GE Nuclear Energy

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April 26, 1991

MFN No. 040-91
Docket No. STN 50-605
EEN-9131

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Charles L. Miller, Director
Standardization and Non-Power Reactor Project Directorate

Subject: Summary Status of GE/NRC March 4-6, 1991, Meeting on Plant Systems Open Items

Reference: Summary Status of GE/NRC March 4-6, 1991, Meeting on Plant Systems Open Items, MFN No. 028-91, dated March 28 1991

Enclosed are thirty four (34) copies of the second portion of the GE responses to the subject open items. The first portion of these responses was provided to the NRC via the above reference.

It is intended that GE will amend the SSAR, as appropriate, with these responses in a future amendment.

Sincerely,

P. W. Marriott

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SUMMARY STATUS OF GE/NRC MARCH 4-6, 1991 MEETING ON
PLANT SYSTEMS OPEN ISSUES

Number	Subject	Action	Comments
	3.11 EQUIPMENT QUALIFICATION		
3.11(1)	Time Margin	GE	Response provided 3/28/91.
3.11(2)	IEEE Edition	GE	Response provided 3/28/91.
3.11(3)a	Gammma Accident Dose	GE	Response provided 3/28/91.
3.11(3)b	Worst Case Expected Environment	GE	Response provided 3/28/91.
	CLARIFICATION OF PAST ISSUES AND RESPONSES FOR APPENDIX 3I		
	Chemical Environmental Conditions		Response provided 3/28/91.
	Spray and/or Submergence Enviroment Data		Response provided 3/28/91.
	Beta Radiation Enviromemt Data		Response provided 3/28/91.
	Limiting Accident Scope		Response provided 3/28/91.
	Significant Enveloping Abnormal		Response provided 3/28/91.
	Environmentally Mild or Harsh Zones		Response provided 3/28/91.
	Typical Equipment Located in Zones		Response provided 3/28/91.
	Limited Locations of Safety-Related Equipment		Response provided 3/28/91.
	Inconsistancy in Presssure Units		Response provided 3/28/91.
	Deletion of Tables		Response provided 3/28/91.

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Item Number	Subject	Action	Comments
	3.5.1.1 PROTECTION OF SAFETY-RELATED EQUIPMENT		
3.5.1.1 (1)	Separation of Satety-Related and non Safety-Related Equipment	GE	Response provided 3/28/91
3.5.1.4	DESIGN BASIS TORNADO		
(1)	ANS 2.3 to SRP	GE	GE still evaluating impact. Also, pursuing E-7 recurrence interval.
	3.5.2 PROTECTION OF CHARCOAL DELAY TANKS		
3.5.2 (1)	Relative position of tanks in turbine building.	None	Closed.
	3.6.1 WORST CASE FLOODING		
3.6.1 (1)	Total failure of non-Seismic Piping Systems	NRC	NRC committed to providing a reference regulatory basis.
	3.6.1 STEAM TUNNEL		
3.6.1 (1)	Analysis of Steam Tunnel for Pipe Breaks	GE	Response provided on pages 3.6-5 and 3.6-27.
	3.6.1 HIGH ENERGY PIPING LINES		
3.6.1 (1)	Exemption of Selected High Energy Pipes	GE	Response provided 3/28/91.
	3.6.1 DBA RUPTURE OF HIGH OR MODERATE ENERGY LINE		
3.6.1 (1)	Habitability of Control Room Due to Pipe Break and DBA Analysis		
	- Habitability of Control Room Portion	GE	See response to 3.6.1(1) (Steam Tunnel) above.
	- DBA Analysis Portion	GE	Response provided on pages 6.2-22, 22a, 23, and 44 attached.

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Item Number	Subject	Action	Comments
	6.2.6 CONTAINMENT LEAKAGE TESTING		
6.2.6 (1)	Systems not Vented or Drained (Type A)	GE	Response provided 3/28/91.
6.2.6 (2)	Systems not be be Vented or Drained	GE	Response provided 3/28/91.
6.2.6 (3)	Type B Tests at Power	GE	Response provided 3/28/91.
6.2.6 (4)	Air Lock Seal Testing	GE	Response provided 3/28/91.
6.2.6 (5)	Penetrations	GE	Response provided 3/28/91.
6.2.6 (7)	ECCS Isolation Valve Test Type C	GE	Response provided 3/28/91.
6.2.6 (8)a	List of CIVs for C Testing	GE	Response provided 3/28/91.
6.2.6 (8)b	List of Valves Reverse Tested	GE	Response provided 3/28/91.
6.2.6 (8)c	Testing of valves with no 30-Day Seal	GE	Response provided 3/28/91.
6.2.6 (8)d	Containment Purge Isolation Time	NRC	NRC will consider further and will discuss with GE at a later time.
6.2.6 (9)	Secondary Containment Inleakage/ Bypass	GE	Response provided 3/28/91.
6.2.6 (10)	Hydrogen Recombiner System Effects on ILRT	GE	Response provided 3/28/91.
6.2.6 (11)	Control of Test, Vents, and Drains	GE	Response provided 3/28/91.
6.2.6 (12)	ESF System Leak Testing	GE	Response provided 3/28/91.
6.2.6 (13)	Type C Tests for Containment Boundary Lines	None	Resolved.

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Number	Subject	Action	Comments
	6.5.3 FISSICN PRODUCT CONTROL SYSTEMS & STRUCTURES		
6.5.3 A(1)	Supression Pool Scrubbing Factor	None	Resolved.
6.5.3 A(2)	Standby Gas Treatment Single Filter Train	GE	Response will be provided in May.
6.5.3 B (2)(1)	Single filter reliability, Availability	GE	See item 6.5.3 A(2) above.
6.5.3 B(2)(2)	SGTS Instrumentation	NRC	This item is to be revisited.
6.5.3 B(2)(3)	Effects of Routine Operational use of the SGTS on its Reliability and Availability for use During Post-Accident Conditions	GE	Response will be provided in May in conjunction with the item 6.5.3 A(2).
	6.5.1 ESF ATMOSPHERE CLEANUP SYSTEM		
6.5.1 (1)	Normal Air Handling System	NRC	Provided on Amendment 16. NRC will review.
6.5.1 CI(1)	Confirmatory Item (1) - Intake Design Capacity	GE	Response will be provided following Chap.15 reanalysis.
6.5.1 (2)	Fire Protection for CR ESF Filter System	None	Resolved.
6.5.1 (3)	ESF Components List	NRC	Provided on Amendment 16. NRC will review.
6.5.1 CI(2)	Redundancy of ESF Filter Trains for CR Intake	None	Resolved.

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Number	Subject	Action	Comments
	6.4 CONTROL ROOM HABITABILITY SYSTEMS		
6.4 (1)	Location of Makeup Air Inlets	GE	Resolved pending recalculating all of the Chapter 15 radiological accidents.
6.4 (2)	Protection from Confined Area Releases	None	Resolved.
6.4 (3)	Instrumentation	NRC	Provided on Amendment 16. NRC will review.
6.4 (4)	Positive Pressure in Control and Mechanical equipment Rooms	GE	Response provided 3/28/91.
6.4 CI(1)	Thickness of Charcoal Adsorber	GE	Response provided 3/28/91.
	15.7.3 LIQUID RADWASTE TANK FAILURE		
15.7.3	Design of Radwaste Substructure	GE	Response will be provided following Chap. 15 reanalysis.
	9.2.9 MAKEUP WATER SYSTEM (CONDENSATE)		
9.2.9 (1)	Automatic Switchover of Suction from CST to SP	NRC	NRC will review and evaluate.
9.2.9 (2)	Auto Switchover for SP Cleanup Pump Suction	GE	Response provided 3/28/91.
9.2.9 (3)	Analysis for Potential Flooding From Failure of MUWC System	GE	Response provided on page 9.2-16 attached.
9.2.9 (3)	Required CST Inventory	GE	Response provided on page 9.2-16 attached.

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Item Number	Subject	Action	Comments
	9.2.10 MAKEUP WATER WATER SYSTEM (PURIFIED)		
9.2.10 (1)	MUWP interface with SR Systems	GE	Response provided 3/28/91.
9.2.10 (4)	Demineralized Water Makeup and Storage Tank Capacities	GE	Response provided on page 9.2-13 attached.
9.2.10 (5b)	Water Supply Specifications	GE	Response provided 3/28/91.
9.2.10 (5c)	Applicant Scope	GE	Response provided on page 9.2-13 attached.
	9.2.11 REACTOR BUILDING COOLING WATER SYSTEM (RCW)		
9.2.11 (1)	Missile Protection	None	Resolved.
9.2.11 (2)	Heat Exchangers	GE	Response provided on pages 9.2-6 and 9.2-19.1 attached.
		NRC	Will reexamine the basis for need to address 4 hour shutdown heat load.
9.2.11 (3)	Sizing of Heat Exchangers	GE	Response provided on page 9.2-13 attached.
9.2.11 (4)	Four Hour Shutdown with Loss of AC Power	None	Resolved in Amendment 14.
9.2.11 (5)	Protection of RCW from HELB/MELB	GE	Response provided 3/28/91.
9.2.11 (7)	Service Water System Description and Interface with Sea Water	GE	Response provided on pages 9.2.11 and 9.2-25d attached. P&ID will be provided in May.

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Item Number	Subject	Action	Comments
	9.2.12 HVAC NORNAL COOLING WATER SYSTEM		
9.2.12 (1)	HVAC Isolation Valves Seismic Category		
	Part a: Secondary Containment Isolation Valves	None	Resolved.
	Part b: Seismic Category I Class. of Primary Containment Penetrations	GE	Response provided 9/28/91.
	Part c: Leakage Concerns	NRC	NRC will review.
9.2.12 (2)	Number of Chillers and Pumps in the System	GE	Response will be provided in May.
	9.2.13 HVAC EMERGENCY COOLING WATER SYSTEMS		
9.2.13	Missile Protection	GE	Response provided 3/28/91.
9.2.13 (2)	Protection from Water Hammer	GE	Response provided 3/28/91.
9.2.13 (5)	Chemical Feed Tank	GE	Response will be provided in May.
9.2.13 (6)	Number of Divisions and Associated Cooling for EDG	GE	Response will be provided in May.
9.2.13 (7)	Referenced Number of P&IDs	GE	Response will be provided in May.
9.2.13 (8)	Pressure and Functional Testing	GE	Response provided 3/28/91.

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Item Number	Subject	Action	Comments
	10.2 TURBINE GENERATOR*		
10.2 (1)	Periodic Tests of Turbine Valves	GE	Response provided on page 10.2-8 attached.
	10.3 MAIN STEAM SUPPLY SYSTEM		
10.3 (1)	Main Steam Line Classification	GE	GE is still dicsussing with Mechanical Engineering Branch.
	10.4.2 MAIN CONDENSER EVAUATION SYSTEM		
10.4.2 (1)	Radiation Monitoring of Exhaust	GE	Response provided on page 10.4-5 attached.
	104.3 TURBINE GLAND SEAL SYSTEM		
10.4.3 (1)	Local Exhaust Radiation Monitoring	GE	Response provided on pages 10.4-6 and 10.4-7 attached.
10.4.3 (2)	Interface Regarding the Switch-over to Auxiliary Steam Supply	GE	Response provided on page 10.4-17 attached.
	10.4.4 TURBINE BYPASS SYSTEM		
10.4.4 (1)	Turbine Bypass Valves	GE	Provided in Amendment 16.
	10.4.5 CIRCULATING WATER SYSTEM		
10.4.5 (1)	CWS SSAR Table Reference	GE	Provided in Amendment 15.
10.4.5 (2)	Flooding Protection	GE	Response provided on page 10.4-10 attached.

* Several clarifying changes were made to Chapter 10 in addition to responding to the open items.

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Item Number	Subject	Action	Comments
	10.4.7 <u>CONDENSATE AND FEEDWATER SYSTEM</u>		
10.4.7 (1)	Size of Feedwater Line	GE	Resolved.
10.4.7 (2)	Power Source for Motor Operated Gate Valve.	NRC	NRC will provide further guidance.
10.4.7 (3)	CFS Seismic Category and Group Classifications	NRC	Amendment 14 addressed this item. NRC will review.
	9.1.3 <u>SPENT FUEL POOL COOLING AND CLEANUP SYSTEM</u>*		
9.1.3 (1)	Isolation of FPC from Suppression Pool Cleanup System	GE	Response provided on page 9.1-5 attached.
9.1.3 (2)	Emergency Source of SPF Water	NRC	NRC will evaluate GE's response to previous questions. (See GE response on page 9.1-5 attached)
9.1.3 (3)	FPC Design Seismic Classification	NRC	NRC will reevaluate previous similar information.
9.1.3 (4)	FPC Design-single Active Failure, LOOP Sizing of Heat Exchangers	GE	Response provided on page 9.1-5 attached.
9.1.3 (5)	Provision of a FPC System Components Description Table	GE	Response provided on page 9.1-3 attached.
	9.1.5 <u>OVERHEAD HEAVY LOAD HANDLING SYSTEM</u>		
9.1.5 (A.1)	OHLHS Design/RG 1.29 and RG 1.13	GE	See Attachment 1.
9.1.5 (A.2)	Non-Seismic Category I Load Handling Equipment	GE	See Attachment 1.
9.1.5 (A.3)	Refueling Bridge Crane References and Seismic Classifications	GE	See Attachemnt 1.
9.1.5 (A.4)	Housing of Load Handling Equipment for Steam Tunnel Servicing	GE	See Attachment 1.
9.1.5 (B.1)	Spent Fuel Crane Lifting Height	GE	See Attachment 1.

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*Several clarifying changes were made to Section 9.1.3 in addition to responding to the open items.

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Item Number	Subject	Action	Comments
9.1.5 (B.2)	Control of Heavy Load Movement Over Spent Fuel Pool	GE	See Attachment 1.
9.1.5 (B.3)	Protection of Safety-Related Equipment During Heavy Load Oper.	GE	See Attachment 1.
9.1.5 (C.1)	Additional Details Concerning Hoists	GE	See Attachment 1.
9.1.5 (C.2)	Single-Failure Criteria for Hoists and lifting Devices	GE	See Attachment 1.
9.1.5 (C.3)	Limit and Safety Devices and OHLHS FMEA		
	- Limit and Safety Devices Portion	GE	See Attachment 1.
	- OHLHS FMEA Portion	GE	See Attachment 2.
9.1.5 (C.4)	Heavy Load Operations in the Control Building	GE	See Attachment 1.
	11.3.1 GASEOUS WASTE MANAGEMENT		
11.3.1 (1)	Monitoring of the Exhaust from the Turbine Building	GE	See response to item 10.3(1).
11.3.1 (2)	Sensitivity of Secondary Cont. Exhaust Monitoring	GE	Response provided on page 11.5-3 attached.
11.3.1 (3)	Relative Location of the Plant Release Point	NRC	Provided in Amendment 16. NRC will evaluate.
	11.4.1 SOLID WASTE MANAGEMENT SYSTEM		
11.4.1 (1)	Compliance with 10CFR61	GE	Response provided 3/28/91.
	11.4.2 EVALUATION FINDINGS		
11.4.2 (1)	Radwaste Storage Capacity	GE	Response provided 3/28/91.
11.4.2 (5)	Cement Glass as a Waste Solidification Agent	GE	Response provided 3/28/91.
11.4.2 (5)(1)	Incinerator Description	GE	Response provided 3/28/91.
11.4.2 (5)(2)	Inconsistencies in Addressing Estimated Waste Shipments	GE	Response will be provided in May.
11.4.2 (5)(3)	Illegible P&IDs	None	Resolved.

11.5.1 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEM*

11.5.1 (1.a)	Classification of Exhausts as Non-Radioactive	GE	Response provided on page 11.5-1 attached.
11.5.1 (1.b)	Direct Effluent Release Paths to the Environment	GE	Response provided on pagrs 11.5-1 and 11.5-8 attached.
11.5.1 (1.c)	Reactor Service Water Effluent Monitoring	GE	Response provided 3/28/91.
11.5.1 (2)	Continuous Monitors Channel Ranges and Sensitivities	GE	Response provided on pages 11.5-13,14,15,16,17,18 and 21 attached.
11.5.1 (3)	RB Fuel Area Vent Exhaust Monitoring	GE	See response to 11.3.1(2).
11.5.1 (4)	Plant Vent Exhaust Sampling	GE	Response provided on pages 11.5-20 and 11.5-22 attached.
11.5.1 (5)	Design and Qualification of Accident Monitoring Instrumentation	GE/NRC	Both NRC and GE will review the adequacy of the info. provided in Amendment 16.

11.2 LIQUID WASTE MANAGEMENT SYSTEM

(This additional open item was not discussed during the meeting)

	Local Alarm Capability for the Condensate Storage Tank	GE	Response provided on next page.

* Several clarifying changes were made to Section 11.5 in addition to responding to the open items.

11.2 Liquid Waste Management System - Additional Open Item

Discussion of condensate storage tank overflow is provided in Response to Question 430.156. Local display of the condensate storage tank water level is not provided since there are no operator functions (such as starting or stopping of pumps) at the condensate storage tank.

ATTACHMENT 1

9.1.5 Overhead Heavy Load Handling Systems

9.1.5.1 Design Bases

The equipment covered by this subsection handle items considered as heavy loads that are handled under conditions that mandate critical load handling compliance.

Critical load handling conditions include loads, equipment, and operations, which if inadvertent operations or equipment malfunctions either separately or in combination, could cause; (1) a release of radioactivity, (2) a criticality accident, (3) the inability to cool fuel within reactor vessel or spent fuel pool or (4) prevent safe shutdown of the reactor. This includes risk assessments to spent fuel and storage pool water levels, cooling of fuel pool water, new fuel criticality. This includes all components and equipment used in moving any load weighing more than one fuel assembly including the weight of its associated handling devices (i.e., one ton).

The reactor building crane as designed shall provide a safe and effective means for transporting heavy loads including the handling of new and spent fuel, plant equipment and service tools. Safe handling includes design considerations for maintaining occupational radiation exposures as low as practicable during transportation and handling.

Where applicable, the appropriate seismic category, safety class quality group, ASME, ANSI, industrial and electrical codes have been identified (see Table 3.2-1 and Table 9.1-6). The designs will conform to the relevant requirements of General Design Criterion 2, 4 and 61 of 10CFR Part 50 Appendix A.

The lifting capacity of each crane or hoist is designed to at least the maximum actual or anticipated weight of equipment and handling devices in a given area serviced. The hoists, cranes, or other lifting devices shall comply with requirements of ANSI N14.6, ANSI B30.9, ANSI B30.10 and NUREG-0612 Subsection 5.1.1(4) or 5.1.1(5). Cranes and hoists are also designed to the criteria and guidelines of NUREG-0612 Subsection 5.1.1(7), ANSI B30.2 and CMAA-70 specifications for electrical overhead traveling cranes, including ANSI B30.11, ANSI B30.16, and NUREG-0554 as applicable.

9.1.5.2 System Description

9.1.5.2.1 Reactor Building Crane

The Reactor Building is a reinforced concrete structure which encloses the reinforced concrete containment vessel (RCCV), the refueling floor, new fuel storage vault, the storage pools for spent-fuel and the dryer and separator and other equipment. The reactor building crane provides heavy load lifting capability for the refueling floor. The main hook (150 ton capacity) will be used to lift the concrete shield blocks, drywell head, reactor pressure vessel (RPV) head insulation, RPV head, dryer/separator strong back, RPV head strongback/carousel, new fuel shipping containers, and spent fuel shipping cask. The orderly placement and movement paths of these components by the reactor building main crane precludes transport of these heavy loads over the spent fuel storage pool or over the new fuel storage vault.

