TEXAS UTILITIES GENERATING COMPANY

2001 BRYAN TOWER · DALLAS, TEXAS 75201

R. J. GARY
EXECUTIVE VICE PRESIDENT
AND GENERAL MANAGER

December 10, 1982

Mr. Harold R. Denton
Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NOS. 50-445 AND 50-446
TRANSMITTAL OF AMENDMENT

Dear Mr. Denton:

In accordance with 10 CFR Part 50 (c) (1) (i), Texas Utilities Generating Company, as lead Applicant and as agent for the Owners and Applicants (Dallas Power & Light Company, Texas Electric Service Company, Texas Power & Light Company, Texas Municipal Power Agency, and Brazos Electric Power Cooperative, Inc.), herewith submits Amendment 36 of the Final Safety Analysis Report (FSAR).

Enclosed are the following prescribed documents and the number of each:

FINAL SAFETY ANALYSIS REPORT - AMENDMENT 36	3 signed (and sworn) originals and 60 copies

Respectfully submitted,

R. J. Gary

R. J. Gary

RJG:tls

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
TEXAS UTILITIES GENERATING COMPANY	)	Docket Nos. 50-445
	)	50-446
(Comanche Peak Steam Electric	)	
Station, Units 1 and 2)	)	

AFFIDAVIT

R. J. Gary being duly sworn, hereby deposes and says that he is Executive Vice President and General Manager of Texas Utilities Generating Company, the Applicant herein; that he is duly authorized to sign and file with the Nuclear Regulatory Commission this Amendment to its Application and supporting documentation for Operating Licenses for the captioned facilities; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information and belief.

R. J. Gary

R. J. Gary
Executive Vice President
and General Manager

STATE OF TEXAS	)	
	)	ss
COUNTY OF DALLAS	)	

Subscribed and sworn to before me, a Notary Public in and for Dallas County, on this 8th day of December, 1982.

Glenda Benson

Notary Public

My commission expires Feb. 17, 1985.

TEXAS UTILITIES GENERATING COMPANY
2001 BRYAN TOWER · DALLAS, TEXAS 75201

R. J. GARY
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COMANCHE PEAK STEAM ELECTRIC STATION

FINAL SAFETY ANALYSIS REPORT

INSTRUCTION SHEET

The following instructional information and check list is being furnished to help insert Amendment 36 into the Comanche Peak Steam Electric Station FSAR.

Since in most cases the original FSAR contains information printed on both sides of a sheet of loose leaf paper, a new sheet is furnished to replace sheets containing superseded material. Therefore, the front or back of a sheet may contain information that is merely reprinted rather than changed. Pages with editorial changes are also provided.

Discard the old sheets and insert the new sheets, as listed below. Keep these instruction sheets in the front of the Effective Page Listing in the last volume to serve as a record of changes.

Remove Front/Back	Insert Front/Back
	VOLUME I
1.2-21/22	1.2-21/22
1.2-25/26	1/2-25/26
T1.3-1 (9 Sheets)	T1.3-1 (9 Sheets)
T1.3-2 (28 Sheets)	T1.3-2 (29 Sheets)
T1.6-1 (Sh. 17 of 17)	T1.6-1 (Sh. 17 of 17)
1A(B)-5/6	1A(B)-5/6
1A(B)-9/10	1A(B)-9/10
1A(B)-28a/28b	1A(B)-28a/28b
1A(B)-28c/28d	1A(B)-28c/28d

VOLUME IV

3.7B-71/72	3.7B-71/72

VOLUME V

3.10B-1/2	3.10B-1/2
3.10B-5/6	3.10B-5/6
3.11N-1/2	3.11N-1/2
3.11N-3/4	3.11N-3/4
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3.11B-3/4	3.11B-3/4
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T3.11B-2 (18 Sheets)	T3.11B-2
EQR 2-3	EQR 2-3
-	EQR 2-4
EQR T4-1 (5 Sheets)	EQR T4-1 (5 Sheets)
EQR T5-1 (38 Sheets)	EQR T5-1 (37 Sheets)
EQR T5-2 (19 Sheets)	EQR T5-2 (19 Sheets)

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T6.2.5A-4	T6.2.5A-4
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-	212-222
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demineralizers, and tanks. Furthermore, the amount of reactor coolant is automatically adjusted to compensate for changes in volume as a result of coolant temperature changes. The CVCS provides reactivity control by varying the boron concentration in the reactor coolant.

