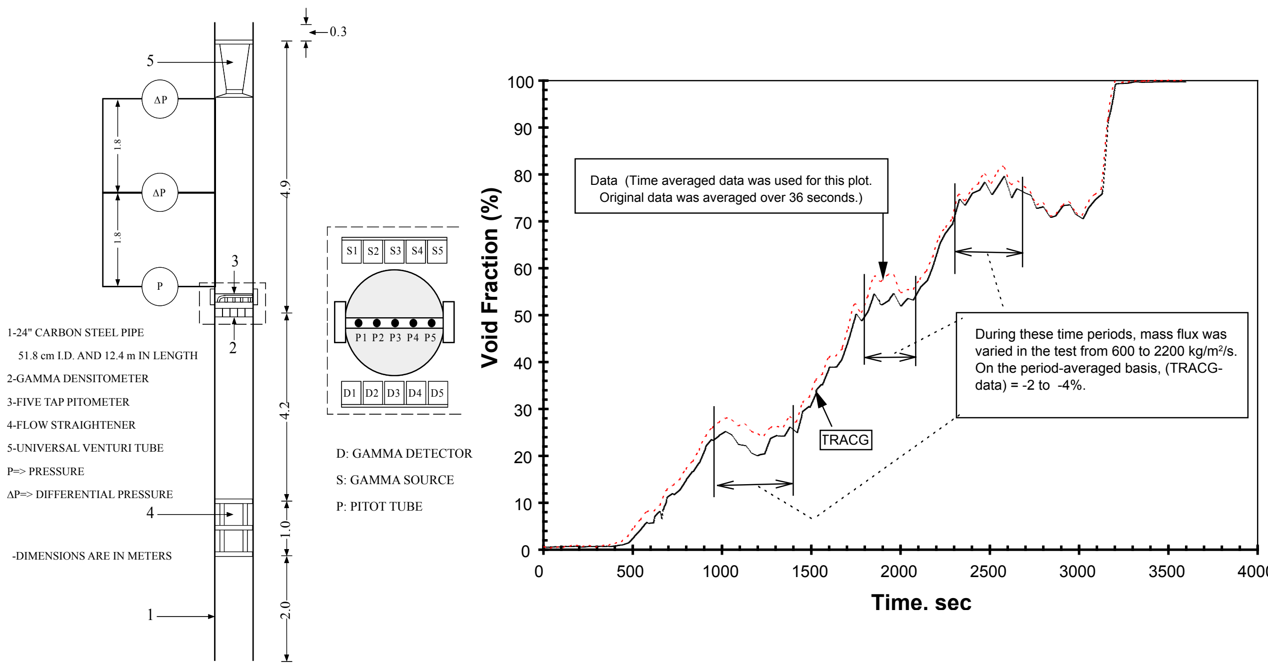


Figure B-5. Comparison of Ontario Hydro Test Data to TRACG

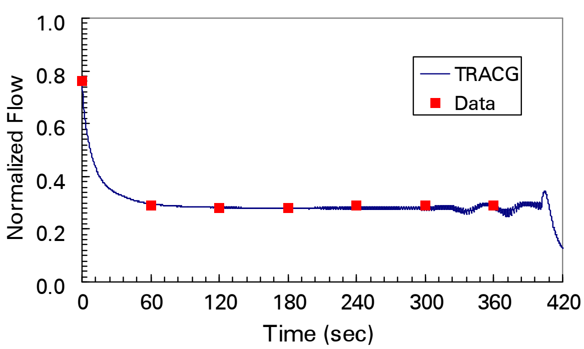


Figure B-6. Comparison of BWR Natural Circulation Flow Following Pump Trip to TRACG

ating conditions for the ESBWR are closer to an uprated BWR in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) region, rather than at natural circulation.