The RB crane will be used during refueling/servicing as well as when the plant is online. During refueling/servicing, the crane handles the shield plugs, drywell and

reactor vessel heads, steam dryer and separators, etc. (see Table 9.1-7). Minimum crane coverage must include RB refueling floor laydown areas, and RB equipment storage pit. During normal plant operation the crane will be used to handle new fuel shipping containers and the spent fuel shipping casks. Minimum crane coverage must include the new fuel vault, the RB equipment hatches, and the spent fuel cask loading and washdown pits. A description of the refueling procedure can be found in section 9.1.4.

The RB crane will be interlocked to prevent movement of heavy loads over the spent fuel storage portion of the spent fuel storage pool. Since the crane is used for handling large heavy objects over the open reactor the crane is of type I design. The reactor building crane shall be designed to meet the single-failure-proof requirements of NUREG-0554.

9.1.5.2.2 Other Overhead Load Handling Systems

9.1.5.2.2.1 Upper Drywell Servicing Equipment

The upper drywell arrangement provides servicing access for the main steam isolation valves (MSIVs), feedwater isolation valves, safety relief valves (SRVs), Emergency Core Cooling Systems (ECCSs) isolation valves, and drywell cooling coils, fans and motors. Access to the space is via the RB through either the upper drywell personnel lock or equipment hatch. All equipment is removed through the upper drywell equipment hatch. Platforms are provided for servicing the Feedwater and mainsteam isolation valves, safety relief valves, and drywell cooling equipment with the object of reducing maintenance time and operator exposure. The MSIVs, SRVs, and feedwater isolation valves all weigh in excess of 2000 kg. Thus are considered heavy loads.

With maintenance activity only being performed during a refueling outage, only safe shutdown ECCS piping and valves need be protected from any inadvertent load drops. Since only one division of ECCS is required to maintain the safe shutdown condition and the ECCS divisions are spatially separated, an inadvertent load drop that breaks more than one division of ECCS is not credible. Inaddition, two levels of piping support structures and equipment platforms separate and shield the ECCS piping from the heavy loads transport path.

This protection is adequate such that no credible load drop can cause either (1) a release of radioactivity, (2) a criticality accident, or (3) the inability to cool fuel within reactor vessel or spent fuel pool; therefore, the upper drywell servicing equipment is not subject to the requirements of subsection 9.1.5.

9.1.5.2.2.2 Lower Drywell Servicing Equipment

The lower drywell (L/D) arrangement provides for servicing, handling and transportation operations for RIP, and FMCRD. The lower drywell OHLHS consists of a rotating equipment service platform, chain hoists, FMCRD removal machine, a RIP removal machine, and other special purpose tools.

The rotating equipment platform provides a work surface under the reactor vessel to support the weight of personnel, tools, and equipment and to facilitate transportation moves and heavy load handling operations. The platform rotates 360° in either direction from its stored or "idle" position. The platform is designed to accommodate the maximum weight of the accumulation of tools and equipment

plus a maximum sized crew. Weights of tools and equipment are specified in the interface control drawings for the equipment used in the lower drywell. Special hoists are provided in the lower drywell and reactor building to facilitate handling of these loads.

(1) Reactor Internal Pump Servicing

There are 10 RIPs and their supporting instrumentation and heat exchangers in the L/D that require servicing. The facilities provided for servicing the RIPs include:

(a) L/D equipment platform with facilities to rotate the motor from vertical to horizontal and place it on a cart for direct pull out to the RB. The equipment platform rotates to facilitate alignment with the installed pump locations.

(b) Attachments points for rigging the RIP heat exchanger into place. The RIP heat exchanger can be lowered straight down to the equipment platform.

(c) Access to the RIP equipment platform is via stairs. There is a ladder access to the RIP heat exchanger maintenance platform.

(d) The L/D equipment tunnel and hatch are utilized to remove the RIP motors from the lower drywell.

(e) The RIP motor servicing area is directly outside the L/D equipment hatch.

The 10 RIPs have wet induction motors in housings which protrude into the lower drywell from the RPV bottom head. These are in a circle at a radius of 3162.5 mm from the RPV centerline. For service, the motor is removed from below and outside, whereas the diffuser, impeller and shaft are removed from above and inside the RPV.

The motor, with its lower flange attached, weighs approximately 3300 kg, is 830 mm in diameter and 1925 mm high. The flange has "ears" that extend from two sides, 180° apart. These ears, which are used to handle the motor, increase the flange diameter to 1200 mm for a width of 270 mm.

The motor, suspended from jack screws, is lowered straight down out of its housing onto the equipment platform. The motor is then moved, circumferentially and lifted onto a rail mounted transport cart for direct removal through the equipment removal L/D equipment tunnel and hatch. The motor is transported horizontally out of the containment and into the motor service shop immediately adjacent to the L/D equipment hatch.

The RIP servicing equipment includes the cart to transport the motor from the service area through the equipment hatch to the L/D equipment platform. The interface for this equipment is the rails on the equipment platform that permit locating the motor below its nozzle on the RPV. The servicing equipment includes a chain hoist for rotating the RIP motor from horizontal to vertical and a hydraulic lift to raise it from the equipment platform to its installed position below the RPV. Facilities are provided for handling stud tensioners, blind flanges, other tools, drains and vents used in RIP servicing.

Servicing of the RIP heat exchanger, such as removal of the tube bundle, will be accomplished by rigging to attachment points on the RPV pedestal and structural steel in the area. A direct vertical removal path is provided from the heat exchanger installed position to the equipment platform. The operation is performed by a chain hoist. This is considered to be a nonroutine servicing operation.

These RIPs are serviced only when the reactor is in a safe shutdown mode. In addition, there is no safety-related equipment below either the RIPs or the RIP heat exchangers. Inadvertent load drops of either component can not cause either (1) a release of radioactivity, (2) a criticality accident, or (3) the inability to cool fuel within reactor vessel or spent fuel pool; therefore, the RIP servicing equipment is not subject to the requirements of subsection 9.1.5.

(2) Fine Motion Control Rod Drive Servicing

There are 205 FMCRDs in the L/D that require servicing. There are two types of servicing operations: replacement of the FMCRD drive mechanism and motor and seal replacement. Separate servicing equipment is provided for each of these operations.

(a) The FMCRD drive servicing machine has its own mechanisms for rotating and raising FMCRD drive assemblies from a carrier on the equipment platform to their installed position. This servicing machine interfaces with the L/D equipment platform, which permits positioning the servicing machine under any of the 205 FMCRDs.

(b) A separate machine and cart are provided for servicing FMCRD motors and seal assemblies and transporting them to the service shop located immediately outside the L/D equipment hatch.

There is no safety-related equipment below either component. Inadvertent load drops by the FMCRD servicing equipment can not cause either (1) a release of radioactivity, (2) a criticality accident, or (3) the inability to cool fuel within reactor vessel or spent fuel pool; therefore, the FMCRD servicing equipment is not subject to the requirements of subsection 9.1.5.

9.1.5.2.2.3 Mainsteam Tunnel Servicing Equipment

The mainsteam tunnel is a reinforced concrete structure that surrounds the mainsteam lines and feedwater lines. The safety-related valve area of the mainsteam tunnel is located inside the reactor building. Access to the mainsteam tunnel is during a refueling/servicing outage. At this time MSIVs or Feedwater Isolation valves and/or feedwater check valves may be removed using permanent overhead monorail type hoists. Transported by monorail out of the steam tunnel and placed on floor. Below a ceiling removal hatch. Valves are then lifted through ceiling hatch by valve service shop monorail. During shutdown, all of the piping and valves are not required to operate. Any load drop can only damage the other valves or piping within the main steam tunnel. Inadvertent load drops by the mainsteam tunnel sevicing equipment can not cause either (1) a release of radioactivity, (2) a criticality accident, or (3) the inability to cool fuel within reactor vessel or spent fuel pool; therefore, the mainsteam tunnel servicing equipment is not subject to the requirements of subsection 9.1.5.

9.1.5.2.2.4 Other Servicing Equipment

Outside the RCCV, the mainsteam tunnel, and the refueling floor no safety-related component of one division shall be routed over any portion of a safety-related portion of another division. The ABWR has three independent and separate ECCS divisions. A load drop accident in one division causing the complete loss of a second division is not credible. Hence inadvertent load drops can not cause either (1) a release of radioactivity, (2) a criticality accident, (3) the inability to cool fuel within reactor vessel or spent fuel pool, or (4) prevent the safe shutdown of the reactor; therefore, all servicing equipment located outside the RCCV, the mainsteam tunnel, or the refueling floor are not subject to the requirements of subsection 9.1.5.

9.1.5.3 Applicable Design Criteria For All OHLH Equipment

All handling equipment subject to heavy loads handling criteria will have ratings consistent with lifts required and the design loading will be visibly marked. Cranes/hoists or monorail hoists will pass over the centers of gravity of heavy equipment that is to be lifted. In locations where a single monorail or crane handles several pieces of equipment, the routing shall be such that each transported piece will pass clear of other parts. If, however, due to restricted overhead space the transported load cannot clear the installed equipment, then the monorail may be offset to provide transport clearance. A lifting eye offset in the ceiling over each piece of equipment can be used to provide a Y-lift so that the load can be lifted upward until free and then swung to position under the monorail for transport.

Pendant control is required for the bridge, trolley and the auxiliary hoist to provide efficient handling of fuel shipping containers during receipt and also to handle fuel during new fuel inspection. The crane control system will be selected considering the long lift required through the equipment hatch as well as the precise positioning requirements when handling the RPV and drywell heads, RPV internals, and the RPV head stud tensioner assembly. The control system will provide stepless regulated variable speed capability with high empty-hook speeds. Efficient handling of the drywell and RPV heads and stud tensioner assembly require that the control system provide spotting control. Since fuel shipping cask handling involves a long duration lift, low speed and spotting control, thermal protection features will be incorporated.

Heavy load equipment is also used to handle light loads and related fuel handling tasks. Therefore, much of the handling systems and related design, descriptions, operations, and service task information of Subsection 9.1.4 is applicable here. The cross reference between the handling operations/equipment and Subsection 9.1.4 is provided in Table 9.1-7. See Table 9.1-8 for a summary of heavy load operation.

Transportation routing drawings will be made covering the transportation route of every piece of heavy load removable equipment from its installed location to the appropriate service shop or building exit. Routes will be arranged to prevent congestion and to assure safety while permitting a free flow of equipment being serviced. The frequency of transportation and usage of route will be documented based on the predicted number of times usage either per year and/or per refueling or service outage.

Safe load paths/routing will comply with the requirements of NUREG-0612 Subsection 5.1.1(1).

9.1.5.4 Equipment Operating Procedures Maintenance and Service

Each item of equipment requiring servicing will be described on an interface control diagram (ICD) delineating the space around the equipment required for servicing. This will include pull space for internal parts, access for tools, handling equipment, and alignment requirements. The ICD will specify the weights of large removable parts, show the location of their centers of gravity, and describe installed lifting accommodations such as eyes and trunnions. An instruction manual will describe maintenance procedures for each piece of equipment to be handled for servicing. Each manual will contain suggestions for rigging and lifting of heavy parts and identify any special lifting or handling tools required.

All major handling equipment components: cranes, hoist, etc., will be provided with an operating instruction and maintenance manual for reference and utilization by operations personnel. Handling equipment operating procedure will comply with the requirements of NUREG-0612 Subsection 5.1.1(2).

The operational programs for maintenance and servicing are described in Subsection 9.1.5.6.

9.1.5.5 Safety Evaluations

The cranes, hoists, and related lifting devices used for handling heavy loads either satisfy the single-failure-proof guidelines of NUREG-0612 Subsection 5.1.6, including NUREG-0554 or evaluations are made to demonstrate compliance with the recommended guidelines of Section 5.1, including Subsection 5.1.4 and 5.1.5.

The equipment handling components over the fuel pool are designed to meet the single failure proof criteria to satisfy NUREG-0554. Redundant safety interlocks and limit switches are provided to prevent transporting heavy loads other than spent fuel by the refueling bride crane over any spent fuel that is stored in the spent fuel storage pool.

A transportation routing study will be made of all planned heavy load handling moves to evaluate and minimize safety risks.

Safety evaluation of related light loads and refueling handling tasks in which heavy load equipment is also used are covered in Subsection 9.1.4.3.

9.1.5.6 Inspection and Testing

Heavy load handling equipment is subject to the strict controls of Quality Assurance (QA), incorporating the requirements of Federal Regulation 10CFR50, Appendix B. Components defined as essential to safety have an additional set of engineering specified "Quality Requirements" that identify safety-related features which require specific QA verification of compliance to drawing/specification requirements.

Prior to shipment, every lifting equipment component requiring inspection will be reviewed by QA for compliance and that the required records are available. Qualification load and performance testing, including nondestructive examination (NDE) and dimensional inspection on heavy load handling equipment will be performed prior to QA acceptance. Tests may include load capacity, safety overloads, life cycle, sequence of operations and functional areas.

When equipment is received at the site it will be inspected to ensure no damage has occurred during transit or storage. Prior to use and at periodic intervals each piece of equipment will be tested again to ensure the electrical and/or mechanical functions are operational including visual and, if required, NDE inspection.

Crane inspections and testing will comply with requirements of ANSI B30.2 and NUREG-0612, Subsection 5.1.1(6).

9.1.5.7 Instrumentation Requirements

The majority of the heavy load handling equipment is manually operated and controlled by the operator's visual observations. This type of operation does not necessitate the need for a dynamic instrumentation system.

Load cells may be installed to provide automatic shutdown whenever threshold limits are exceeded for critical load handling operations to prevent overloading.

9.1.5.8 Operational Responsibilities

Critical heavy load handling in operation of the plant shall include the following documented programs for safe administration and safe implementation of operations and control of heavy load handling systems:

(1) Heavy Load Handling System and Equipment Operating Procedures.

(2) Heavy Load Handling Equipment Maintenance Procedures and/or Manuals.

(3) Heavy Load Handling Equipment Inspection and Test Plans; NDE, Visual, etc.

(4) Heavy Load Handling Safe Load Paths and Routing Plans.

(5) QA Program to Monitor and Assure Implementation and Compliance of Heavy Load Handling Operations and Controls.

(6) Operator Qualifications, Training and Control Program.

ATTACHMENT 2

RESPONSE TO OHLHS PORTION OF OPEN ITEM 9.1.5(c.3)

As discussed in Section 15B.1, FMEAs are provided for two ABWR systems and one major component which presents a significant change from past ABWR designs. Specifically, FMEAs are included in Appendix 15B for:

(1) control drive systems (with emphasis on the fine motion control rod drive),

(2) essential multiplexing system, and

(3) reactor internal pump.

Regulatory Guide 1.70 requires FMEAs to be performed on selected subsystems of Chapters 6,7 and 9. However, GE considers that the plant nuclear safety operational analysis (NSOA) of Appendix 15A and the probabilistic evaluations of Appendix 19d adequately address single failures for those systems and components which are similar to past BWR designs. Since the design of the ABWR OHLHS instrument and control system is similar to past designs, GE believes that it is unnecessary to perform a FMEA on the OHLHS instrument and control system.

3.4 WATER LEVEL (FLOOD) DESIGN

430.229 The types and methods used for protecting the ABWR safety-related structures, systems and components from external flooding shall conform to the guidelines defined in RG 1.102.

Criteria for the design basis for protection against external flooding shall conform to the requirements of RG 1.59. The design criteria for protection against the effects of compartment flooding shall conform to the requirements of ANSI/ANS-56.11. The design basis flood levels are specified in Table 3.4-1.

3.4.1 Flood Protection

430.229 This section discusses the flood protection measures that are applicable to the standard ABWR plant Seismic Category I structures, systems, and components for both external flooding and postulated flooding from plant component failures. These protection measures also apply to other structures that house systems and components important to safety which fall within the scope of plant specific.

3.4.1.1 Flood Protection Measures for Seismic Category I Structures

430.229 The safety-related systems and components of the ABWR Standard Plant are located in the reactor, control, and radwaste buildings which are seismic category 1 structures. These structures together with those identified in Table 3.4-1 are protected against external flood damage. Flood protection of safety-related systems and components is provided for all postulated design flood levels and conditions 430.229 described in Table 2.0-1. Postulated flooding from component failures in the building compartments does not adversely affect plant safety nor does it represent any hazard to the public.

430.229 Structures which house the safety-related equipment and offer flood protection are identified in Table 3.4-1. Descriptions of these structures are provided in Subsection 3.8.4 and 3.8.5. Exterior or access openings and penetrations that are below the design flood level are identified in Table 6.2-9.

3.4.1.1.1 Flood Protection from External Sources

Seismic Category I structures that may be affected by design basis floods are designed to withstand the floods postulated in Table 2.0-1 430.230 using the hardened protection approach with structural provisions with incorporated in the plant design to protect safety-related structures, systems, and components from postulated flooding. Seismic Category I structures required for safe shutdown remain accessible during all flood conditions.

Safety-related systems and components are flood-protected either because of their location above the design flood level or because they are enclosed in reinforced concrete Seismic Category I structures which have the following requirements:

(1) wall thicknesses below flood level of not less than two feet;

(2) water stops provided in all construction joints below flood level;

(3) watertight doors and equipment hatches installed below design flood level; and

(4) waterproof coating of external surfaces. 430.232

Waterproofing of foundations and walls of Seismic Category I structures below grade is accomplished principally by the use of water stops at expansion and construction joints. In addition to water stops, waterproofing of the plant structures that house safety related systems and components is provided up to 8 mm (3 in) above the plant ground level to protect the external surfaces from exposure to water. 430.232

CWm

INSERT 3.4.1.1.1 9.2.9(3)

Additional specific provisions for flood protection include administrative procedures to assure that all watertight doors and hatch covers are locked in the event of a flood warning. If local seepage occurs through the walls, it is controlled by sumps and sump pumps.

In the event of a flood, flood levels take a relatively long time to develop. This allows

INSERT 3.4.1.1.1

9.2.9(3)

The flood protection measures that are described above also guard against flooding from on-site storage tanks that may rupture. The largest is the condensate storage tank that has a capacity of 2,110 cubic meters. This tank is constructed form stainless steel and is located between the turbine building and the radwaste building where there are no direct entries to these buildings. All plant entries start one foot above grade. Any flash flooding that may result from tank rupture will drain away from the site and cause no damage to site equipment.

requirement for redundant separation is met. Other redundant divisions are available for safe shutdown of the plant and no further evaluation is performed.

(4) If damage could occur to more than one division of a redundant essential system within 30 ft of any high energy piping, other protection in the form of barriers, shields, or enclosures is used. These methods of protection are discussed in Subsection 3.6.1.3.2.3. Pipe whip restraints as discussed in Subsection 3.6.1.3.2.4 are used if protection from whipping pipe is not possible by barriers and shields.

3.6.1.3.2.3 Barriers, Shields, and Enclosures

Protection requirements are met through the protection afforded by the walls, floors, columns, abutments, and foundations in many cases. Where adequate protection is not already present due to spatial separation or existing plant features, additional barriers, deflectors, or shields are identified as necessary to meet the functional protection requirements.

Barriers or shields that are identified as necessary by the use of specific break locations in the drywell and steam tunnel are designed for the specific loads associated with the particular break location.

INSERT 3.6.1.3.2.3 3.6.1 HABIT-IBILITY PORTION

Barriers or shields that are identified as necessary by the HELSA evaluation (i.e., based on no specific break locations), are designed for worst-case loads. The closest high-energy pipe location and resultant loads are used to size the barriers.