Water for the reactor coolant pump shaft seals is supplied from the charging pump.

The centrifugal charging pumps associated with the CVCS also serve as the high-head pumps for the ECCS. In the event of a LOCA, the CVCS is isolated except for the charging pumps, which inject borated water into the reactor core.

1.2.2.8.2 Sampling Systems

The CPSES is equipped with three sampling systems: the Primary Sampling System, the Secondary Sampling System and the Post Accident Sampling System (PASS). The Primary Sampling System serves the RCS and 36
its auxiliary systems while the Secondary Sampling System serves the feedwater and main steam systems. The PASS functions as described in FSAR Section II.B.3. These systems provide a determination of both the chemical and radiological makeup of various plant fluids. Samples drawn from radioactive sources are passed through sample coolers or delay coils, or both, as required.

1.2.2.8.3 Compressed Air Systems

Each of the two CPSES units is equipped with nonlubricated air compressors, which discharge compressed air into either the Service Air System or the Instrument Air System. There is one instrument air compressor and aftercooler for each unit, and one common instrument air compressor and aftercooler is shared by the two units.

The Service Air System provides compressed air for routine maintenance at various stations throughout the plant.

The Instrument Air System provides compressed air, which is dried and
4 filtered, for all air-operated instruments and valves.

1.2.2.8.4 Plant Ventilation Systems

To facilitate the independent control of the atmosphere in various plant areas, separate ventilation systems have been provided. The following areas are served by separate ventilation systems:

1. Containment Building

2. Auxiliary Building

3. Safeguards Building

4. Fuel Building

5. Turbine Building

6. Diesel Generator Building

7. Control Room

In addition, a Containment Purge System and a Containment Preaccess Filtration System are provided for the containment atmosphere.

1.2.2.8.5 Station Service Water System

The SSWS removes heat from the CCWS to meet the cooling requirements of the plant as follows:

The Fire Detection System uses fire, smoke, and heat detection devices located throughout the entire plant; they include the following:

a. Ionization smoke detectors

Ionization detectors are of the two-chamber-type design. The first is a reference chamber to compensate against sensitivity changes due to temperature, barometric pressure, and humidity variations. The second chamber is a sensing chamber open to the outside elements through a protective screening which permits combustion products to enter while preventing insects and foreign matter from entering and causing false alarms.

b. Thermal detectors

The thermal detectors are of the fixed-temperature or combination fixed-temperature rate-of-rise type, as required.

Continuous strip thermistor heat detectors are used to monitor the temperature of transformers, charcoal adsorber beds of the atmosphere cleanup units, and other areas where accurate measurement of temperatures for fire detection is required. The strip thermistors are of the negative coefficient of resistance type.

The detectors are strategically positioned throughout the facility to detect fires, annunciate alarms in the Control Room, sound fire alarms over the plant paging system, and indicate the location of the fire on the Control Room fire detection panel or local zone panel. In addition to these functions, detectors located in areas protected by automatic fixed extinguishing systems are used to actuate the extinguishing systems in their respective areas.

The detection system is electrically supervised against shorts and open wiring faults in the detection and alarm circuits. A short or open wiring fault causes an audible and visible trouble indication at the fire detection control panel in the Control Room. The system is designed for Class A operation in accordance with NFPA 72D. 36

2. Fire Extinguishing System

The fire extinguishing system uses portable extinguishers, in conjunction with hose stations and fixed water spray systems, as primary and secondary means of suppression. Water is supplied for the standpipes and spray systems from the SSI via an underground piping distribution system and water supply lines for each building and transformer branch from the underground loop; every branch has a post indicator gate valve for isolation of the branch or building. The main loop is divided into sections by postindicator valves to allow isolation of the loop in case of a line break. There are two redundant and completely separate trains to supply water to the system. Train A has an electric motor-driven pump whereas Train B has a diesel engine-driven pump. Water is supplied to the underground fire loop by the lead pump, the electric motor-driven pump, when the jockey pump cannot maintain the system pressure above 110 psig.