Additionally, two other factors improve ESBWR stability significantly: the ratio of single phase pressure drop (stabilizing) to two-phase pressure drop (destabilizing) is higher for the shorter ESBWR fuel; and the ratio of fuel time constant to the time period for a stability resonance is higher. This reduces the destabilizing neutronic feedback resulting from void perturbations. The net result is a very stable reactor that easily meets the very conservative design /licensing criteria. Stability performance is measured in terms of decay ratios for the channel, core wide and regional stability modes. A lower decay ratio implies a more stable plant. Figure B-9 compares the calculated decay ratio for the ESBWR for these modes of stability with the licensing limits of 0.8 for each stability mode. The figure shows that even at the 2 sigma (95% confidence) level, there is substantial margin to the stability limits. At power levels lower than rated, ESBWR stability improves further.

Plant Startup and Load Following

The startup procedure for ESBWR will follow the established process at Dodewaard. The Dodewaard plant operated through twenty three (23) cycles for 30 years with no problems during startup. TRACG calculations for Dodewaard startup 22 are shown in Figure 10. TRACG captured the main trends of the startup sequence. TRACG calculated some noise in the downcomer flow at the initiation of voiding in the chimney. This occurs at very low power levels, before the start of boiling in the core. However, this phenomenon was not noticeable in the measurements of the downcomer flow or the on

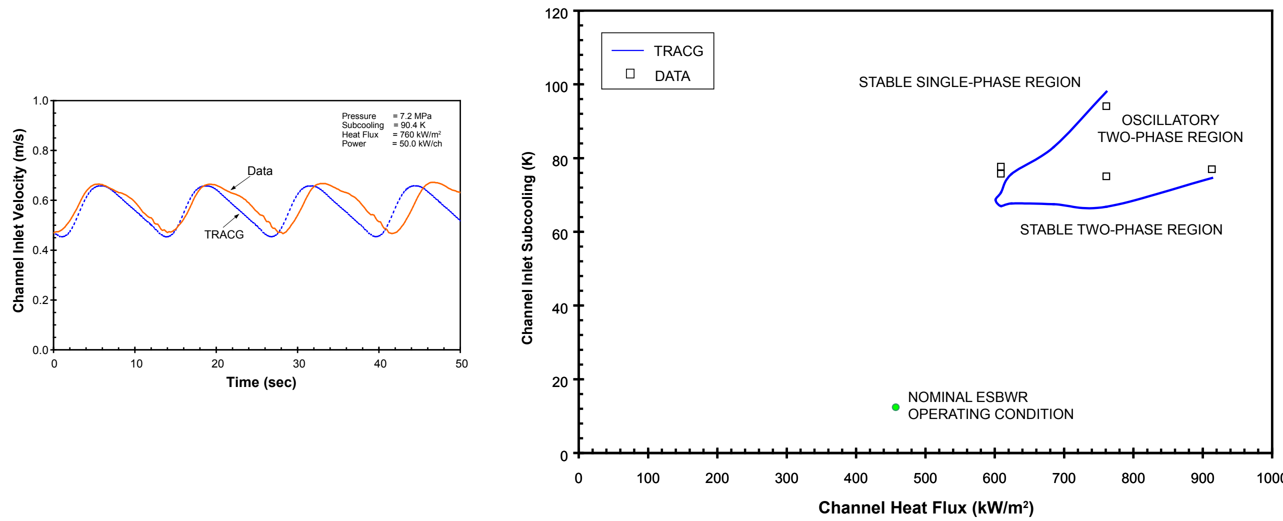


Figure B-7. Comparison of SIRIUS Test Data to TRACG

the in-core flux instrumentation. TRACG may be magnifying the flow noise in its calculations. In any event, at the low power levels, no thermal margins are approached. TRACG simulations of ESBWR startup demonstrate that the plant can be started up and reach rated conditions without difficulty, while maintaining large margins to thermal limits.

Because of its size, ESBWR will not normally be operated in a load follow mode. However, changes in core power can be readily accomplished through movement of the Fine Motion Control Rod Drives (FMCRD). The FMCRDs can accommodate a duty corresponding to daily load following cycles for 10 years.