3.6.1.3.2.4 Pipe Whip Restraints

Pipe whip restraints are used where pipe break protection requirements could not be satisfied using spatial separation, barriers, shields, or enclosures alone. Restraints are located based on the specific break locations determined in accordance with Subsections 3.6.2.1.4.3 and 3.6.2.1.4.4. After the restraints are located, the piping and essential systems are evaluated for jet impingement and pipe whip. For those cases where jet impingement damage could still occur, barriers, shields, or enclosures are utilized.

The design criteria for restraints is given in Subsection 3.6.2.3.3.

3.6.1.3.3 Specific Protection Measures

(1) Nonessential systems and system components are not required for the safe shutdown of the reactor, nor are they required for the limitation of the offsite release in the event of a pipe rupture. However, while none of this equipment is needed during or following a pipe break event, pipe whip protection is considered where a resulting failure of a nonessential system or component could initiate or escalate the pipe break event in an essential system or component, or in another nonessential system whose failure could affect an essential system.

(2) For high energy piping systems penetrating through the containment, isolation valves are located as close to the containment as possible.

(3) The pressure, water level, and flow sensor instrumentation for those essential systems, which are required to function following a pipe rupture, are protected.

(4) High-energy fluid system pipe whip restraints and protective measures are designed so that a postulated break in one pipe could not, in turn, lead to a rupture of other nearby pipes or components if the secondary rupture could result in consequences that would be considered unacceptable for the initial postulated break.

(5) For any postulated pipe rupture, the structural integrity of the containment structure is maintained. In addition, for those postulated ruptures classified as a loss of reactor coolant, the design leak tightness of the containment fission product barrier is maintained.

(6) Safety/relief valves (SRV) and the reactor core isolation cooling (RCIC) system steamline are located and restrained so that a pipe failure would not prevent depressurization.

INSERT 3.6.1.3.2.3

Barriers or shields that are identified as necessary by the use of specific break locations in the drywell are designed for the specific loads associated with the particular break location.

The steam tunnel is made of reinforced concrete 2m thick. A steam tunnel subcompartment analysis was performed for the postulated rupture of a mainsteam line and for a feedwater line (see subsection 6.2.3.3.1). The peak pressure from a mainsteam line break was found to be 11psig. The peak pressure from a feedwater line break was found to be 3.9psig. The steam tunnel is designed for the effects of an SSE coincident with a high energy line break inside the steam tunnel. Under this conservative load combination, no failure in any portion of the steam tunnel was found to occur; therefore, a high energy line break inside the steam tunnel will not effect control room habitability.

The MSIVs and the feedwater isolation and check valves being inside the tunnel shall be designed for the effects of a line break. The details of how the MSIV and feedwater isolation and check valves functional capabilities are protected against the effects of these postulated pipe failures will be provided by the applicant referencing the ABWR design (see subsection 3.6.4.1, items 4 and 6).

including deadweight and SSE (inertial) components.

Shielded Metal Arc (SMAW) and Submerged Arc (SAW) Welds:

The flow stress used to construct the master curve is 51 ksi

The value of SI used to enter the master curve for SMAW and SAW is

$$SI = M (P_m + P_b + P_e) Z \quad (8)$$

where

P_b = the combined primary bending stress, including deadweight and seismic components.

P_e = combined expansion stress at normal operation.

$Z = 1.15 [1.0 + 0.013 (OD-4)]$ for SMAW, (9)

$Z = 1.30 [1.0 + 0.010 (OD-4)]$ for SAW, (10)

and

OD = pipe outer diameter in inches.

When the allowable flaw length is determined from the master curve at the appropriate SI value, it can be used to determine if the required margins on load and flaw size are met using the following procedure.

For the method of load combination described in item (5), let M = 1.4, and if the allowable flaw length from the master curve is at least equal to the leakage size flaw, then the margin on load is met.

3.6.4 Interfaces

3.6.4.1 Details of Pipe Break Analysis Results and Protection Methods

The following shall be provided by the applicant referencing the ABWR design (See Subsection 3.6.2.5):

(1) A summary of the dynamic analyses applicable to high-energy piping systems in accordance with Subsection 3.6.2.5 of Regulatory Guide 1.70. This shall include:

(a) Sketches of applicable piping systems showing the location, size and orientation of postulated pipe breaks and the location of pipe whip restraints and jet impingement barriers.

(b) A summary of the data developed to select postulated break locations including calculated stress intensities, cumulative usage factors and stress ranges as delineated in BTP MEB 3-1.

(2) For failure in the moderate-energy piping systems listed in Table 3.6-6, descriptions showing how safety-related systems are protected from the resulting jets, flooding and other adverse environmental effects. 410.21

(3) Identification of protective measures provided against the effects of postulated pipe failures in each of the systems listed in Tables 3.6-1, 3.6-2 and 3.6-4. 410.22

(4) The details of how the MSIV functional capability is protected against the effects of postulated pipe failures. 410.26

(5) Typical examples, if any, where protection for safety-related systems and components against the dynamic effects of pipe failures include their enclosure in suitably designed structures or compartments (including any additional drainage system or equipment environmental qualification needs). 410.28

3.6.4.2 Leak-Before-Break Analysis Report

INSERT 3.6.4.1 3.6.1 HABITABILITY PORTION

As required by Reference 1, an LBB analysis report shall be prepared for the piping systems proposed for inclusion from the analyses for the dynamic effects due to their failure. The report shall include only the piping stress analysis

INSERT 3.6.4.1

(6) The details of how the feedwater line check and feedwater isolation valves functional capabilities are protected against the effects of postulated pipe failures.

forms a part of the secondary containment boundary is designed to at least Seismic Category I and ASME Section III, Class 3 requirements. Some lines have no special isolation provisions and are not ASME Section III or Seismic Category I if an analysis shows that exfiltration would not occur in the event of failure of that pipe [i.e., the -1/4 in. water gage pressure differential would be maintained].

For architectural openings the inleakage is based on -1/4 in. water gage pressure differential. All doors have a vestibule with a second (outer) door. HVAC and electrical penetrations are designed to minimize leaks, and HVAC system is designed and tested for isolation under accident conditions.

430.34 Table 6.2-9 provides a listing of secondary containment openings. All piping and cabletray penetrations will be sealed with a sealing compond for leakage and fire protection. All doors are vestibule type with card reader access security systems that are monitored (See Subsection 13.6.3.4). The HVAC penetrations are designed to close on a design basis accident (See Subsection 9.4.3 on reactor building HVAC). Testing procedure and frequency can be found in the plant technical specifications.

6.2.3.3 Design Evaluation

The design of the secondary containment boundaries is described in the preceding subsection. Evaluation of this design, such that all regulatory requirements are met, are given in the following subsections:

(1) 6.5.1 Standby Gas Treatment System

(2) 9.4.5 Reactor Building HVAC System

6.2.3.3.1 Compartment Pressurization

6.2.3.3.1.1 Design Bases

The design of the secondary containment compartments with respect to pressurization is based upon the worst-case DBA rupture of a high or moderate energy line postulated to occur in each compartment (see Subsection 3.6.2 for rupture details). The rupture producing the greatest blowdown mass and enthalpy is selected for the analysis of each compartment. For the room design, the peak differential pressures are not to exceed the design differential pressure.

INSERT 6.2.3.3.1 (A)

INSERT 6.2.3.3.1 (B)

6.2.3.3.1.2 Design Features

The following paragraphs are brief descriptions of the compartments analyzed for pressurization. A more detailed description will be found in Subsection 3.8.1. Figure 6.2-37 shows the schematic layout of the secondary containment compartments with the interconnected vent paths and blowout panels. Figures 6.2-28 through 6.2-36 are the plan and elevation drawings showing component and equipment locations and configurations. Tables 6.2-3 and 6.2-4 tabulate the compartment free volumes and initial room conditions, flow path parameters and blowout panel characteristics. INSERT 6.2.3.3.1 (C)

6.2.3.3.1.2.1 Reactor Core Isolation Cooling (RCIC) Compartment

The RCIC compartment is located in the secondary containment at El(-)13200 mm. The design basis break for the RCIC compartment is the double-ended break of the 6-in. RCIC steam supply line. This line is a high energy line out to the normally closed isolation valve inside the RCIC compartment, ~~and~~ it supplies high energy steam to the RCIC turbine in the event of reactor vessel isolation. INSERT 6.2.3.3.1 (D)

6.2.3.3.1.2.2 Reactor Water Cleanup (RWCU) Equipment and Valves Rooms

The RWCU equipment (pump, heat exchanger, and filter/demineralizer) and valves rooms are located in the 0^{o}-270^{o} quadrant of the reactor building. The floor elevations are from (-)13200 mm to (-)200 mm with separate rooms for the equipment and valves. High energy piping connects the equipment and valve rooms and is routed to the steam tunnel and the primary containment vessel through special pipe chases. INSERT 6.2.3.3.1 (E)

6.2.3.3.1.2.3 Main Steam Tunnel

The reactor building main steam tunnel is located between the primary containment vessel and the turbine building. The steam tunnel houses the high energy and radioactive main steam and feedwater lines along with some portions of the RCIC, RHR, and RWCU piping.

DBA PORTION OF 3.6.1(1)

No. 9 The DBA for the steam tunnel is the double-ended break of one of the 28-in. main steam lines. which is routed from the reactor vessel to the turbine building. The steam tunnel blowout panels vent into the turbine building in the event of the postulated DBA. ← INSERT 6.2.3.3.1 F

6.2.3.3.1.3 Design Evaluation

~~Blowout panels are used in place of open~~

→ INSERT 6.2.3.3.1 G

DBA PORTION OF 3.6.1(1)

blowout panels are used in place of open vent pathways when the environmental conditions of one compartment must be isolated from the environment in another compartment. The panels are designed to open upon a differential pressure of 0.25 psid. ~~and are assumed to be fully opened after 0.1 sec. following their release.~~

0.50

The RELAP4 computer program is used to calculate the mass and energy release rates and the resultant compartment pressures and temperatures. A detailed discussion of the methodology and assumptions used in the program can be found in Subsection 6.2.7 of Reference 4. The initial conditions for the analysis include the assumption of 102% rated reactor power and the compartment pressures, temperatures and relative humidity to maximize the mass and energy release rates.

6.2.3.4 Tests and Inspections

Testing and inspection of the integrity of secondary containment will be made as part of the testing of the STGS (Subsection 6.5.1).

Status lights and alarms for door opening of secondary containment will be tested periodically by their operation, with observation of lights and alarms. Leakage testing and inspection of all other architectural openings will be made as they are utilized periodically.

6.2.3.5 Instrumentation Requirements

By their nature, electrical penetrations of secondary containment do not have any instrumentation requirements. Piping and HVAC penetrations instrumentation requirements are discussed as part of each system's description in this SAR. Details of the initiating signals for isolation are given in Subsection 7.3.1.1.10.

Certain doors are fitted with status indication lights.

6.2.4 Containment Isolation System

The primary objective of the containment isolation system is to provide protection against releases of radioactive materials to the environment as a result of accidents occurring in the systems inside the containment. The objective is accomplished by isolation of lines or ducts that penetrate the containment vessel. Actuation of the containment isolation system is automatically initiated at specific limits defined for reactor plant operation. After the isolation function is initiated, it goes through to completion.

6.2.4.1 Design Bases

6.2.4.1.1 Safety Design Bases

(1) Containment isolation valves provide the necessary isolation of the containment in the event of accidents or other conditions and prevent the unfiltered release of containment contents that cannot be permitted by 10CFR50 or 10CFR100 limits. Leaktightness of the valves shall be verified by Type C test.

(2) Capability for rapid closure or isolation of all pipes or ducts that penetrate the containment is provided by means that provide a containment barrier in such pipes or ducts sufficient to maintain leakage within permissible limits.

(3) The design of isolation valving for lines penetrating the containment follows the requirements of General Design Criteria 54 through 57 to the greatest extent practicable consistent with safety and reliability.

(4) Isolation valves for instrument lines that penetrate the drywell/containment conforms to the requirements of Regulatory Guide 1.11.

(5) Isolation valves, actuators and controls are protected against loss of their safety function from missiles and postulated effects of high- and moderate energy line ruptures.

(6) Design of the containment isolation valves and associated piping and penetrations meets the requirements for Seismic Category I components.

(7) Containment isolation valves and associated

DBA PORTION OF 3.6.1(1)

6.2.7 References

1. W.J. Bilanin, *The G.E. Mark III Pressure Suppression Containment Analytical Model*, June 1974, (NEDO-20533).

2. F.J. Moody, *Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels*, General Electric Company, Report No. NEDO-21052, September, 1975.

3. W.J. Bilanin, *The G.E. Mark III Pressure Suppression Containment Analytical Model*, Supplement 1, September 1975 (NEDO-20533-1).

4. Idaho National Engineering Laboratory, *RELAP4/MOD5--A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems, User's Manual*, September, 1976 (NUREG-1335).

← INSERT 6.2.7

INSERTS FOR
DBA PORTION OF 3.6.1(1)

(A) , in conjunction with a worst case single active component failure,

(B) For this analysis, a worst case single active component failure is defined as the failure to close of an isolation valve which separates the reactor pressure vessel from the high or moderate energy pipe break in the secondary containment.

(C) Table 6.2-3 tabulates the free volumes, initial environment conditions and DBA break characteristics for the compartments which are analyzed. Table 6.2-4 enumerates the flow path and blowout panel characteristics.

(D) In the event of a postulated design basis high energy line break, the steam/air mixture is directed into adjoining compartments and is eventually purged into the steam tunnel.

INSERTS FOR 6.2.3.3.1

(E) The design basis break for the RWCU system compartment network is either an 8-in or 6-in double-ended break of the water supply line. This depends upon which break diameter produces the maximum pressurization in the break compartment and any adjoining compartments. This high energy piping, which connects the RWCU equipment , originates at the reactor pressure vessel. After being routed through the RWCU system the high energy line is directed back to the reactor pressure vessel through special pipe chases and the steam tunnel. In the event of a postulated design basis high energy line break, the steam/air mixture is directed into adjoining compartments and eventually purged into the steam tunnel.

(F) These lines originate at the reactor pressure vessel and are routed through the main steam tunnel to the turbine building. In the event of a postulated design basis high energy line break, the pressurized steam/air mixture is held up in the main steam tunnel and purged into the turbine building through blowout panels, as required.

(G) The blowdown mass and enthalpy release rates for the high energy line breaks are determined using Moody's homogeneous equilibrium model. A discussion of the methodology and assumptions used in this model can be found in Reference 2. The resulting compartment pressures and temperatures are calculated by the engineering computer program SCAM. A detailed discussion of the methodology and assumptions used in this program can be found in Reference 4.

INSERT 6.2.7

J.P. Dougherty, SCAM - "Subcompartment Analysis Method," January 1977, (NEDE-21526).

9.1.3 Fuel Pool Cooling and Cleanup System

9.1.3.1 Design Bases

The fuel pool cooling and cleanup (FPC) system shall be designed to remove the decay heat from the fuel pool, maintain pool water level and quality and remove radioactive materials from the pool to minimize the release of radioactivity to the environs.

The FPC system shall:

(1) minimize corrosion product buildup and shall control water clarity, so that the fuel assemblies can be efficiently handled underwater;

(2) minimize fission product concentration in the water which could be released from the pool to the reactor building environment;

(3) monitor fuel pool water level and maintain a water level above the fuel sufficient to provide shielding for normal building occupancy;

(4) maintain the pool water temperature below 125°F under normal operating conditions. The temperature limit of 125°F is set to establish an acceptable environment for personnel working in the vicinity of the fuel pool. The design basis normal heat load from spent fuel stored in the pool is the sum of decay heat of the most recent 35% batch plus the heat from the previous 4 fuel batches. The RHR system will be used to supplement the FPC system under the maximum load condition as defined in Subsection 9.1.3.2.

for the FPC system is to provide cooling after closure of the fuel gates (21 days) at the

[CLARIFICATION]

[CLARIFICATION] after closure of the fuel gates.

9.1.3.2 System Description

9.1.3 (5)

and Table 9.1-11

The FPC system (Figures 9.1-1a and b, and 9.1-2) maintains the spent fuel storage pool below the desired temperature at an acceptable radiation level and at a degree of clarity necessary to transfer and service the fuel bundles.

The FPC system cools the fuel storage pool by transferring the spent fuel decay heat through two 6.55 x 10^6 Btu/hr heat exchangers to the reactor building closed cooling water system (RCW). Each of the two heat exchangers is designed to transfer one half the system design heat load. The system utilizes two parallel 250 m^3/hr pumps to provide a system design flow of 500 m^3/hr. Each pump is suitable for continuous duty operation. The equipment is located in the reactor building.

The system pool water temperature is maintained at or below 125°F. The decay heat released from the stored fuel is transferred to the RCW. The residual heat removal system (RHR) can supplement the FPC system to remove the additional heat generated should the reactor be defueled beyond the design-basis 35% batch.

Fuel storage pool water is circulated by means of overflow through skimmers around the periphery of the pool and a scupper at the end of the transfer pool. The overflow is collected in the fuel pool drain tanks and the flow passes through the heat exchangers and filter-demineralizers and back to the pool through the diffusers.

INSERT 9.1.3.2 [CLARIFICATION]

Clarity and purity of the pool water are maintained by a combination of filtering and ion exchange. The filter-demineralizers maintain total corrosion product metals at 30 ppb or less with pH range of 5.6 to 8.6 at 25°F for compatibility with fuel storage racks and other equipments. Conductivity is maintained at less than 1.2 μS/cm at 25°C and chlorides less than 20 ppb. Each filter unit in the filter-demineralizer subsystem has adequate capacity to maintain the desired purity level of the pools under normal operating conditions. The flow rate is designed to be approximately that required for two complete water changes per day for the fuel transfer and storage pools. The maximum system flow rate is twice that needed to maintain the specified water quality. 281.12 281.11

The FPC system is designed to remove suspended or dissolved impurities from the following sources:

(1) dust or other airborne particles;

INSERT 9.1.3.2

[CLARIFICATION]

During refueling prior to 21 days following shutdown, the reactor (shutdown cooling) and fuel pool cooling are provided jointly by the Residual Heat Removal (RHR) and FPC systems in parallel. The reactor cavity communicates with the fuel pool since the reactor well is flooded and the fuel gates are open. RHR suction is taken from the vessel shutdown suction lines, pumped through RHR heat exchangers and discharged into the upper pools to improve water clarity for refueling. For the FPC system, fuel pool water is circulated by means of overflow through skimmers around the periphery of the pool and a scupper at the end of the transfer pool drain tanks, pumped through the FPC heat exchangers and filter-demineralizers and back to the pool through the pool diffusers.

After 21 days, the fuel gates are closed. At this point FPC system provides solely the fuel pool cooling function. However, when the reactor is defueled more than the design-basis 35% batch (maximum heat load condition), RHR can provide supplemental cooling to remove additional decay heat. RHR supplemental cooling suction is taken from the skimmer surge tank, passed through RHR heat exchanger and back to the fuel pool.

(2) surface dirt dislodged from equipment immersed in the pool;

(3) crud and fission products emanating from the reactor or fuel bundles during refueling;

(4) debris from inspection or disposal operations; and

(5) residual cleaning chemicals or flush water.

A post-strainer in the effluent stream of the filter-demineralizer limits the migration of filter material. The filter-holding element can withstand a differential pressure greater than the developed pump head for the system.

The filter-demineralizer units are located separately in shielded cells with enough clearance to permit removing filter elements from the vessels.

Each cell contains only the filter-demineralizer and piping. All valves (inlet, outlet, recycle, vent, drain, etc.) are located on the outside of one shielding wall of the room, together with necessary piping and headers, instrument elements and controls. Penetrations through shielding walls are located so as not to compromise radiation shielding requirements.