A siamese fire department connection is provided for emergency fill of the system by a fire truck or a portable auxiliary pump. This fill is used as a backup to the pumps. As required by National Fire Protection Association (NFPA) No. 24, a check valve

TABLE 1.3-1
(Sheet 1 of 9)

DESIGN COMPARISON

Chapter	Systems and Components	CPSES/FSAR Section	Significant Similarities	Significant Differences
4.0	Reactor Systems			
	Fuel	4.2	W.B. McGuire, Watts Bar and Trojan	none
	Nuclear design	4.3	W.B. McGuire, Watts Bar and Trojan	none
	Reactor vessel	4.3 4.4	W.B. McGuire and Watts Bar	none
	Thermal-hydraulic design	4.4	W.B. McGuire, Watts Bar and Trojan	none
	Reactivity control	4.6	W.B. McGuire, Watts Bar and Trojan	Part length control rods not utilized in the CPSES.
5.0	RCS and Connected Systems	5.1 5.2	W.B. McGuire and Watts Bar	none
	Reactor vessel*	5.3	W.B. McGuire and Watts Bar	none
	Reactor coolant pumps*	5.4.1	W.B. McGuire and Watts Bar	none
	Steam generators*	5.4.2	W.B. McGuire and Watts Bar	CPSES utilities model D4 steam generators in Unit 1 and model D5 steam generators in in Unit 2.
	Piping*	5.4.3	W.B. McGuire and Watts Bar	none

DESIGN COMPARISON

Chapter	Systems and Components	CPSES/FSAR Section	Significant Similarities	Significant Differences
	RHR system	5.4.7	W.B. McGuire and Joseph M. Farley	none
	Pressurizer*	5.4.10	W.B. McGuire and Watts Bar	none

*All components are designed and manufactured to applicable code edition in effect.

Chapter	Systems and Components	CPSES/FSAR Section	Significant Similarities	Significant Differences
6.0	ESF Systems			
	Containment	3.8 6.2.1	North Anna (Units 1 and 2)	North Anna containment is maintained at subatmospheric pressure.
	Containment spray system	6.2.2	North Anna (Units 3 and 4)	North Anna injection and recirculation are two separate subsystems.
	Hydrogen purge system	6.2.5	W.B. McGuire	none
	Hydrogen recombiners	6.2.5	Joseph M. Farley	none
	Containment Hydrogen Monitoring System	6.2.5	Priarie Island	none

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DESIGN COMPARISON

Chapter	Systems and Components	CPSES/FSAR Section	Significant Similarities	Significant Differences
	ECCS	6.3	W.B. McGuire and Watts Bar	Specific stress limits, requirements for fracture toughness testing, and ANS safety classifications have been added or changed.
7.0	Instrumentation and Control Systems			
	Reactor trip system	7.2	W.B. McGuire	CPSES utilizes an N-16 power monitor and an in-line Tcold measurement which replaces the RTD bypass line Thot and Tcold measurements.
	ESF actuation system	7.3	W.B. McGuire	CPSES employs an improved steamline break protection system.
	Systems required for a safe shutdown	7.4	W.B. McGuire	none
	Safety-related display instrumentation	7.5	W.B. McGuire	Actual physical configuration differs according to customer design philosophy.

DESIGN COMPARISON

Chapter	Systems and Components	CPSES/FSAR Section	Significant Similarities	Significant Differences
	All other instrumentation systems required for safety	7.6	W.B. McGuire	CPSES has incorporated a reactor coolant system cold overpressure control system.
				CPSES does not employ the following systems: annulus ventilation, containment air return and hydrogen skimmer, ice condenser instrumentation, groundwater drainage and containment pressure control.
	Control systems not required for safety	7.7	W.B. McGuire and Fort Calhoun I	Load rejection capability differs according to Applicant; part length control rods are not utilized in the CPSES.
8.0	Electric Power Systems			
	Onsite ac power system (diesel generators)	8.3.1	Watts Bar 1 & 2	none
	Onsite dc power system	8.3.2	Vogtle 1 & 2	none

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DESIGN COMPARISON

Chapter	Systems and Components	CPSES/FSAR Section	Significant Similarities	Significant Differences
9.0	Auxiliary Systems			
	New-fuel storage system	9.1.1	Watts Bar	Watts Bar has storage racks for one full core as compared to two-thirds core for CPSES.
	Spent-fuel storage system	9.1.2	Watts Bar	Watts Bar has one- and two-thirds core storage for each pool as compared to 1116 fuel assemblies total storage for CPSES. CPSES uses 16" center to center spacing fuel racks.
	Spent-fuel pool cooling and cleanup system	9.1.3	Surry Power Station (common system for two units)	Surry has system service for one common pool with a distinct skimmer system as compared to two pools serviced by CPSES system.
	Station service water system (SSWS)	9.2.1	North Anna Units 1 and 2	The ultimate heat sink is site-related.
	CCWS	9.2.2	Surry Power Station	Surry has four pumps total with three pumps used for cooldown as compared to four-pump total with two used for cooldown for the CPSES.