Natural Circulation Benefits

In summary, natural circulation is a proven technology that provides numerous benefits. Natural circulation allows for the elimination of several systems, including recirculation pumps and associated piping, valves, heat exchangers, motors, adjustable speed drives and controllers. The larger RPV employed for natural circulation provides synergy with the use of passive ECCS and improves the response to operational transients and increased safety margins. Flow transients resulting from recirculation pump anomalies are not present; i.e. no runbacks or trips that would challenge stability.

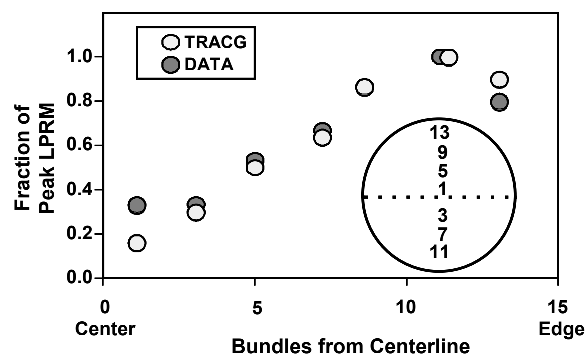


Figure B-8. European BWR Measured and Calculated Regional Oscillation Contour

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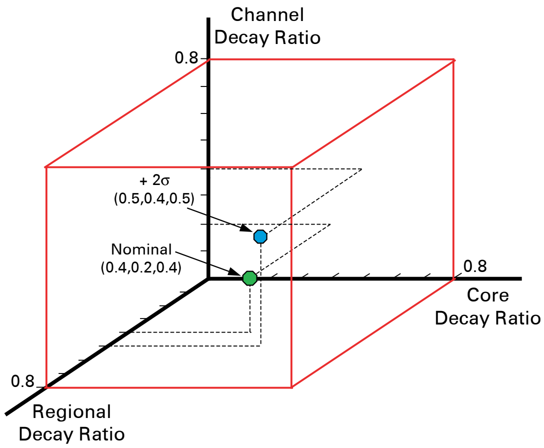

Figure B-9. ESBWR Stability Margins

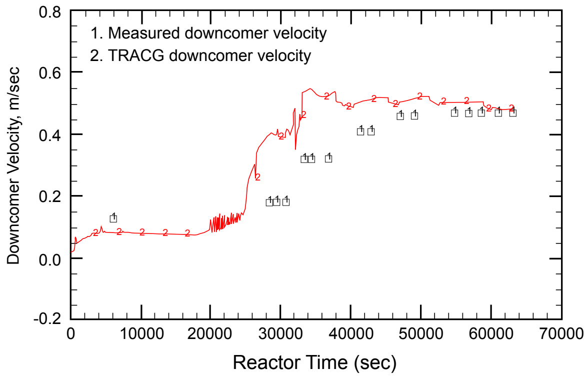

Figure B-10. Comparison of Measured and Calculated Downcomer Velocities for Dodewaard Startup

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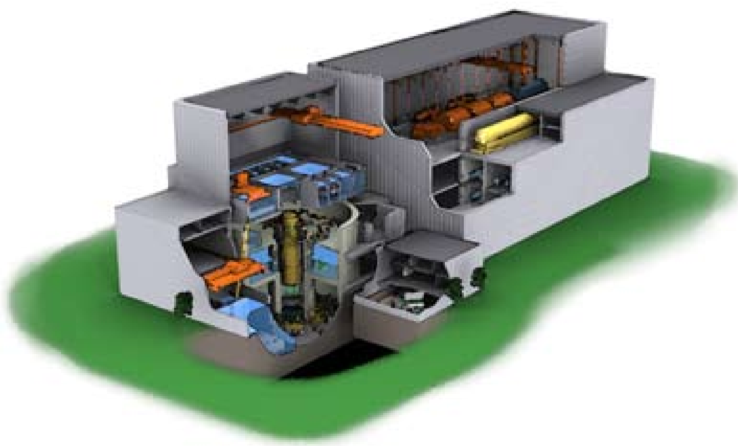
ESBWR General Description



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