The filter-demineralizers are controlled from a local panel. A differential pressure and conductivity instruments provided for each filter-demineralizer unit indicate when backwash is required. Suitable alarms, differential pressure indicators and flow indicators monitor the condition of the filter-demineralizers.

System instrumentation is provided for both automatic and remote-manual operations. A low-low level switch stops the circulating pumps when the fuel pool drain tank reserve capacity is reduced to the volume that can be pumped in approximately one minute with one pump at rated capacity (250 m^3/hr). A level switch is provided in the fuel pool to alarm on high and low level. A temperature element is provided to display pool temperature in the main control room.

The circulating pumps are controlled from the control room and a local panel. Pump low suction pressure automatically turns off the pumps. A pump low discharge pressure alarm is indicated in the control room and on the local panel. The circulating pump motors can be powered from the diesel- generators if normal power is not available. Circulating pump motor loads are considered nonessential loads and will be operated as required under accident conditions.

The water level in the spent fuel storage pool is maintained at a height which is sufficient to provide shielding for normal building occupancy. Radioactive particulates removed from the fuel pool are collected in filter-demineralizer units which are located in shielded cells. For these reasons, the exposure of plant personnel to radiation from the FPC system is minimal. Further details of radiological considerations for this system are described in Chapter 12.

The circulation patterns within the reactor well and spent fuel storage pool are established by placing the diffusers and skimmers so that particles dislodged during refueling operations are swept away from the work area and out of the pools.

Check valves prevent the pool from siphoning in the event of a pipe rupture.

Heat from pool evaporation is handled by the building ventilation system. Makeup water is provided through a remote-operated valve.

9.1.3.3 Safety Evaluation

for the FPC system upon closure of the fuel gates (21 days)

The maximum possible heat load is the decay heat of the full core load of fuel at the end of the fuel cycle plus the remaining decay heat of the spent fuel discharged at previous refuelings; the maximum capacity of the spent fuel storage pool is 270% of a core. The temperature of the fuel pool water may be permitted to rise to approximately 140°F under these conditions. During cold shutdown conditions, if ~~if~~ it appears that the fuel pool temperature will exceed 125°F, the operator can connect the FPC system to the RHR system. Combining the capacities enables the two systems to keep the

upon closure of the fuel gates

[CLARIFICATION]

water temperature below 125°F. The RHR system will be used only to supplement the fuel pool cooling when the reactor is shut down. The reactor will not be started up whenever portions of the RHR systems are needed to cool the fuel pool. The connecting piping from the fuel storage pool to the RHR system is designed Seismic Category I and can be isolated, assuming a single active failure, from the remainder of the fuel pool system.

These connections may also be utilized during emergency conditions to assure cooling of the spent fuel regardless of the availability of the fuel pool cooling system. The volume of water in the storage pool is such that there is enough heat absorption capability to allow sufficient time for switching over to the RHR system for emergency cooling.

9.1.3 (4)

INSERT 9.1.3.3 A

The 140°F temperature limit is set to assure that the fuel building environment does not exceed equipment environmental limits.

The spent fuel storage pool is designed so that no single failure of structures or equipment will cause inability to: (1) maintain irradiated fuel submerged in water; (2) re-establish normal fuel pool water level; or (3) remove decay heat from the pool. In order to limit the possibility of pool leakage around pool penetrations, the pool is lined with stainless steel. In addition to providing a high degree of integrity, the lining is designed to withstand abuse that might occur when equipment is moved about. No inlets, outlets or drains are provided that might permit the pool to be drained below a safe shielding level. Lines extending below this level are equipped with siphon breakers, check valves, or other suitable devices to prevent inadvertent pool drainage. Interconnected drainage paths are provided behind the liner welds. These paths are designed to: (1) prevent pressure buildup behind the liner plate; (2) prevent the uncontrolled loss of contaminated pool water to other relatively cleaner locations within the containment or fuel-handling area; and (3) provide liner leak detection and measurement. These drainage paths are designed to permit free gravity drainage or pumping to the equipment drain tank.

Seismic Category I

A makeup water system and pool water level instrumentation are provided to replace evaporative and leakage losses. Makeup water during normal operation will be supplied from condensate. The suppression pool cleanup system can be used as a source of makeup water in case of failure of the normal makeup water system.

9.1.3 (1)

INSERT 9.1.3.3 B

Connections from the RHR system to the FPC system provide a Seismic Category I, safety-related makeup capability to the spent fuel pool. The FPC system from the RHR connections to the spent fuel pool are Seismic Category I, safety-related.

9.1.3 (2)

INSERT 9.1.3. C

From the foregoing analysis, it is concluded that the FPC system meets its design bases.

9.1.3.4 Inspection and Testing Requirements

No special tests are required because, normally, one pump, one heat exchanger and one filter-demineralizer are operating while fuel is stored in the pool. The spare unit is operated periodically to handle abnormal heat loads or to replace a unit for servicing. Routine visual inspection of the system components, instrumentation and trouble alarms is adequate to verify system operability.

9.1.3.5 Radiological Considerations

The water level in the spent fuel storage pool is maintained at a height which is sufficient to provide shielding for normal building occupancy. Radioactive particulates removed from the fuel pool are collected in filter-demineralizer units which are located in shielded cells. For these reasons, the exposure of plant personnel to radiation from the FPC system is minimal. Further details of radiological considerations for this and other systems are described in Chapters 11, 12, and 15.

INSERT-9.1.3.3 A

{9.1.3 (4)}

During the initial stages of refueling, the reactor cavity communicates with the fuel pool since the reactor well is flooded and the fuel pool gates are open. Decay heat removal is provided jointly by RHR and FPC systems and che pool temperature kept below 140°F. Evaluation studies concluded that after 150 hours decay following shutdown (fuel pool gates open), the combined decay heat removal capacity of the 1-RHR and 1-FPC heat exchangers (single active failure postulated) can keep the pool temperature well below 140°F. The RHR-FPC joint decay heat removal performance evaluation is shown in Table 9.1-12.

INSERT-9.1.3.3 B

{9.1.3 (1)}

... utilizing either one of the two Seismic Category I makeup lines: (1) the makeup line to the spent fuel pool storage (2) or the makeup line to the dryer/separator storage pool. Makeup water from the dryer/separator fool is delivered to the spent fuel pool by opening pool gates to the spent fuel pool.

Both FPC and SPCU systems are Seismic Category I Quality Group C design with the exception of the filter demineralizer portion which is shared by both systems. Following an accident or seismic event, the filter demineralizers are isolated from FPCS cooling portion and the SPCU by two block valves in series at both the inlet and outlet of the common filter demineralizer portion. Seismic Category I Quality Group C bypass lines are provided on both FPC and SPCU systems to allow continued flow of cooling and makeup water to the spent fuel pool.

INSERT-9.1.3.3 C

{9.1.3 (2)}

Furthermore, firehoses can be used as an alternate makeup source. The fire protection standpipes in the reactor building and their water supply (yard main, one motor driven pump and water source) are seismically designed. The motor driven pump is powered from a bus which has a safety-related diesel generator as one of its power sources. A second seismically designed pump, directly driven by a diesel engine is also provided.

Table 9.1-11

9.1.5
(5)

FUEL POOL COOLING HEAT EXCHANGER AND PERFORMANCE DATA

Number of units	2
Seismic	Category I design and analysis
Types of exchangers	Horizontal U-Tube/Shell
Maximum primary/secondary side pressure	16.0 kg/cm^2g/14.0 kg/cm^2g
Design Condition	Normal heat load operating mode

Primary side (tube side) performance data:

(1) Flow	250 m^3/h
(2) Inlet temperature	52^oC maximum
(3) Allowable pressure drop	0.7 kg/cm^2 max.
(4) Exchanged heat	1.65×10^6 kcal/h

Secondary side (shell side) performance data:

(1) Flow	280 m^3/h
(2) Inlet temperature	35^oC maximum
(3) Allowable pressure drop	0.7 kg/cm^2 max.
(4) Type of cooling water	RCW water

TABLE 9.1-12

9.1.5 (4)

RHR-FPC JOINT DECAY HEAT REMOVAL PERFORMANCE TABLE
(150 HOURS FOLLOWING SHUTDOWN)

RHR-FPC Cooling Loops Combination	Maximum Heat Load @ time = 0 t_o = 150 hrs	Pool Temp. @ time = 0 t_o = 150 hrs	Maximum Pool Temp.	Cooling Time To Max. Temp. From t = 0
2-RHR HX's + 2-FPC HX's	11×10^6 kcal/hr	125°F	125°F	t = 0
2-RHR HX's + 1-FPC HX	11×10^6 kcal/hr	125°F	125°F	t = 0
1-RHR HX + 2-FPC HX's	11×10^6 kcal/hr	125°F	129°F	~ 8 hrs
1-RHR HX + 1-FPC HX	11×10^6 kcal/hr	125°F	136°F	~ 12 hrs

normally closed to flow can be tested to ensure operability and integrity of the system.

Flow to the various systems is balanced by means of manual valves at the individual takeoff points.

9.2.11 Reactor Building Cooling Water System

9.2.11.1 Design Bases

9.2.11.1.1 Safety Design Bases

(1) The reactor building cooling water (RCW) system shall be designed to remove heat from plant auxiliaries which are required for a safe reactor shutdown, as well as those auxiliaries whose operation is desired following a LOCA, but not essential to safe shutdown.

The heat removal capacity is based on the heat removal requirement during LOCA with the maximum ultimate heat sink temperature, 95°F. As shown in Table 9.2-4, the heat removal requirement is higher during other plant operation modes, such as shutdown at 4 hours. However, the RCW system is ~~not~~ designed to remove this larger amount of heat ~~when the ultimate heat sink is at the maximum temperature.~~ to meet the requirements in Subsection 5.4.7.1.1.7.

9.2.11(3)

(2) The RCW system shall be designed to perform its required cooling functions following a LOCA, assuming a single active or passive failure.

(3) The safety-related portions and valves isolating the nonsafety-related portions of

System components and piping materials are selected where required to be compatible with the available site cooling water in order to minimize corrosion. Cathodic protection of the tubing side of the heat exchanger shall be provided. Adequate corrosion safety factors are used to assure the integrity of the system during the life of the plant.

During all plant operating modes, all divisions have at least one RCW cooling water pump operating. Therefore, if a LOCA occurs, the RCW cooling water system required to shut down the plant safely is already in operation. If a loss of offsite power occurs during a LOCA, the pumps momentarily stop until transfer to standby diesel generator power is completed. The pumps are restarted automatically according to the diesel loading sequence. If a LOCA occurs, most nonsafety-related components are automatically isolated from the RCW system. Consequently, no operator action is required, following a LOCA, to start the RCW system in its LOCA operating mode.

All heat exchangers and pumps will be required during the following plant operating conditions, in addition to LOCA: shutdown at 4 hours, shutdown at 20 hours and hot standby with loss of AC power.

Loss of one RCW division will result in loss of RCW cooling to every other RIP (five total) as shown on RRS P&ID (Figure 5.4-4) and will cause those five RIPs to runback to minimum speed. The RIP M-G set in the same electrical division, which is cooled by the same RCW division which failed and powers two more RIPs, would stop by M-G set cooling water prptection. This would completely shutdown three RIPs and would have the resulting total of seven RIPs either at minimum speed or stopped. Assuming the event began at full power on the 100% Control Rod Line, the resulting temporary reactor power would be approximately 60% power. The operator would then correct the RCW problem or initiate a normal plant shutdown.

The drywell cooling system can perform its function after the loss of any RCW division. With only one RCW division and one drywell cooler operating, the drywell temperature will increase but not to a temperature that would damage equipment or require an immediate shutdown.

9.2.11.4 Testing and Inspection Requirements

The RCW system is designed to permit periodic in-service inspection of all system components to assure the integrity and capability of the system.

The RCW system is designed for periodic pressure and functional testing to assure: (1) the structural and leaktight integrity by visible inspection of the components; (2) the operability and the performance of the active components of the system; and (3) the operability of the system as a whole.

The tests shall assure, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for LOCA, including operating of applicable portions of the Reactor Protection System and the transfer between normal and standby power sources. ← INSERT 9.2.11.4 9.2.11(2)

The RCW system is supplied with a chemical addition tank to add chemicals to each division. The RCW system is initially filled with demineralized water. A corrosion inhibitor can be added if desired. These measures are adequate to protect the RCW system from the ill effects of corrosion or organic fouling.

The RCW system is designed to conform with the foregoing requirements. Initial tests shall be made as described in Subsection 14.2.12.

9.2.11.5 Instrumentation and Control Requirements

All equipment is provided with either globe or butterfly valves to give the capability for manual control. These valves are accessible downstream of the equipment for regulation of flow through the equipment or for balancing the circuits. The isolation valves to the nonessential RCW system are automatically and remote-manually operated.

INSERT 9.2.11.4 9.2.11(2)

These tests shall include periodic testing of the heat removal capability of each RCW heat exchanger. Each of these heat exchangers has been designed to provide 20% margin above the heat removal capability required for LOCA in Tables 9.2-4 a, b and c. The revised heat removal capacity of the heat exchangers is shown in Table 9.2-4d. This 20% margin is provided to compensate for the combined effects of fouling and tube plugging. When this margin is no longer present, the heat exchanger heat removal capacity will be increased by either cleaning or retubing.

inhibited demineralized water through the shell side of two of the three 50% capacity TCW heat exchangers in service. The heat from the TCW system is rejected to the turbine service water system which circulates water on the tube side of the TCW system heat exchangers.

The standby TCW system pump is automatically started on detection of low TCW system pump discharge pressure. The standby TCW system heat exchanger is placed in service manually.

The cooling water flow rate to the electro-hydraulic control (EHC) coolers, the turbine lube oil coolers and aftercoolers, and generator exciter air cooler is regulated by control valves. Control valves in the cooling water outlet from these units are throttled in response to temperature signals from the fluid being cooled.

The flow rate of cooling water to all of the other coolers is manually regulated by individual throttling valves located on the cooling water outlet from each unit.

The minimum system cooling water temperature is maintained by adjusting the TCW system heat exchanger bypass valve.

The surge tank provides a reservoir for small amounts of leakage from the system and for the expansion and contraction of the cooling fluid with changes in the system temperature and is connected to the pump suction.

Demineralized makeup water to the TCW system is controlled automatically by a level control valve which is actuated by sensing surge tank level. A corrosion inhibiter is manually added to the system.

9.2.14.3 Safety Evaluation

The TCW system has no safety design bases and serves no safety function.

9.2.14.4 Tests and Inspections

All major components are tested and inspected as separate components prior to installation, and as an integrated system after installation to ensure design performance. The systems are preoperationally tested in accordance with the requirements of Chapter 14.

The components of the TCW system and associated instrumentation are accessible during plant operation for visual examination. Periodic inspections during normal operation are made to ensure operability and integrity of the system. Inspections include measurements of cooling water flows, temperatures, pressures, water quality, corrosion-erosion rate, control positions, and set points to verify the system condition.

9.2.14.5 Instrumentation Application

Pressure and temperature indicators are provided where required for testing and balancing the system. Flow indicator taps are provided at strategic points in the system for initial balancing of the flows and verifying flows during plant operation.

Surge tank high and low level and TCW pump discharge pressure alarms are retransmitted to the main control room from the TCW local control panels.

Makeup flow to the TCW system surge tank is initiated automatically by low surge tank water level and is continued until the normal level is reestablished.

Provisions for taking TCW system water samples are included.

9.2.15 Reactor Service Water System

9.2.15.1 Design Bases

9.2.15.1.1 Safety Design Bases

(1) The reactor service water (RSW) system shall be designed to remove heat from the reactor cooling water system which is required for safe reactor shutdown, and which also cools those auxiliaries whose operation is desired following a LOCA, but not essential to safe shutdown. *
**

(2) The RSW system shall be designed to

* in three divisions
** three divisions of the
(CLARIFICATION)

Seismic Category I and ASME Code, Section III, Class 3, Quality Assurance B, Quality Group C, IEEE-279 and IEEE-308 requirements.

(3) The RSW system shall be protected from flooding, spraying, steam impingement, pipe whip, jet forces, missiles, fire and the effect of failure of any non-Seismic Category I equipment, a required.

(4) The RSW system shall be designed to meet the foregoing design bases during a loss of preferred power.

9.2.15.1.2 Power Generation Design Bases

The RSW system shall be designed to cool the reactor building cooling water (RCW) as required during: (a) normal operation; (b) emergency shutdown; (c) normal shutdown; and (d) testing.

9.2.15.2 System Description

The RSW system provides cooling water during various operating modes, during shutdown and post-LOCA operations. The system removes heat from the RCW system and transfers it to the ultimate heat sink. Figure 9.2-7 shows the RSW system diagram. Component discriptions are provided in Table 9.2-13.

9.2.11 (7)

The RSW system is able to function during abnormally high or low water levels and steps are taken to prevent organic fouling that may degrade system performance. These steps include trash racks and provisions for biocide treatment (where discharge is allowed). Where discharge of biocide is not allowed, non-biocide treatment will be provided. Thermal backwashing capability will be provided at sea water sites where infestations of macrobial growth can occur.

9.2.15.3 Safety Evaluation

The components of the RSW system are separated and protected to the extent necessary to assure that sufficient equipment remains operating to permit shutdown of the unit in the event of any of the following (Separation is applied to electrical equipment and instrumentation and controls as well as to mechanical equipment and piping.):

(1) flooding, spraying or steam release due to pipe rupture or equipment failure;

(2) pipe whip and jet forces resulting from postulated pipe rupture of nearby high energy pipes;

(3 missiles which result from equipment failure; and

(4) fire.

Liquid radiation monitors are provided in the RCW system. Upon detection of radiation leakage in a division of the RCW system, that system is isolated by operator action from the control room, and the cooling load is met by another division of the RCW system. Consequently, radioactive contamination released by the RSW system to the environment does not exceed allowable limits defined by 10CFR100.

System low point drains and high point vents are provided as required.

System components and piping materials are selected to be compatible with the available site cooling water in order to minimize corrosion. Adequate corrosion safety factors are used to assure the integrity of the system during the life of the plant.

During all plant operating modes each division shall have at least one service water pump operating. Therefore, if a LOCA occurs, the system is already in operation. If a loss of offsite power occurs during a LOCA, the pumps momentarily stop until transfer to standby diesel-generator power is completed. The pumps are restarted automatically according to the diesel loading sequence. No operator action is required, following a LOCA, to start the RSW system in its LOCA operating mode.

9.2.15.4 Testing and Inspection Requirements

The RSW system is designed for periodic pressure and functional testing to assure:

(1) the structural and leaktight integrity by visible inspection of the components;

9.2.17 Interfaces

9.2.17.1 Ultimate Heat Sink Capability

Interface requirements pertaining to ultimate heat sink capability are delineated in Subsection 9.2.5 as follows:

Subsection	Title
9.2.5.1	Safety Design Bases
430.204 9.2.5.2	Power Generation Design Bases
9.2.5.6	Evaluation of UHS Performance
9.2.5.7	Safety Evaluation
9.2.5.8	Conformance to Regulatory Guide 1.27
9.2.5.9	Instrumentation and Alarms
9.2.5.10	Tests and Inspections

9.2.17.2 Makeup Water System Capability

The raw water treatment and preparation of the demineralized water is sent to the makeup water system (purified) described in Subsection 9.2.10. ← INSERT 9.2.17.2 9.2.10(4)

The makeup water preparation system shall be located in a building which does not contain any safety-re ated structures, systems or components. If the system is not available,
430.205 demineralized water can be obtained from mobile equipment. The system shall be designed so that any failure in the system, including any that cause flooding, shall not result in the failure of any safety-related structure, system or component.

9.2.17.3 Potable and Sanitary Water System

The potable and sanitary water system shall be designed with no interconnections with systems
430.201 having the potential for containing radioactive materials. Protection shall be provided through the use of air gaps, where necessary. (See Subsection 9.2.4).

INSERT 9.2.17.2 9.2.10(4)

The demineralized water preparation system shall consist of at least two divisions capable of producing at least 200 gpm of demineralized water each. Storage of demineralized water shall be at least 200,000 gallons. If additional demineralized water is needed during peak usage periods, rented portable demineralizers shall be used as required.

TABLE 9.2-3

CAPACITY REQUIREMENTS FOR CONDENSATE STORAGE TANK

Function	Capacity Required
dead space-top of pool	7,900g (Note 1)
normal operation variation and receiving volume for plant startup return water	264,000g
minimum storage volume	66,000g
dead space-middle of pool	34,320g (Note 1)
water source for station blackout	150,480g (Note 2)
dead space-bottom of pool	34,320g (Note 1)
Total	557,020g

NOTE

(1) These values are based on a bottom area of 1,400 ft^2.

~~*(2) This volume includes 121,000 gallons for RCIC operation for eight hours, 12,000 gallons for reactor system leakage for eight hours and margin. However, it is not a dedicated volume and the RCIC can use water from the suppression pool.*~~

(2) Water for operation of RCIC is taken from the condensate storage tank and the suppression pool as described in the EPGs of Appendix 18A.

See next page for response

9.2.9(4)

No text change to SSAR. Table 9.2-3 has been modified accordingly.

Normal alignment for removal of decay heat is with the condensate storage tank. Water for RCIC operation is taken from either the condensate storage tank or the suppression pool as described in the EPGs of Appendix 18A. The volume of water in these two sources, as required by paragraph 3.3.2 of Regulatory Guide 1.155, is sufficient to permit core cooling during station blackout for a duration of eight hours. The switchover from the condensate storage tank to the suppression pool (or the reverse) is performed using station dc power and is not dependent upon either offsite ac power systems or onsite emergency ac power systems.

TABLE 9.2-4d

DESIGN CHARACTERISTICS FOR REACTOR BUILDING COOLING WATER SYSTEM COMPONENTS

RCW Pumps (Two per Division)

	RCW (A)/(B)	RCW (C)
Discharge Flow Rate	5,720 gpm/pump	4,840 gpm/pump
Pump Total Head	82 psig	75 psig
Design Pressure	200 psig	200 psig
Design Temperature	158°F	158°F

RCW Heat Exchangers (Three per Division)

	RCW (A)/(B)	RCW (C)
Capacity (for each heat exchanger)	45 ~~66~~ x10^6 BTU/h	42 ~~63~~ x10^6 BTU/h

RCW Surge Tanks

Capacity	Equal to 30 days of normal leakage
Design Pressure	Static Head
Design Temperature	158°F

RCW Chemical Addition Tanks

Design Pressure	200 psig
Design Temperature	158°F

RCW Piping

Design Pressure	200 psig
Design Temperature	158°F

410.58

9.2.11 (7)

Table 9.2-13

Reactor Service Water System

RSW Pumps (two per division)	
Discharge Flow Rate	7,920 gpm
Pump Total Head	50 psi
Design Pressure	115 psi
Design Temperature	122 F
RSW Piping and Valves	
Design Pressure	115 psi
Design Temperature	122 F

(1) Visual examination of all accessible surfaces of rotors

(2) Visual and surface examination of all low-pressure buckets

(3) 100-percent visual examination of couplings and coupling bolts

The inservice inspection of valves important to overspeed protection includes the following:

(1) All main stop valves, control valves, extraction nonreturn valves, and CBIVs will be tested under load. Test controls installed on the main control room turbine panel and permit full stroking of the stop valve, control valves, and CBIVs. Valve position indication is provided on the panel. No load reduction is necessary before testing main stop and control valves, and CBIVs. Extraction nonreturn valves are tested by equalizing air pressure across the air cylinder. Movement of the valve arm is observed upon action of the spring closure mechanism.

10.2 (1)

43065

(2) Main stop valves, control valves, extraction nonreturn valves, and CBIVs will be tested at least ~~monthly, and more often if recommended by the turbine manufacturer.~~ once a week by closing each valve and observing by the valve position indicator that it moves smoothly to a fully closed position. At least ~~Once~~ once per month, closure of each ~~valve~~ main stop valve, control valve and CBIV during test will be verified by direct observation of the valve motion.

Tightness tests of the main stop and control valves are performed at least once per maintenance cycle by checking the coastdown characteristics of the turbine from no load with each set of four valves closed alternately.

(3) All main stop valves, main control valves, and CBIVs will be inspected once during the first three refueling or extended maintenance shutdowns. Subsequent inspections will be scheduled so that each valve is inspected at 3 to 5 year interval and at least, one valve of each type is inspected after each fuel cycle or 3 1/3 year interval, whichever is less. The inspections will be conducted for:

(a) Wear of linkages and stem packings

(b) Erosion of valve seats and stems

(c) Deposits on stems and other valve parts which could interfere with valve operation

(d) Distortions, misalignment

Inspection of all valves of one type will be conducted if any unusual condition is discovered

10.2.4 Evaluation

The turbine-generator is not nuclear safety related and is not needed to effect or support a safe shutdown of the reactor.

The turbine is designed, constructed, and inspected to minimize the possibility of any major component failure.

The turbine has a redundant, testable overspeed trip system to minimize the possibility of a turbine overspeed event.

Unrestrained stored energy in the extraction steam system has been reduced to an acceptable minimum by the addition of nonreturn valves in selected extraction lines.

The turbine-generator equipment shielding requirements and the methods of access control for all areas of the turbine building ensure that the dose criteria specified in 10CFR20 for operating personnel are not exceeded.

All areas in proximity to turbine generator equipment are zoned according to expected occupancy times and radiation levels anticipated under normal operating conditions.

Specification of the various radiation zones in accordance with expected occupancy is listed in Chapter 12.

If deemed necessary during unusual occurrences, the occupancy times for certain areas will be reduced by administrative controls enacted by health physics personnel.

The design basis operating concentrations of N-16 in the turbine cycle are indicated in Section 12.2.

The connection between the low-pressure turbine exhaust hood and the condenser is made by means of a stainless steel expansion joint.

(2) High condenser pressure turbine trip at 22 inches Hg vaccum

(3) Bypass valve closure at 12 inches Hg vacuum

(4) Main steam isolation valve closes at 7 to 10 inches Hg vacuum

Condenser pressure is an input to the reactor recirculation system. Recirculation pump runback is initiated upon the trip of a circulating water pump when condenser pressure is higher than some site specific preset valve. Runback is automatically initiated when required to avoid a turbine trip on high condenser pressure.

10.4.1.5.3 Temperature

Temperature is measured in each LP turbine exhaust hood by pneumatic temperature controllers. The controllers modulate a control valve in the water spray line protecting the exhaust hoods from overheating.

Circulating water temperatures are monitored upstream and downstream of each condenser tube bundle and are fed to the plant computer and a main control room recorder for use during periodic condenser performance evaluations.

10.4.1.5.4 Leakage

Leakage of circulating water into the condenser shell is monitored by the on-line instrumentation and the process sampling system described in Subsection 9.3.2.

Conductivity of the condensate is continuously monitored at selected locations in the condenser. Conductivity and sodium are continuously monitored at the discharge of the condensate pumps. High condensate conductivity and sodium content, which indicate a condenser tube leak, are individually alarmed in the main control room.

10.4.2 Main Condenser Evacuation System

Noncondensable gases are removed from the power cycle by the main condenser evacuation system (MCES). 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 ~~ventilation system~~ exhaust system at the beginning of each start up.

10.4.2.1 Design Bases

10.4.2.1.1 Safety Design Bases

The MCES does not serve or support any safety function and has no safety design bases.

10.4.2.1.2 Power Generation Design Bases

Power Generation Design Basis One - The MCES is designed to remove air and other power cycle noncondensable gases from the condenser during plant startup, cooldown, and power operation and exhaust them to the offgas system or turbine building compartment ~~ventilation system~~ exhaust system.

Power Generation Design Basis Two - The MCES establishes and maintains a vacuum in the condenser during power operation by the use of steam jet air ejectors, and by the mechanical vacuum pump during early startup.

10.4.2.2 Description

The condenser evacuation system is illustrated in Figure 10.4-1. The system consists of two 100%-capacity, double stage, steam jet air ejector (SJAE) units (complete with intercondenser) for power plant operation, and a mechanical vacuum pump for use during startup. The last stage of the SJAE is a noncondensing stage. One SJAE unit is normally in operation and the other is on standby.

During the initial phase of startup, when the desired rate of air and gas removal exceeds the capacity of the steam jet air ejectors, and nuclear steam pressure is not adequate to operate the air ejector units, the mechanical vacuum pump establishes a vacuum in the main condenser and other parts of the power cycle. The discharge from the vacuum pump is then routed to the turbine building compartment ~~ventilation system~~ exhaust system since there is then little or no effluent radioactivity present. Radiation detectors in the turbine building compartment exhaust system ~~the vacuum pump disharge~~ and plant vent alarm in the main control room if abnormal radioactivity is detected (see Section 7.6). Radiation monitors are provided on the main steam lines which trip the vacuum pump if abnormal radioactivity is detected in the steam being supplied to the condenser.

The steam jet air ejectors ~~is~~ are placed in service to remove the gases from the main condenser after a pressure of about 10 to 15 in Hg absolute is established in the main condenser by the mechanical vacuum pump and when sufficient nuclear steam pressure is available.

430.78 430.79

During normal power operation the steam jet air injectors are normally driven by crossaround ~~condensed~~ steam, with the main steam supply on automatic standby. The main steam supply, however, is normally used during startup and low load operation, and auxiliary steam is available for normal use of the steam jet air ejectors during early startup, should the mechanical vacuum pump prove to be unavailable.

10.4.2.3 Evaluation

The offgas from the main condenser is one source of radioactive gas in the station. Normally it includes the activation gases nitrogen-16, oxygen-19, and nitrogen-13, plus the radioactive noble-gas parents of strontium-89, strontium-90, and cesium-137. An inventory of radioactive contaminants in the effluent from the steam jet air ejectors is evaluated in Section 11.3.

Steam supply to the second stage ejector is maintained at a minimum specified flow rate to ensure adequate dilution of hydrogen and prevent the offgas from reaching the flammable limit of hydrogen.

430.78 430.79

The MCES has no safety-related function as discussed in Section 3.2. Failure of the system will not compromise any safety-related system or component and will not prevent safe reactor shutdown.

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Should the system fail completely, a gradual reduction in condenser vacuum would result from the buildup of noncondensable gases. This reduction in vacuum would first cause a lowering of turbine cycle efficiency due to the increase in turbine exhaust ~~s~~ pressure. If the MCES remained inoperable, condenser pressure would then reach the turbine trip set point and a turbine trip would result. The loss of condenser vacuum incident is discussed in Subsection 15.2.5.

10.4.2.4 Tests and Inspections

Testing and inspection of the system is performed prior to plant operation in accordance with applicable codes and standards.

Components of the system are continuously monitored during operation to ensure satisfactory performance. Periodic inservice tests and inspections of the evacuation system are performed in conjunction with the scheduled maintenance outages.

10.4.2.5 Instrumentation Applications

Local and remote indicating devices for such parameters as pressure, temperature, and flow indicators are provided as required for monitoring the system operation.

10.4.2.5.1 Steam Jet Air Ejectors

430.78 430.79

Steam pressure and flow is continuously monitored and controlled in the ejector steam supply lines. Redundant pressure controllers sense steam pressure at the second stage inlet and modulate the steam supply control valves upstream of the air ejectors. The steam flow transmitters provide inputs to logic devices. These logic devices provide for isolating the offgas flow from the air ejector unit on a two-out-of-three logic, should the steam flow drop below acceptable limits for offgas stream dilution.

10.4.2.5.2 Mechanical Vacuum Pump

430.78 430.79

Pressure is measured on the suction line of the mechanical vacuum pump by a pressure switch. Upon reaching a preset vacuum, the pressure switch energizes a solenoid valve which allows additional seal water to be pumped to the vacuum pumps. Seal pump discharge pressure is locally monitored. Seal water cooler discharge temperature is measured by a temperature indicating switch. On high temperature, the switch activates an annuciator in the main control room. The vacuum pump ~~discharge stream is monitored for radiation prior to entering the turbine ventilation system exhaust.~~ exhaust stream is discharged to the turbine building compartment system which provides for radiation monitoring of the system effluents prior to their release to the monitored vent stack and the atmosphere. The vacuum pump is tripped and its discharge valve is closed upon receiving a main steam high- high radiation signal.

10.4.3 Turbine Gland Seal System

The turbine gland seal system (TGSS) prevents the escape of radioactive steam from the turbine shaft/casing penetrations and valve stems and prevents air inleakage through subatmospheric turbine glands.

10.4.2
(1)

10.4.3.1 Design Bases

10.4.3.1.1 Safety Design Bases

The TGSS does not serve or support any safety function and has no safety design bases.

10.4.3.1.2 Power Generation Design Bases

Power Generation Design Basis One - The TGSS is designed to prevent atmospheric air leakage into the turbine casings and to prevent radioactive steam leakage out of the casings of the turbine-generator.

430.82 Power Generation Design Basis Two - The TGSS returns the condensed steam to the condenser and exhausts the noncondensable gases, via the turbine building compartment exhaust ~~ventilation~~ system, to the plant vent.

Power Generation Design Basis Three - The TGSS has enough capacity to handle steam and air flows resulting from twice the normal packing clearances.

10.4.3.2 Description

10.4.3.2.1 General Description

The turbine gland sealing system is illustrated in Figure 10.4-2. The turbine gland seal system consists of a sealing steam pressure regulator, sealing steam header, a gland steam condenser, with two full-capacity exhauster blowers, and the associated piping, valves and instrumentation.

10.4.3.2.2 System Operation

The annular space through which the turbine shaft penetrates the casing is sealed by steam supplied to the shaft seals. Where the gland seals operate against positive pressure, the sealing steam acts as a buffer and flows either inwards for collection at an intermediate leakoff point, or, outwards and into the vent annulus. Where the gland seals operate against vacuum, the sealing steam either is drawn into the casing or leaks outward to a vent annulus. At all gland seals, the vent annulus is maintained at a slight vacuum and also receives air in leakage from the outside. From each vent annulus, the air-steam mixture is drawn to the gland steam condenser.

430.78 430.79 The seal steam header pressure is regulated automatically by a pressure controller. During startup and low load operation, the seal steam is supplied from the main steam line or auxiliary steam header. Above approximately 50% load, however, sealing steam is normally provided from the heater drain tankvent header. At all loads, gland sealing can be achieved using auxiliary steam so that plant power operation can be maintained without appreciable radioactivity releases even if highly abnormal levels of radioactive contaminants are present in the process steam, due to unanticipated fuel failure in the reactor. 430.78 430.79

The outer portion of all glands of the turbine and main steam valves are connected to the gland steam condenser which is maintained at a slight vacuum by the exhauster blower. During plant operation, the gland steam condenser and one of the two installed 100% capacity motor-driven blowers are in operation. The exhauster blower ~~discharges gland air inleakage~~ to the turbine building compartment ~~ventilation system~~ exhaust system air. The gland steam condenser is cooled by main condensate flow. 430.78 430.79 10.4.3 (1)

10.4.3.3 Evaluation

effluent stream is continously monitored prior to being discharged

430.78 430.79 The turbine gland seal system is designed to prevent leakage of radioactive steam from the main turbine shaft glands and the valve stems. The high-pressure turbine shaft seals must accommodate a range of turbine shell pressure from full vacuum to approximately 220 psia. The low-pressure turbine shaft seals operate against a vacuum at all times. The gland seal outer portion steam air mixture is exhausted to the gland steam condenser via the seal vent annulus (i.e., end glands) which is maintained at a slight vacuum. The radioactive content of the sealing steam which eventually exhausts to the plant vent and the atmosphere is evaluated in Section 11.3 and makes a negligible contribution to overall plant radiation release. In addition, the auxiliary steam system is designed to provide a 100% backup to the normal gland seal process steam supply. A full capacity gland steam condenser is provided, and equipped with two 100% capacity blowers. 430.78 430.79

Relief valves on the seal steam header prevent excessive seal steam pressure. The valves discharge to the condenser shell.

10.4.3.4 Tests and Inspections

Testing and inspection of the system will be performed prior to plant operation. Components of the system are continuously monitored during operation

to ensure that they are functioning satisfactorily. Periodic tests and inspections may be performed in conjunction with maintenance outages.

10.4.3.5 Instrumentation Application

10.4.3.5.1 Gland Steam Condenser Exhausters

10.4.3.5.1.1 Pressure

Gland steam condenser exhauster suction pressure is continuously monitored and reported to the main control room and plant computer. A low vacuum signal actuates a main control room annunciator.

10.4.3.5.1.2 Level

levels

Water levels in the gland steam condenser drain leg are monitored and makeup is added as required to maintain loop seal integrity. Abnormal ~~loads~~ are annunciated in the main control room.

10.4.3.5.2 Sealing Steam Header

Sealing steam header pressure is monitored and reported to the main control room and plant computer. Header steam temperature is also measured and recorded.

INSERT 10.4.3.5.1.3

11.4.3 (1)

10.4.4 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 system is also used to discharge main steam during reactor hot standby and cooldown operations.

10.4.4.1 Design Bases

10.4.4.1.1 Safety Design Bases

The TBS does not serve or support any safety function and has no safety design basis.

10.4.4.1.2 Power Generation Design Bases

Power Generation Design Basis One - The TBS has the capacity to bypass 33 percent of the rated main steam flow to the main condenser.

Power Generation Design Basis Two - The TBS is designed to bypass steam to the main condenser during plant startup and to permit a normal manual cooldown of the reactor coolant system from a hot shutdown condition to a point consistent with initiation of residual heat removal system operation.

Power Generation Design Basis Three - The TBS is designed, in conjunction with the reactor systems, to provide for a 40-percent electrical step-load reduction without reactor trip. The systems will also allow a turbine trip but without lifting the main steam relief and safety valves.

10.4.4.2 Description

10.4.4.2.1 General Description

The TBS is shown in Figure 10.3-1, Main Steam System. The TBS consists of a three valve chest that is connected to the main steam lines upstream of the turbine stop valves, and of three dump lines that connect separately each regulating valve outlet to one condenser shell. The system is designed to bypass 33 percent of the rated main steam flow directly to the condenser. The system and its components are shown in Figures 10.4-10 and 10.4-11.

The turbine bypass system, in combination with the reactor systems, provides the capability to shed 40 percent of the turbine-generator rated load without reactor trip and without the operation of relief and safety valves. A load rejection in excess of 40 percent is expected to result in reactor trip but without operation of any steam relief and safety valve.

10.4.4.2.2 Component Description

One valve chest is provided and houses three individual bypass valves. Each bypass valve is an angle body type valve operated by hydraulic fluid pressure with spring action to close. The valve chest assembly includes hydraulic supply and drain piping; three hydraulic accumulators, one for each bypass valve; servo valves; fast acting servo valves; and, valve position transmitters.

The turbine bypass valves are provided with a separate hydraulic fluid power unit. The unit includes high-pressure fluid pumps, filters, and heat exchangers. High pressure hydraulic fluid is provided at the bottom valve actuator and drained

INSERT 10.4.3.5.1.3

11.4.3
(1)

10.4.3.5.1.3 Effluent Monitoring

The TGSS effluents are first monitored by a system dedicated continuous radiation monitor installed on the gland steam condenser exhauster blower discharge. High monitor readings are alarmed in the main control room. The system effluents are then discharged to the turbine building compartment exhaust system and the plant vent stack where further effluent radiation monitoring is performed. See Subsection 10.4.10.1 for interface requirements pertaining to the radiological analysis of the TGSS effluents.

suction side of the drain pump. This switch will automatically stop the pump in the event of low water level in the standpipe to protect the pump from excessive cavitation.

10.4.5.3 Evaluation

The CWS is not a safey-related system; however, a flooding analysis of the turbine building is performed on the CWS postulating a complete rupture of a single expansion joint. The analysis assumes that the flow into the condenser pit comes from both the upstream and downstream side of the break and, for conservatism, it assumes that one system isolation valve does not fully close.

Based on the above conservative assumptions, the CWS and related facilities are designed such that the selected combination of plant physical arrangement and system protective features ensures that all credible potential circulating water spills inside the turbine building remain confined inside the condenser pit. Further, plant safety is ensured in case of multiple CWS failures or other negligible probability CWS related events by the plant safety related general flooding protection provisions that are discussed in Section 3.4.

10.4.5.4 Tests and Inspections

The CWS and related systems and facilities are tested and checked for leakage integrity prior to initial plant startup and, as may be appropriate, following major maintenance and inspection.

All active and selected passive components of the circulating water system are accessible for inspection and maintenance/testing during normal power station operation.

10.4.5.5 Instrumentation Applications

Temperature monitors are provided upstream and downstream of each condenser shell section.

Indication is provided in the control room to identify open and closed positions of motor-operated butterfly valves in the CWS piping.

All major circulating water system valves which control the flow path can be operated by local controls or by remote manual switches located on the main control board. The pump discharge isolation valves are interlocked with the circulating water pumps so that when a pump is started, its discharge valve will be opening while the pump is coming up to speed, thus assuring there is water flow through the pump. When the pump is stopped, the discharge valve closes automatically to prevent or minimize backward rotation of the pump and motor.

Level switches monitor water level in the condenser discharge water boxes and provide a permissive for starting the circulating water pumps. These level switches ensure that the supply piping and the condenser are full of water prior to circulating water pump startup thus preventing water pressure surges from damaging the supply piping or the condenser.

To satisfy the bearing lubricating water and shaft sealing water interlocks during startup, the circulating water pump bearing lubricating and shaft seal flow switches, located in the lubricating seal water supply lines, must sense a minimum flow to provide pump start permissive.

Monitoring the performance of the circulating water system is accomplished by differential pressure transducers across each half of the condenser with remote differential pressure indicators located in the main control room. Thermal element signals from the supply and discharge sides of the condenser are transmitted to the plant computer for recording, display and condenser performance calculations.

To prevent icing and freeze up when the ambient temperature of the ultimate heat sink falls below 32°F, warm water from the discharge side of the condenser is recirculated back to the screen house intake. Thermal elements, located in each condenser supply line and monitored in the main control room, are utilized in throttling the warm water recirculation valve, which maintains the minimum inlet temperature of approximately 40°F.

10.4.5.6 Flood Protection
See response to question 430.73(b) protection against a CWS pipe, water box or expansion joint failure.

10.4.6 Condensate Cleanup System

The condensate cleanup system (CCS) purifies and treats the condensate as required to maintain reactor feedwater purity, using filtration to remove corrosion products, ion exchange to remove condenser leakage and other impurities, and water treatment additions to minimize corrosion/erosion product releases in the power cycle.

10.4.5 (2)

430.90 through the auxiliary condensers (off-gas recombiner condenser/coolers, gland steam condenser, and steam jet air ejector condensers) and maintains condensate pump minimum flow. Measurements of pump suction and discharge pressures are provided for all pumps in the system.

130.90 The high pressure feedwater heater isolation valves are interlocked such that if a string of heaters were to be removed from service the extraction non-return valves and/or isolation valves for those heaters would automatically close and the heater string bypass valve open. The low pressure feedwater heater isolation valves are interlocked such that, if a string of heaters were removed from service, the extractions to the affected heaters which are equiped with nonreturn valves would automatically close.

Sampling means are provided for monitoring the quality of the condensate and final feedwater, as described in Subsection 9.3.2. Temperature measurements are provided for each stage of feedwater heating. Steam pressure measurements are provided at each feedwater heater. Level instrumentation and controls are provided for automatically regulating the heater drain flow rate to maintain the proper level in each feedwater heater shell or heater drain tank. High-level control valves provide automatic dump-to-condenser of heater drains on detection of high level in the heater shell.

The total water volume in the condensate and feedwater system is maintained through automatic makeup and rejection of condensate to the condensate storage tank. The system makeup and rejection are controlled by the condenser hotwell level controllers.

10.4.8 Steam Generator Blowdown System (PWR)

Not applicable to ABWR.

10.4.9 Auxiliary Feedwater System (PWR)

Not applicable to ABWR.

INSERT →
10.4.10
10.4.3 (1)
10.4.3 (2)

10.4.3
(1)

10.4.3
(2)

10.6.10 Interfaces

10.6.10.1 Radiological of the TGSS effluents

The Applicant referencing the ABWR design shall perform a radiological analysis of the TGSS effluents based on conservative site specific parameters. From this analysis, the Applicant shall determine the various actions to be taken if and when the TGSS effluent radation monitor detects preset levels of effluent containinations, including the level at which the TGSS steam supply will be switched over to auxiliary steam. (See Subsection 10.3.5.1.

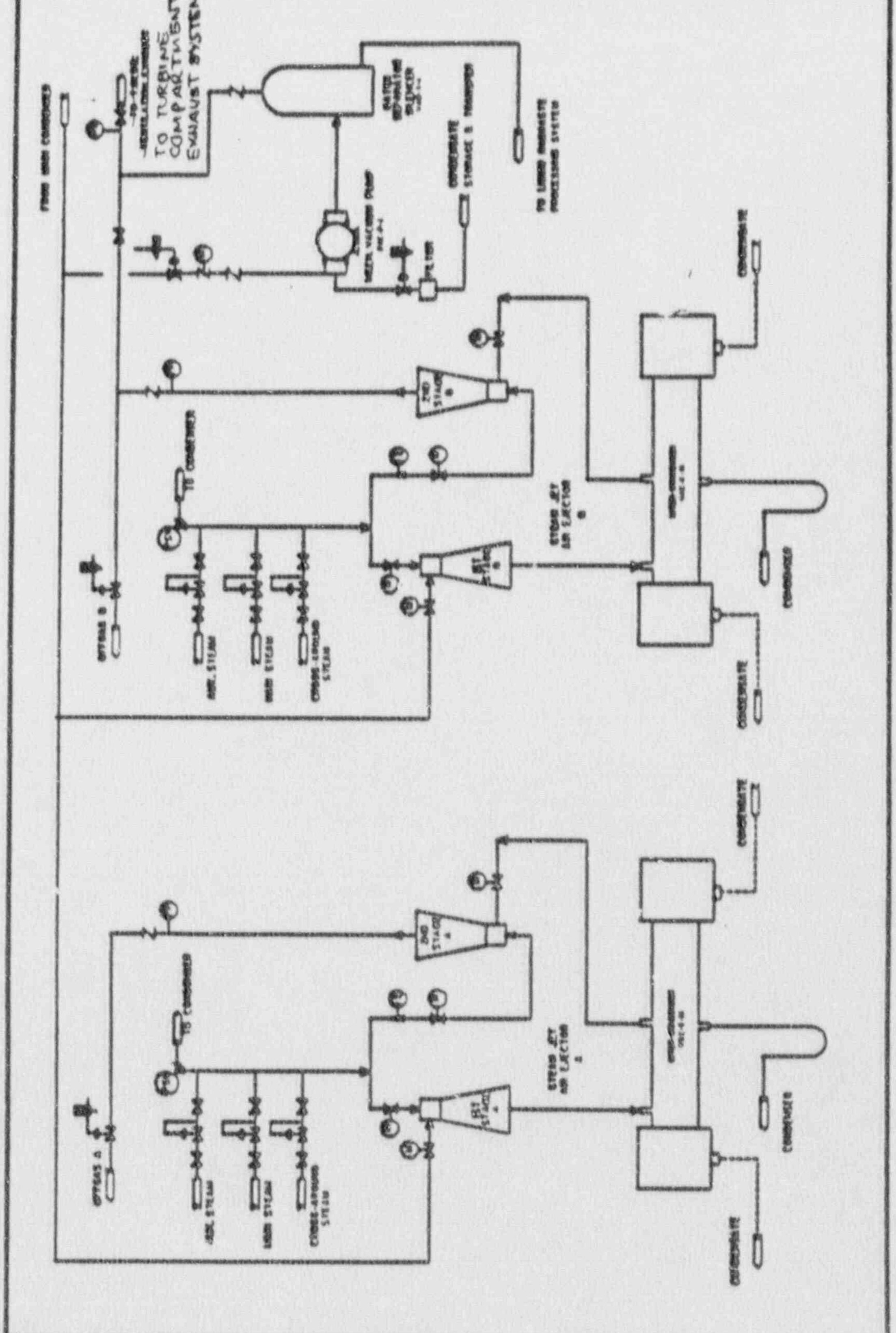

Figure 10.4-1 MAIN CONDENSER EVACUATION SYSTEM

430.78

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11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The process and effluent radiological monitoring and sampling systems are provided to allow determination of the content of radioactive material in various gaseous and liquid process and effluent streams. The design objective and criteria are based on the following requirements:

(1) Radiation instrumentation required for safety and protection.

(2) Radiation instrumentation required for monitor and plant operation.

INSERT 11.E

11.5.1 Design Bases

11.5.1 (1.a)

11.5.1.1 Design Objectives

11.5.1 (1.b)

11.5.1.1.1 Radiation Monitors Required for Safety and Protection

The main objective of this radiation monitoring is to initiate appropriate protective action to limit the potential release of radioactive materials from the reactor vessel and primary and secondary containment if predetermined radiation levels are exceeded in major process/effluent streams. Another objective is to provide control room personnel with an indication of the radiation levels in the major process/effluent streams plus alarm annunciation if high radiation levels are detected.

The process radiation monitoring system provides the following design objectives:

(1) Main steamline tunnel area radiation monitoring;

(2) Reactor building heating, ventilating, and air conditioning (HVAC) exhaust air radiation monitoring;

(3) Fuel handling area HVAC exhaust air radiation monitoring;

(4) Control building HVAC air suply radiation monitoring;

(5) standby gas treatment system off-gas radiation monitoring

11.5.1.1.2 Radiation Monitors Required for Plant Operation

The main objective of this radiation monitoring is to provide operating personnel with measurements of the content of radioactive material in all effluent and important process streams. This ~~allows~~ demonstrat~~ion of~~es compliance with plant normal operational technical specifications by providing gross radiation level monitoring and by collection of halogens and particulates on filters (gaseous effluents) as required by Regulatory Guide 1.21. Additional objectives are to initiate discharge valve isolation on the offgas or liquid radwaste systems if predetermined release rates are exceeded, and to provide for sampling at certain radiation monitor locations to allow determination of specific radionuclide content.

The process radiation monitoring systems also provides the following design objectives:

(1) Monitors gaseous effluent streams

(a) Plant vent ~~off-gas~~ discharge through stack

(b) Turbine building compartment ~~Offgas system area~~ exhaust

(c) Radwaste building ventilation exhaust

(d) Turbine gland steam condenser exhaust ~~and mechanical vacuum pump releases~~.

(2) Monitors liquid effluent streams

(a) Radwaste effluent radioactivity

(b) Drywell sumps drain radioactivity

(3) Monitors gaseous process streams

(a) Off-gas pre-treatment sampling

(b) Off-gas post-treatment sampling

(c) Carbon bed vault gross gamma radiation levels

INSERT 11.5

All critical radioactive release points/paths within the plant are identified and monitored by this system. All other release points/paths of the plant are located in clean areas where radiological monitoring is not required.

(4) Monitors liquid process streams

(a) Reactor building closed cooling water intersystem radiation leakage

11.5.1.2 Design Criteria

INSERT 11.5.1.2

11.5.1.2.1 Radiation Monitors Required for Safety

The design criteria for the main steamline and containment ventilation exhaust plenum radiation monitoring includes the following functional requirements:

(1) Withstand the effect of natural phenomena (e.g., earthquakes) without loss of capability to perform their functions;

(2) Perform the intended safety functions in the environment resulting from normal and abnormal conditions (e.g., loss of HVAC and isolation events)

(3) Meet the reliability, testability, independence, and failure mode requirements of engineered safety features

(4) Provide continuous output in the main control room

(5) Permit checking of the operational availability of each channel during reactor operation with provisions for calibration function and instrument checks

(6) Assure an extremely high probability of accomplishing safety functions in the event of anticipated operational occurrences

(7) Initiate prompt protective action prior to exceeding plant technical specification limits

(8) Provide warning of increasing radiation levels indicative of abnormal conditions by alarm annunciation

(9) Insofar as practical, provide self-monitoring of components to the extent that power failure or component malfunction causes annunciation and channel trip

(10) Register full-scale output if radiation detection exceeds full scale

(11) Use instrumentation with sensitivities and ranges compatible with anticipated radiation levels.

The applicable General Design Criteria of 10CFR50, Appendix A, are 2, 4, 13, 20, 21, 22, 23, 24, and 28 as specified in Section 7.6.2.2. The process radiation safety-related subsystems shall meet the design requirements for Safety Class 2, Seismic Category I, systems along with the quality assurance requirements of 10CFR50, Appendix B.

11.5.1.2.2 Radiation Monitors Required for Plant Operation

The design criteria for operational radiation monitoring ~~shall~~ includes the following functional requirements:

(1) Provide continuous indication of radiation levels in the main control room

(2) Provide warning of increasing radiation levels indicative of abnormal conditions by alarm annunciation

(3) Insofar as practical, provide self-monitoring of components to the extent that power failure or component malfunction causes annunciation and discharge valve isolation channel trip

(4) Monitor a sample representative of the bulk stream or volume

(5) Incorporate provisions for calibration, and functional checks

(6) Use instruments with sensitivities and ranges compatible with anticipated radiation levels

(7) Register full-scale output if radiation detection exceeds full scale.

The radiation system that monitors discharges from the gaseous and liquid radwaste treatment system shall have provisions to alarm and to initiate automatic closure of the waste

INSERT 11.5.1.2

Design criteria of this system are based on meeting the relevant requirements of General Design Criteria (GDC) 60, 63, and 64 of 10CFR50, Appendix A, in accordance with SRP 11.5 of NUREG-0800. These GDCs are in addition to those GDCs that are specified in Section 7.6.2.2 for system instrumentation.

Also, the system is designed to meet the applicable provisions of 10CFR20.106, RG 1.21 and RG 1.97.

The safety-related process radiation monitoring subsystems are classified Safety Class 2, Seismic Category 1. These subsystems conform to the quality assurance requirements of 10CFR50, Appendix B.

discharge valve on the affected treatment system prior to exceeding the normal operation limits specified in technical specifications as required by Regulatory Guide 1.21.

The applicable General Design Criteria of 10CFR50, Appendix A, are 60, 63, and 64 in accordance with the Standard Review Plan for Section 11.5 (NUREG-0800).

11.5.2 System Description

11.5.2.1 Radiation Monitors Required for Safety

Information on these monitors is presented in Table 11.5-1 and the arrangements are shown in Subsection 7.6.1.2.

11.5.2.1.1 Main Steamline (MSL) Radiation Monitoring

This subsection monitors the gamma radiation level exterior to the main steamlines in the MSL tunnel. The normal radiation level is produced primarily by coolant activation gases plus smaller quantities of fission gases being transported with the steam. In the event of a gross release of fission products from the core, the monitoring channels provide trip signals to the leak detection and isolation system.

The MSL radiation monitors consists of four redundant instrument channels. Each channel consists of a local detector ~~(gamma-sensitive~~ ion chamber) and a control room radiation monitor with a trip auxiliary unit. Power for channels A, B, C, and D monitors is supplied from vital 120 Vac divisions 1, 2, 3 and 4 respectively. All four channels are physically and electrically independent of each other.

The detectors are physically located near the main steamlines (MSL) just downstream of the outboard main steamline isolation valves in the stream tunnel. The detectors are geometrically arranged and are capable of detecting significant increases in radiation level with any number of main steamlines in operation. Table 11.5-1 lists the location and range of the detectors.

Each radiation monitor has four trip circuits: two upscale (high-high and high), one downscale (low), and one inoperative. Each trip is visually displayed on the affected radiation monitor. A high-high or inoperative trip in the radiation monitor results in a channel trip which is provided to the reactor protection system (RPS) and to the leak detection and isolation system (LDS). Any two-out-of-four channel trip results in initiation of main steamline isolation valve closure, reactor scram, main condenser mechanical vacuum pump (MVP) shutdown, and MVP line discharge valve closure. A high trip actuates a MSL high control room annunciator common to all channels. High and low trips do not result in a channel trip. Each radiation monitor ~~visually~~ displays the measured radiation level ~~[illegible]~~ in mR/hr

11.5.2.1.2 Reactor Building HVAC Radiation Monitoring

This subsystem monitors the radiation level in the secondary containment of the reactor building ventilation system exhaust duct. A high activity level in the ductwork could be due to fission gases from a leak or an accident.

The system consists of four redundant instrument channels. Each channel consists of a digital gamma-sensitive GM detector and a control room radiation monitor. Power is supplied to each channel, A, B, C, and D monitors from vital 120 Vac Divisions 1, 2, 3 and 4 respectively. A two-pen recorder powered from the 120 Vac instrument bus allows the output of any two channels to be recorded by the use of selection switches. The detectors are located adjacent to the exhaust ducting upstream of the ventilating system isolation valves,

and monitor the HVAC vent exhausts from the primary containment during purging and from the secondary containment. These detectors have sufficient sensitivity to detect high radiation levels during ~~in the~~ primary containment purge to alert the operator for corrective action and to initiate the appropriate protective measures.

Each radiation monitor has four trip circuits: two upscale, one downscale and one inoperative similar to MSL radiation monitors.

11.3.1 (2)

11.5.1 (3)

A high-high or inoperative/downscale trip in the radiation monitor results in a channel trip which is provided to LDS. Any two-out-of-four channel trips will result in the initiation by LDS of the standby gas treatment system (SGTS) and in the isolation of the secondary containment (including closure of the containment purge and vent valves and closure of the reactor building ventilating exhaust isolation valves).

The high-high trip will initiate an alarm in the control room common to all channels.

A downscale inoperative trip is displayed on the radiation monitor and actuates a control room annunciator common to all four channels.

The high radiation trip is provided and actuates a control room annunciator common to all channels.

Each radiation monitor will display the measured radiation level.

~~11.5.2.1.2.1 (Deleted)~~

11.5.2.1.3 Fuel Handling Area Ventilation Exhaust Radiation

This subsystem monitors the off-gas radiation level in the fuel handling area ventilation exhaust duct. The system consists of four channels which are physically and electrically independent of each other. Each channel consists of a digital gamma-sensitive GM detector and a control room radiation monitor. Power for channels (A, B, C, and D) is supplied from the vital 120-Vac divisions 1, 2, 3 and 4 respectively.

Each radiation monitor has four circuits: two upscale, one downscale and one inoperative similar to the MSL radiation monitors. This subsystem performs the same trip functions as those described in Subsection 11.5.2.1.2 for the reactor building HVAC exhaust radiation monitoring.

11.5.2.1.4 Standby Gas Treatment Radiation Monitoring

This subsystem monitors the off-gas radiation level in the SGTS exhaust duct to the stack using four channels.

Two ionization chamber detectors are physically located downstream of the exhaust and heat removal fans and dampers on the exhaust duct to the stack. Two other scintillation detectors are used during off-gas sampling of the gas exhaust to the stack.

The subsystem consists of four instrumented channels. Each channel consists of a detector and a main control room radiation monitor.

Power for the channels is supplied from the non-IE vital 120Vac scurce.

Each radiation monitor has four trip circuits: two upscale, one/inoperative and one downscale. All trips are displayed on the appropriate radiation monitor and each actuates a common main control room annunciator for high-high, high and low/inoperative indications.

11.5.2.1.5 Control Building HVAC Radiation Monitoring

The control building HVAC radiation monitoring subsystem is provided to detect high radiation level in the normal outdoor air supply, automatically close the outdoor air intake ~~dampers~~ and the exhaust dampers, and initiate automatically the outdoor air cleanup system in the emergency recirculation air supply loop. The emergency recirculation fans shall be started and area exhaust fans stopped on high radiation.

The radiation monitors for each of the control building HVAC systems consist of four redundant channels to monitor the air supply to the building. Each radiation monitor is physically separated and powered from separate vital 120 Vac divisional power busses. Failure of one channel will not cause isolation of the HVAC system.

Also, another channel is provided to monitor for radioactive contamination in the air that is being supplied to the control room complex downstream from the supply fans. The detector is located at the common HVAC duct that supplies the air to the various areas in the controlbuilding. Power to this monitor is provided from the non-1E vital 120 Vac bus.

The monitors ~~shall~~ meet the requirements for Class 1E components to provide appropriate reliability. The system will warn of the presence of significant air contamination in inlet air, from any source, ~~but the design~~ and will provide isolation of intake of leakage from accident sources escaping from other plant buildings.

Each radiation channel consists of a digital gamma-sensitive GM detector and a radiation monitor which is located in the control room.

Each radiation monitor has four trip circuits: two upscale, one/inoperative and one downscale. All trips are displayed for the appropriate radiation monitor and each actuates a control room annunciator.

a

11.5.2.2 Radiation Monitors Required for Plant Operation

Information on these monitors is presented in Table 11.5-1.

11.5.2.2.1 Off-gas Pretreatment Radiation Monitoring

This subsystem monitors radioactivity in the condenser offgas at the discharge of the delay pipe after it has passed through the offgas condenser and moisture separator. The monitor detects the radiation level which is attributable to the fission gases produced in the reactor and transported with steam through the turbine to the condenser.

A continuous sample is extracted from the offgas pipe via a stainless steel sample line. It is then passed through a sample chamber and a sample panel before being returned to the suction side of the steam jet air ejector (SJAE). The sample chamber is a stainless steel pipe which is internally polished to minimize plateout. It can be purged with room air to check detector response to background radiation by using a three-way solenoid-operated valve. The valve is controlled by a switch located in the main control room. The sample panel measures and indicates sample line flow. Two ~~ionization chamber~~ detectors are positioned adjacent to the vertical sample chamber and are connected to radiation monitors in the main control room.
digital gamma-sensitive GM

Power is supplied from 120 Vac instrument bus for radiation monitor and detector and for the sample and vital sampler panels.

The radiation monitor has four trip circuits: two upscale (high-high and high), one downscale and one inoperative.

The trip outputs are used for alarm function only. Each trip is visually displayed on the radiation monitor and actuates a control room annunciator: offgas high-high, offgas high, and offgas downscale/inoperative. High or low sample line flow measured at the sample panel actuates a main control room offgas sample high-low flow annunciator.

The radiation level output by the monitor can be directly correlated to the concentration of the noble gases by using semiautomatic vial sampler panel to obtain a grab sample. To draw a sample, a serum bottle is inserted into a sampler holder; the sample lines are evacuated, and a solenoid-operated sample valve is opened to allow offgas to enter the bottle. The bottle is then removed and the sample is analyzed in the counting room with a multichannel gamma pulse height analyzer to determine the concentration of the various noble gas radionuclides. A correlation between the observed activity and the monitor reading permits calibration of the monitor.

11.5.2.2.2 Off-gas Post-Treatment Radiation Monitoring

This system monitors radioactivity in the offgas piping downstream of the offgas system charcoal adsorbers and upstream of the offgas system discharge valve. A continuous sample is extracted from the offgas system piping, passed through the offgas post-treatment sample panel for monitoring and sampling, and returned to the offgas system piping. The sample panel has a pair of filters (one for particulate collection and one for halogen collection) in parallel (with respect to flow) with two identical ~~scintillation~~ detectors. Two radiation monitors in the main control room analyze and visually display the measured gross radiation level.
GM

The sample panel shielded chambers can be purged with room air to check detector response to background radiation by using solenoid valves operated from the control room. The sample panel measures and indicates sample line flow. A solenoid operated check source for each detector assembly operated from the control room can be used to check operability of the gross radiation channel.

Power is supplied from a 120-Vac instrument bus to the radiation monitors and to the two-pen

recorder. A 120-Vac local bus supplies the sample panel.

Each radiation monitor has four trip circuits: two upscale (high-high and high), one downscale (low) and one inoperative. Each trip is visually displayed on the radiation monitor. The trips actuate corresponding main control room annunciators: offgas post treatment high-high radiation, offgas post treatment high radiation, and offgas post treatment downscale/inoperative.

High or low ~~sample~~ flow measured at the sample panel actuates ~~a main control room offgas vent pipe sample~~ the abnormal flow annunciator in the control room.

A trip auxiliary unit in the control room takes the high-high and downscale trip/inoperative outputs to initiate closure of the offgas system discharge and bypass valves. The high-high trip setpoints are determined so that valve closure is initiated prior to exceeding technical specification limits. Any one high upscale trip initiates closure of offgas system bypass line valve and permits opening of the treatment line valve.

A vial sampler panel similar to the pre-treatment sampler panel is provided for grab sample collection to allow isotopic analysis and gross monitor calibration.

11.5.2.2.3 Carbon Bed Vault Radiation Monitoring

The carbon vault is monitored for gross gamma radiation level with a single instrument channel. The channel includes a digital sensor and converter, and a radiation monitor. The sensor is located outside the vault on the HVAC exhaust line from the vault. The radiation monitor is located in the main control room. The channel provides for sensing and readout of gross gamma radiation over a range of six logarithmic decades (1 to 10^6 mR/hr).

The monitor has one adjustable upscale trip circuit for alarm and one downscale trip for instrument trouble. Power is supplied from 120 Vac instrument bus.

through the stack

11.5.2.2.4 Plant Vent Discharge Radiation Monitoring

This system monitors the plant vent discharges for gross radiation level during normal plant operation and collects halogen and particulate samples for laboratory analysis. The discharge through this common plant vent include HVAC exhausts from the reactor, turbine, radwaste and service buildings. Also, this system utilizes a high-range radiation monitor that measures fission products in plant gaseous effluents during and following an accident. A representative sample is continuously extracted from the ventilation ducting through two isokinetic probes in accordance with ANSI N13.1 and passed through the containment ventilation sample panels for monitoring and sampling, and returned to the ventilation ducting. Each sample panel has a pair of filters (one for particulate collection and one for halogen collection) in parallel (with respect to flow) for continuous gaseous radiation sampling. The gross radiation detection assembly consists of a shielded chamber, beta-gamma-sensitive GM tubes, and a check source. The extended range detector assembly consists of an ionization chamber which ~~could~~ measure radiation levels up to 10^5 mCi/cc. A radiation monitor in the main control room analyzes and visually displays the measured radiation level.

The sample panel shielded chambers can be purged with room air by using two solenoid valves operated from the control room to check detector response to background radiation, thus checking operability of the gross radiation channel.

Power is supplied from 120-Vac local bus for the radiation monitor and for the sample panel.

The radiation monitor initiates trips for alarm indications on high-high, high, and low radiation from each detector assembly. Also, the sampled line is monitored for high or low flow indications and alarming.

Table 11.5-2 presents the gaseous and airborne monitors for the effluent radiation monitoring system.

11.5.2.2.5 Liquid Process and Effluent Monitoring

These subsystems monitor the gamma radiation levels of liquid process and effluent streams. With the exception of the radwaste system effluent, the streams monitored normally contain only background levels of radioactive materials. Increases in radiation level may be indicative of heat exchanger leakage or equipment malfunction.

11.5.1 (1.6)

11.5.2.2.5.1 Radwaste Effluent Radiation Monitoring

This subsystem continuously monitors the radioactivity in the radwaste effluent prior to its discharge and drainage.

Liquid waste can be discharged from the sample tanks containing liquids that have been processed through one or more treatment systems such as evaporation, filtration, and ion exchange. Prior to discharge, the liquid is extracted from the liquid drain treatment process pipe, passed through a liquid sample panel which contains a detection assembly for gross radiation monitoring, and returned to the process pipe. The detection assembly consists of a scintillation detector mounted in a shielded sample chamber equipped with a check source. A radiation monitor in the control room analyzes and visually displays the measured gross radiation level.

The sample panel chamber and lines can be drained to allow assessment of background buildup The panel measures and indicates sample line flow. A solenoid-operated check source operated from the control room can be used to check operability of the channel.

Based on acceptable radiation levels, discharge is permitted at a specified release rate and dilution rate.

The radiation monitor has ~~four~~ three trip circuits. Two upscale trips (high-high and high), and one downscale ~~trip, and one~~ inoperative trip.

The high-high upscale trip and the downscale/inoperative trip are used to stop the HCW effluent pump. Also

The two upscale trips and the low downscale/inoperative trip actuate annunciators in the main control room and in the radwaste building control room. Table 11.5-3 describes the liquid monitors used for process radiation monitoring.

11.5.2.2.~~5~~6 Reactor Building Cooling Water Radiation Monitoring

This subsystem consists of three channels: one for each RCW A, B and C loop for monitoring intersystem radiation leakage into the reactor building cooling water system.

Each channel consists of a scintillation detector which is located in a well near the RCW heat exchanger exit pipe. Radiation detected from the three channels are multiplexed and fed into a common ~~process~~ radiation monitor. This monitor provides individual channel trips on high radiation level and downscale/inoperative indication for annunciation in the control room. Power to the monitors is provided from the non-1E vital 120 Vac source.

11.5.2.2.~~6~~7 Radwaste Building HVAC ~~Radiation~~ Ehaust Monitoring

This subsystem monitors the radwaste building ventilation discharge to the stack, including radwaste storage tank vents, for gross radiation level. The system consists of two redundant instrument channels, each channel ~~has~~ having a local detector, a converter, and a main control room radiation monitor. Power is supplied to each channel by the 120-Vac instrument bus.

Each radiation monitor provides two trip circuits: one for upscale (high) radiation and one for downscale/inoperative trip.

The trip signals are annunciated in the radwaste building control room and in the main control room.

Each radiation monitor visually displays the radiation level and supplies an output signal to the computer.

A gamma check source is provided for channel calibration.

11.5.2.2.~~7~~8 ~~Offgas Area Exhaust Radiation~~ Turbine Building Compartment Exhaus Monitoring System

This subsystem monitors the off-gas vent discharge in the turbine building ~~equipment areas~~ compartment for gross radiation levels. The monitoring is provided by four channels (two redundant sets). Two redundant channels monitor radiation in the equipment area air exhaust duct and the other two redundant channels monitor the radiation in the SJAE area air exhaust duct. Each channel uses a digital detector located adjacent to the monitored exhaust duct. The outputs from each set of detectors are multiplexed and then fed into two separate

process radiation monitors for display, recording and annunciation. Each monitor provides alarm trips on radiation high and on radiation low (downscale/inoperative).

11.5.2.2.~~8~~ 9 Turbine Gland ~~Seal~~ Steam Exhaust & Discharge Monitoring Condenser ~~and Vacuum Pump Offgas Radiation Monitoring~~

This subsystem monitors the off-gas releases to the stack from the turbine gland seal system. ~~and from the steam condenser vacuum pump~~. The off-gas releases are continuously sampled and monitored for noble gases by a scintillation detector. The output signal is multiplexed and then fed to a shared radiation monitor in the main control for display, recording and annunciation. This monitor provides ~~alarm~~ two trip alarms, one on radiation high and one on radiation low (downscale/inoperative).

A grab sample of the off-gas is provided for laboratory analysis. Also, samples of halogens and particulates are collected on filters for periodic analysis.

A gamma source check is provided for channel calibration purposes.

11.5.2.2.~~9~~ 10 Drywell Sumps Drain Line Radiation Monitoring

This subsystem monitors the radiation level in the liquid waste that is transferred in the drain line from the drywell LCW and HCW sumps to the radwaste system. One monitoring channel is provided ~~for the drain line from~~ in each sump drain Line. Each channel uses an ionization chamber which is located on the drain line from the sump just downstream from the outboard isolation valve. The output from each sensor is multiplexed and then fed into a shared ~~process~~ radiation monitor for display, recording and annunciation.

The radiation monitor provides ~~four~~ three trip circuits: two upscale (radiation high-high and high), one downscale/~~and one~~ inoperative. The high-high signal is used to close the outboard isolation valve in its respective drain line. All trips are annunciated in the main control room.

11.5.3 Effluent Monitoring and Sampling

All potentially radioactive effluent materials are monitored for radioactivity releases in accordance with Criterion 64 of General Design Criteria, 10CFR50, Appendix A, as follows:

(1) liquid releases are monitored for gross gamma radioactivity;

(2) solid wastes are monitored for gross gamma radioactivity; and

(3) gaseous releases are monitored for gross gamma radioactivity.

11.5.3.1 Basis for Monitor Location Selection

Monitor locations are selected to assure that all effluent materials comply with regulatory requirements as covered in Regulatory Guide 1.21.

11.5.3.2 Expected Radiation Levels

Expected radiation levels are within the ranges specified in Tables 11.5-2 and 11.5-3.

11.5.3.3 Instrumentation

The process radiation monitors used for measuring radioactivity are listed in Table 11.5-1.

Grab samples are analyzed to identify and quantify the specific radionuclides in effluents and wastes. The results from the sample analysis are used to establish relationships between the gross gamma monitor readings and concentrations or release rates of radionuclides in continuous effluent releases.

11.5.3.4 Setpoints

The radiation level trip setpoints for actuation of automatic control features that initiate actuation of isolation valves, dampers or diversion valves are specified in the plant technical specifications as indicated in Table 11.5-1.

11.5.4. Process Monitoring and Sampling

11.5.4.1 Implementation of General Design Criterion 60

All potentially significant radioactive discharge paths are equipped with a control system to automatically isolate the discharge on indication of a high radiation level. These include:

(1) Off-gas post-treatment

(2) Reactor building ~~containment~~ HVAC air exhaust.

(3) Fuel handling area air exhaust.

(4) Drywell Sump Liquid Waste drain

(5) Radwaste effluent

11.5.4.2 Implementation of General Design Criteria 64

Radiation levels in radioactive and potentially radioactive process streams are monitored for radioactivity releases. These include:

(1) Main steamline

(2) Off-gas pre-treatment and post-treatment

(3) Carbon bed vault

(4) Intersystem leakage into reactor building cooling water

11.5.4.3 Basis for Monitor Location Selection

Monitor locations are selected to assure compliance with Regulatory Guide 1.21 in that sample points are located where there is a minimum of disturbance due to fittings and other physical characteristics of the equipment and components. Sample nozzles are inserted into the flow or liquid volume to ensure sampling the bulk volume of pipes and tanks. In the case of both liquid and gas flow, care is taken to assure that individual samples are actually representative of the effluent mixture. A more detailed discussion is given in ANSI N13.1.

11.5.4.4 Expected Radiation Levels

Expected radiation levels are listed in Tables 11.5-2 and 11.5-3.

11.5.4.5 Instrumentation

The process radiation monitors used for measuring radioactivity are listed in Table 11.5-1.

Grab samples are analyzed to identify and quantify the specific radionuclides in process streams. The results from the sample analysis are used to establish relationships between the gross gamma monitor readings and concentration and radionuclides in the process streams.

11.5.4.6 Setpoints

The radiation trip set points for the various monitors are listed in Table 11.5-1.

11.5.5 Calibration and Maintenance

11.5.5.1 Inspection and Tests

During reactor operation, daily checks of system operability are made by observing channel behavior. At periodic intervals during reactor operation, the detector response of each monitor provided with a remotely positioned check source will be recorded together with the instrument background count rate to ensure proper functioning of the monitors. Any detector whose response cannot be verified by observation during normal operation or by using the remotely positioned check source will have its response checked with portable check source. A record will be maintained showing the background radiation level and the detector response.

The system has electronic testing and calibrating equipment which permits channel testing without relocating or dismounting channel components. An internal trip test circuit adjustable over the full range of the readout meter is used for testing. Each channel is tested at least semiannually prior to performing a calibration check. Verification of channel operation and trip function will be done at this time if it can be done without jeopardizing plant safety. The test will be documented.

The following monitors have alarm trip circuits which can be tested by using test signals or portable gamma sources:

(1) Main steamline

(2) Reactor building HVAC

(3) Fuel handling area HVAC

(4) Control building HVAC

(5) Reactor building cooling water system

(6) SGTS

(7) Turbine building compartment exhaust ~~equipment area offgas~~

(8) Offgas pretreatment ~~and~~

(9) Carbon bed vault

The following monitors include built-in check sources and purge systems which can be operated from the main control room:

(1) Offgas post-treatment

(2) Plant vent discharge

(3) Liquid waste discharge

(4) SGTS ~~offgas vent~~

(5) Radwaste building exhaust

(6) Gland Steam Condenser exhaust ~~and Condenser vacuum pump exhaust~~

11.5.5.2 Calibration

The continuous radiation monitor calibration is according to certified National Bureau of Standards of commercial radionuclide standards, and is accurate to at least + or - 15%. The source-detector geometry during primary calibration is identical to the sample-detector geometry in actual use. Secondary standards which were counted in reproducible geometry during the primary calibration are supplied with each continuous monitor for calibration after installation. Each continuous monitor is calibrated during plant operation or during the refueling outage if the detector is not readily accessible. A calibration can also be performed by using liquid or gaseous radionuclide standards or by analyzing particulate iodine or gaseous grab samples with laboratory instruments.

The offgas pretreatment monitor shall respond to a gross gamma signal obtained from the periodic analyses of grab samples. The readout units shall be mR/hr per mCi/sec.

The following monitors r~~espond to a~~ display the gross gamma signal o~~btained from the periodic analyses of grab samples to read the rate~~ in counts/min

(1) Off-gas post-treatment~~;~~

(2) Plant ~~stack~~ vent discharge~~;~~

(3) Radwaste effluent discharge ~~and~~

(4) Gland steam condenser ~~and vacuum pump discharge~~ exhaust

(5) Reactor building cooling water system

The following monitors are calibrated to ~~read~~ provide measurements of the gross gamma dose rate in mR/hr:

(1) Main steamline;

(2) Reactor building HVAC~~;~~

(3) Fuel handling area HVAC~~;~~

(4) Carbon bed vault~~;~~

(5) Control building HVAC~~;~~

(6) Turbine building ~~equipment~~ compartment exhaust ~~off-gas [illegible]~~

(7) Radwaste building HVAC exhaust~~.~~

8. Off-gas pre-treatment

9. Drywell sump liquid drain line

11.5.5.3 Maintenance

All channel detectors, electronics, and recorder are serviced and maintained on an annual basis or in accordance with manufacturers recommendations to ensure reliable operations. Such maintenance includes cleaning, lubrication, and assurance of free movement of the recorder in addition to the replacement or adjustment of any components required after performing a test or calibration check. If any work is performed which would affect the calibration, a recalibration is performed at the completion of the work.

11.5.5.4 Audits and Verifications

Audits and verification during normal plant operation are out-of-scope for the Standard ABWR Plant.

TABLE 11.5-1

PROCESS AND EFFLUENT RADIATION MONITORING SYSTEMS

~~Example~~ Setpoints

Monitored Process	No. of Channels	Detector Type	Sample Line or Detector Location	Channel Range (Note 1) 11.5.1 (2)	Warning Alarm	ACF Trip	Scale
A. Safety-Related Monitors							
Main steam-line tunnel area	4	GS/C	Immediately downstream of plant main steamline isolation valve	$1 \cdot 10^6$ mR/hr (10^{-13} to 10^{-6} Amps)	above full power background, below trip	technical specification	6 dec. log
Reactor building HVAC exhaust	4	S/C	Exhaust duct upstream of exhaust ventilation isolation isolation valve	0.01 to 100 mR/hr	above background, below trip	technical specification	4 dec. log
Control building HVAC air supply	~~4~~ 8*	S/C	Intake duct upstream of intake ventilation isolation valve	0.01 to 100mR/hr	above background, below trip	technical specification	4 dec. log
Standby gas treatment system off-gas	2	S/D	SGTS exhaust air duct downstream of exhaust and heat removal fans and dampers	0.1 to 10^5 cpm ~~0.01 to 100 mR/hr~~	above background, below trip	technical specification	~~4~~ 6 dec. log
	2	IC		10^{-13} to 10^{-6} Amps ~~up to 10^4 μCi/cc~~	above background	None	6 dec. log
Fuel handling area air exhaust	4	S/C	Locally above operating floor	0.1 to 10^3 mR/hr	above background, below trip	technical specification	4 dec. log

* 4 Channels for each air intake

TABLE 11.5-1

PROCESS AND EFFLUENT RADIATION MONITORING SYSTEMS
(Continued)

Monitored Process	No. of Channels	Detector Type	Sample Line or Detector Location	Channel Range (Note 1) 11.5.1 (2)	~~Hi~~ Setpoints Warning Alarm	ACF Trip	Scale
B. Monitors Required for Plant Operation							
Radwaste liquid discharge ~~to canal~~	1	S/D	Sample line	10^{-1} to 10^{4} ~~10 to 10^6~~ counts/min	above background, below trip	technical specification ~~None~~	5 dec. log
Reactor building cooling water system	3	S/D	RCW Hx line exit	10^{-1} to 10^{4} ~~10 to 10^6~~ counts/min	above background	None	5 dec. log
Offgas ~~charcoal~~ post ~~vault exhaust~~ treatment	2	GM-B	Sample line	10 to 10^6 counts/min	above background, below trip	technical specification	6 dec. log
Offgas ~~charcoal~~ pre-~~vault inlet~~ treatment	1	S/C ~~D/S~~	Sample line	1 to 10^6 mR/hr	at tech spec report level	None	6 dec. log
Charcoal vault	1	S/C ~~D/S~~	On Charcoal vault HVAC exhaust Line	1 to 10^6 mR/hr	above background	None	6 dec. log
Plant vent ~~stack~~ discharge	1	GM-B	Sample line	10 to 10^6 counts/min	at quarterly tech spec level	None	5 dec. log
	1	IC	Sample line	10^{-13} to 10^{-6} A ~~up to 10^{7} µCi/sec~~	above background, below trip	None	6 dec. log
Radwaste building HVAC vent	2	GM-B	Exhaust ducts	0.01 to 100 mR/hr	above background, below trip	None	4 dec. log

TABLE 11.5-1

PROCESS AND EFFLUENT RADIATION MONITORING SYSTEMS
(Continued)

Monitored Process	No. of Channels	Detector Type	Sample Line or Detector Location	Channel Range (Note 1) 11.5.1 (2)	Setpoints: Warning Alarm	Setpoints: ACF Trip	Scale
B. Monitors Required for Plant Operation							
Compartment T/B ~~[illegible]~~ exhaust	4	S/C	Exhaust duct	0.01 to 100 mR/hr	above background	None	4 dec. log
Drywell sump ~~area~~ liquid drain	2	IC	Drain line from LCW & HCW sumps	~~0.01 to 100~~ 1 to 10^6 mR/hr	above background	Technical specification	~~4~~ 6 dec. log
~~Turbine~~ Gland Steam Condenser ~~&~~ ~~Vacuum Pump Effluent~~ exhaust discharge	1	~~S/C~~ S/D	Sample line	~~up to 10^5 µCi/cc~~ 0.1 to 10^5 cpm	above background	None	6 dec. log

Legend

- ACF Automatic Control Function
- ~~D/S Digital Sensor and Converter~~
- GM-B Beta-~~Scensitive~~ sensitive GM Detector
- ~~GSIC Gamma-Sensitive Ion Chamber~~
- IC Ion Chamber
- S/C Digital Gamma-Sensitive GM Detector
- S/D Scintillation Detecter

Note 1. 11.5.1 (2)
The channel range specified in this table is the equipment measuring or display range of the indicated parameter. Refer to Tables 11.5-2 & 11.5-3 for the dynamic detection range of the monitoring channel expressed as concentration in units of microcuries per cubic centimeter, referenced to a specific nuclide.

11.5.1 ENTIRE
(2) TABLE

TABLE 11.5-2

PROCESS RADIATION MONITORING SYSTEM (GASEOUS AND AIRBORNE MONITORS)

Radiation Monitor	Configuration	Type	Sensitivity	Dynamic Detection Range	Principal Radionuclides Measured	Expected Activity**	Alarms & Trips
Offgas ~~charcoal vault exhaust~~ post treatment	Offline	B-GM ~~Particulate~~ Filter - P ~~Iodine~~ Filter - I ~~B-GM~~	0.25 cpm/pCi/cm^3	10^{-5} to 10^{1} µCi/cc ~~10^{1} - 10^{6} cpm~~	Xe-133* ~~Kr-85*~~ Cs-137 I-131	5×10^{-5} µCi/cc ~~5 cps~~	~~Low~~ Flow H/L INOP/Low High High-High
Offgas ~~charcoal vault inlet~~ pre-treatment	Adjacent to sample chamber	S/C ~~D/S~~	700 ~~50~~ mR/hr per µCi/cc	10^{-3} to 10^{4} µCi/cc ~~10^{0} - 10^{6} mR/hr~~	Noble gas fission products	~0.3 µCi/cc ~~~100 mR/hr~~	~~INOP/Low High~~ High-High High Low/INOP Flow H/L
Main Steamline Radiation	Adjacent to steam lines	GSIC	3.7 ~~2~~ x 10^{-10} Amp/R/hr (Co-60)*	10^{0} - 10^{6} mR/hr	Coolant activation gases	~100 mR/hr	INOP Low High High-High
Charcoal vault	In Line ~~Offline~~	S/C ~~D/S~~	0.5 mR/hr per 10^{-3} µCi/cc	10^{-3} to 10^{3} µCi/cc ~~10^{0} - 10^{6} mR/hr~~	Noble gases	Negligible	~~Low~~ High ~~High~~ Low/INOP
T/B compartment ~~equipment~~ ~~Offgas vent~~ exhaust	~~Offline~~ In Line	S/C ~~D/S~~	0.5 mR/hr ~~250 cpm/pCi/cm^3~~ per 10^{-3} µCi/cc	10^{-5} to 10^{-1} ~~10^{1} - 10^{6} cpm~~ µCi/cc	~~Kr-85*~~ Xe-133* Xe-135	~4×10^{-5} ~~~2×10^{-2} mR/hr~~ µCi/cc	~~Low flow~~ High Low/INOP ~~High~~ ~~High~~-High
Reactor building HVAC air exhaust	In Line ~~Offline~~	S/C	0.5 ~~0.01~~ mR/hr per 10^{-3} µCi/cc	10^{-5} to 10^{-1} ~~0.01 - 100 mR/hr~~ µCi/cc	noble gases Xe-133* Xe-135	~4×10^{-5} µCi/cc ~~~2×10^{-2} mR/hr~~	INOP/ Low High High-High Isolate
Plant vent ~~Main stack off-gas~~ discharge (normal range)	Offline	B-GM ~~Particulate~~ Filter - P ~~Iodine~~ Filter - I ~~B-GM~~	250 cpm/pCi per ~~cm^3~~ cc	10^{-7} to 10^{-1} ~~10^{1} - 10^{6} cpm~~ µCi/cc	Xe-133* ~~Kr-85~~ Cs-137 I-131	~5×10^{-5} ~~~5 cps~~ µCi/cc	High/Low flow INOP/Low High High-High

* *Sensitivity based upon this radionuclide.*
** *Expected activities are estimated based on existing plants.*

TABLE 11.5-2

PROCESS RADIATION MONITORING SYSTEM (GASEOUS AND AIRBORNE MONITORS) (Continued)

Radiation Monitor	Configuration	Type	Sensitivity	Dynamic Detection Range	Principal Radionuclides Measured	Expected Activity**	Alarms & Trips
Main Stack (High-Range)	Offline	IC	1.6×10^{-10} ~~2×10^{-2} mCi/cc~~ A/µCi/cc	10^{-2} ~~up~~ to 10^5 µCi/cc	Xe-133*	5×10^{-3} µCi/cc ~~2×10^{-3} mCi/cc~~	High/Low flow INOP/Low High High-High
Radwaste building ventilation discharge ~~to main stack~~	~~In-line~~ ~~Offline~~ Off Line	B-GM	0.5 ~~0.01~~ mR/hr per 10^{-3} µCi/cc Filter-P Filter-I	10^{-5} to 10 ~~0.01-100 mR/hr~~ µCi/cc	Xe-133* ~~Kr-85~~ Cs-137 I-131	~ 10^{-5} µCi/cc ~~2×10^{-2} mR/hr~~	~~Low~~ High-High ~~INOP~~ High ~~High~~ Low/INOP Flow H/L
Glandsteam condenser ~~and mechanical vacuum pump~~ exhaust discharge	Offline	S/D Filter-P Filter-I	1.33 cpm $\times10^5$ ~~1 cps/10~~ per µCi/cc	10^{-5} to 10^1 ~~10 to 10^6 counts/min~~ µCi/cc	Xe-133 ~~Cs-137~~ Cs-137* I-131	~ 10^{-6} µCi/cc ~~~5cps~~	~~Downscale~~ High Low/INOP ~~High~~ ~~High-High~~ High-High/INOP
Control Bldg. HVAC Air Intake	~~Offline~~ In Line	S/C	0.5 ~~0.01~~ mR/hr per 10^{-3} µCi/cc	10^{-5} to 10^{-1} ~~0.01-100 mR/hr~~ µCi/cc	~~Kr-85*~~ Xe-133*	Negligible	~~INOP~~/High Low
Standby Gas Treatment Exhaust	Offline In Line	S/D IC	1.33 cpm $\times10^5$ per ~~1 cps/10~~ µCi/cc ~~$6\times10^{-?}$ mCi/cc~~ 1.6×10^{-10} A/µCi/cc	10^{-7} to 10^{-1} ~~10-10^6 counts/min~~ µCi/cc 10^{-2} ~~up~~ to 10^5 µCi/cc	~~Cs-137~~ Noble Gases ~~Cs-137~~ Noble Gases	~5×10^{-7} µCi/cc ~~~5cps~~ ~~~2~~ 6 $\times10^{-7}$ µCi/cc	~~INOP~~ High-High ~~Low~~ High Low/INOP
Fuel Handling Area Exhaust	In Line ~~Offline~~	S/C	~~2.8×10^1~~ 34 mR/hr per µCi/cc (Cs-137)*	10^{-3} to 10^2 ~~10^{-1}-10^3 mR/hr~~ µCi/cc	~~Cs-137~~ Noble Gases	~ 6×10^{-3} µCi/cc ~~2×10^1 mR/hr~~	~~INOP/High~~ ~~Low~~ High-High High Low INOP Isolate

P = Particulate Filter

I = Iodine or Charcoal Filter

{ Filter - P Cs-137*
{ Filter - I I-131

* Sensitivity based upon this radionuclide.

** Expected activities are estimated and are based on existing plants.

11.5.1 (2) ENTIRE TABLE

TABLE 11.5-3

PROCESS RADIATION MONITORING SYSTEM (LIQUID MONITORS)

Radiation Monitor	Configuration	Sensitivity ~~Type~~	Dynamic Detection Range	Principal Radionuclides Measured	Expected Activity**	Alarms & Trips
Radwaste effluent radiation monitor	Off ~~Inline~~	~~Gamma-Scint~~ 1.33×10^{5} cpm/µCi per cc	10^{-7} to 10^{-2} ~~10 to 10^{6} cpm~~ µCi/cc	Cs-137* Co-60	~10^{-6} ~~10cps~~ µCi/cc	High-High ~~Low~~ High ~~High~~ Low/INOP Isolate
Reactor building cooling water system radiation monitor	Inline	~~Gamma-Scint~~ 1.2×10^{4} cpm/µCi per cc	10^{-5} to 10^{0} ~~10 to 10^{6} cpm~~ µCi/cc	Cs-137* Co-60	~6×10^{-5} ~~40cps~~ µCi/cc	~~High/Low~~ High Low/INOP
Drywell Sump Drain	Inline	~~IC~~ 30mR/hr	10^{-2} to 10^{4} ~~0.03 to 100 mR/hr~~ µCi/cc	Gross Gamma, Cs-137*	~5×10^{-2} ~~mR/hr~~ µCi/cc	High-High/~~Isolate~~ ~~Isolate~~ High Low/INOP Isolate

* Sensitivity based upon this radionuclide.

** Expected activities are estimated and are based on existing plants.

TABLE 11.5-4

RADIOLOGICAL ANALYSIS SUMMARY OF LIQUID PROCESS SAMPLES

Sample Description	Grab Sample Frequency	Analysis	Sensitivity µCi/ml	Purpose
1. Reactor Coolant				
Filtrate	Daily (a)	Gross gamma	10^{-6}	Evaluate reactor water activity
Crud	Daily (a)	Gross gamma	10^{-6}	Evaluate crud activity
Filtrate	Weekly (b)	I-131, I-133	10^{-7}	Evaluate fuel cladding integrity
Crud and filtrate	Weekly	Gamma spectrum	5×10^{-7}	Determine radionuclides present in system
2. Reactor water cleanup system	Biweekly	Gross gamma	10^{-6}	Evaluate cleanup efficiency
3. Condenser demineralizer				
Influent	Monthly	Gross gamma	10^{-6}	Evaluate ~~carryover~~ Leakage
Effluent	Monthly	Gross gamma	10^{-6}	Evaluate demineralizer performance
4. Condensate storage tank	Weekly	Gross β-γ	10^{-6}	~~Tank inventory~~ Evaluate water radioactivity
5. Fuel pool filter - demineralizer				
Inlet and outlet	Periodically	Gross β-γ	10^{-6}	Evaluate system performance
6. ~~Waste~~ LCW collector Sampling tanks (4)	Periodically	Gross β-γ	10^{-6}	Evaluate system performance
7. ~~Floor drain~~ HCW collector tanks (2)	Periodically	Gross β-γ	10^{-6}	Evaluate system performance
8. ~~Chemical waste~~ HSD Sample tanks (2)	Periodically	Gross β-γ	10^{-6}	Evaluate system performance

TABLE 11.5-4

RADIOLOGICAL ANALYSIS SUMMARY OF LIQUID PROCESS SAMPLES
(Continued)

Sample Description	Grab Sample Frequency	Analysis	Sensitivity $\mu Ci/ml$	Purpose
9. Solid waste supply tank (Evaporator bottoms)	Periodically	Gross β-γ	10^{-6}	Compar~~ison~~e activity with that determined by drum readings
10. HCW ~~Evaporator~~ distillate tank ~~tanks (2)~~ (Evaporator)	Periodically	Gross β-γ	10^{-6}	Evaluate evaporator performance
11.5.1 (4) 11. Reactor building Cooling water system	Weekly	Gross β-γ	10^{-6}	Evaluate intersystem Leakage

(a) Daily means five times per week.

(b) Performed more frequently if increase noted on daily gamma count.

TABLE 11.5-5

RADIOLOGICAL ANALYSIS SUMMARY OF GASEOUS PROCESS SAMPLES

Sample Description	Sample Frequency	Analysis	Sensitivity $\mu Ci/ml$	Purpose
1. Containment atmosphere (drywell)	Periodically and prior to entry	Gross α & β Tritium	10^{-11} 10^{-6}	Determine need for respiratory equipment
2. Offgas monitor sample	Weekly	Gamma spectrum	10^{-10}	Determine offgas activity
3. Offgas vent sample	Weekly	Gross β(a) I-131(b) Gamma spectrum	10^{-11} 10^{-10} 10^{-10}	Determine offgas system cleanup

(a) On particulate filter.

(b) On charcoal cartridge.

TABLE 11.5-6

RADIOLOGICAL ANALYSIS SUMMARY OF LIQUID EFFLUENT SAMPLES

Sample Description	Sample Frequency	Analysis	Sensitivity µCi/ml	Purpose
1. Detergent drain tanks	Batch[a]	Gross β Gamma ~~Gamma spectrum~~	10^{-7}	Effluent discharge record
2. Liquid radwaste effluent	Weekly[b]	Ba/La-140 and I-131	5×10^{-7}	Effluent discharge record
Composite of all tanks discharged	Monthly	Gamma spectrum Tritium Gross alpha Dissolved gas [c]	5×10^{-7} 10^{-5} 10^{-7} 10^{-5}	
	Quarterly	Sr/89/90	5×10^{-8}	
3. Circulating water decant line	Weekly grab of continuously collected proportional sample	Gross β Gamma Tritium	10^{-7} 10^{-5}	Effluent discharge record (backup sample)
11.5.1 (2) 4. Reactor Service Water	Weekly	Gross Gamma Tritium	10^{-7} 10^{-5}	Effluent Discharge record

(a) *If tank is to be discharged, analyses will be performed on each batch. If tank is not to be discharged, analyses will be performed periodically to evaluate equipment performance.*

(b) *Typical batch of average release. All other samples are proportional composites.*

(c) *If no discharge event occurs during the week, frequency shall be so adjusted.*

TABLE 11.5-7

RADIOLOGICAL ANALYSIS SUMMARY OF GASEOUS EFF_ENT SAMPLES

	Sample Description	Sample Frequency	Analysis	Sensitivity $\mu Ci/ml$	Purpose
11.5.1 (4)	1. Plant Vent Exhaust ~~Reactor Building exhaust plenum~~ THRU STACK *	Weekly	Gross β (a) I-131(b) ~~and~~ Ba/La-140(a)	10^{-11} 10^{-10} 10^{-9}	Effluent record
		Monthly	Gamma spectrum(a)	10^{-10}	
		Quarterly	Sr-89 and 90(a) Gross alpha(a) I-133 and 135(b) Tritium	10^{-11} 10^{-11} 10^{-10} 10^{-6}	
	~~2. Radwaste building exhaust~~	~~As above~~	~~As above~~		~~Effluent record~~
	2 ~~3.~~ Gland Steam Condenser ~~and vacuum pump off-gas~~ exhaust discharge	As above	As above		Effluent record

(a) *On particulate filter.*

(b) *On charcoal cartridge.*

11.5.1 (4) * This includes off-gas exhausts from the Reactor Building, Turbine Building, Radwaste Building and Service Building.