DESIGN COMPARISON

Chapter	Systems and Components	CPSES/FSAR Section	Significant Similarities	Significant Differences
	CVCS	9.3.4	W.B. McGuire and Trojan	New ANS safety classifications have been added, and some classifications have been changed.
	Diesel Generator Systems	9.5.4	Watts Bar	Watts Bar Utilities two 550 gallon day tanks per generating unit compared to one 2,160 gallon day tank in CPSES.

DESIGN COMPARISON

Chapter	Systems and Components	CPSES/FSAR Section	Significant Similarities	Significant Differences
10.0	Steam and Power Conversion Systems			
	Steam generator blowdown system (SGBS)	10.4.8	W. B. McGuire	CPSES flow rate is 640 gpm. W.B. McGuire flow rate is 150 gpm. CPSES is designed to meet NSSS water chemistry requirements with AVT chemistry program. CPSES processed effluent flows to heater drain tank under normal conditions. Processed blowdown mixed with condensate is used to cool hot blow-down fluid. W.B. McGuire uses a closed cooling water system to cool the hot blowdown.
	Auxiliary Feedwater System	10.4.9	Watts Bar	none
11.0	Radioactive Waste Processing Systems			
	Source terms	11.1	W.B. McGuire	Differences ex-ist because of ANSI N237 standard.

TABLE 1.3-1
(Sheet 8 of 9)

DESIGN COMPARISON

Chapter	Systems and Components	CPSES/FSAR Section	Significant Similarities	Significant Differences
	Liquid waste processing system (LWPS)	11.2	Watts Bar	CPSES incorporates a reverse osmosis system to process laundry and hot shower waste. Watts Bar has no such system.
				CPSES has a waste monitor tank demineralizer to further reduce activity in the waste monitor tanks. Watts Bar does not have this feature.
				CPSES utilizes separate monitor tanks downstream of the processing equipment in each drain channel. Watts Bar utilizes three common monitor tanks.
				Watts Bar incorporates facilities to neutralize and process condensate demineralizer regenerants in their liquid waste management system. CPSES utilizes a powdered resin condensate demineralizer.
	Gaseous waste processing system (GWPS)	11.3	Joseph M. Farley	none
	Solid waste processing system (SWPS)	11.4	Beaver Valley	none

DESIGN COMPARISON

Chapter	Systems and Components	CPSES/FSAR Section	Significant Similarities	Significant Differences
	Process and effluent radiological monitoring (PERMS) and sampling system	11.5		CPSES utilizes a digital, dedicated distributed microprocessor based system.
12.0	Radiation Protection			
	Design considerations	12.1.2	Watts Bar	none
	Facility design features	12.3.1	Watts Bar	none
	Shielding	12.3.2	Watts Bar	none
14.0	Initial Test Programs	14.0	W.B. McGuire	The section for this plant has been written to conform to the NRC regulatory guides for planning, testing, and startup programs.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
I. Structures		
Category I Structures (other than containment)	3.8	Leak chase system behind liner of outdoor Category I tanks has been eliminated.
		The requirement for roof blow out panels above the main steam lines in the Safeguards Building has been deleted.
Control Room	3.11B	Limiting environmental conditions have been reduced to 80°F and 60 percent relative humidity.
	6.4 3.11B	Positive pressure of 1/2 inch water gauge changed to 0.1 inch water gauge during an accident.
Containment Systems	3.8	The containment external pressure design has been changed from a 3 psi differential pressure to a 5 psi differential pressure.
		The containment internal structure through liner anchors was eliminated.
		The containment dome liner has been increased from 3/8 inch thickness to 1/2 inch.
		The containment liner paint has been changed.
		The break flow in the reactor cavity analysis is limited to 144 in^2 by pipe whip restraints instead of the previous 150 in^2.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
		Cathodic protection not required for containment liner, reinforced steel and other such steel encased in concrete.
		Containment penetration sleeves changed from A-333-70, Grade 6 to: 1) SA-333, Grade 6 for sleeve sizes 20 in. and smaller 2) SA-516, Grade 70 or SA-537, Class 2 to sleeve sizes 20 in. and larger
		Inside weld no longer utilized on cold pipe penetration.
		Sleeve with inside weld and guard pipe on hot pipe penetration replaced with sleeve without inside welds or guard pipe.
	3.11B	Maximum operational temperature increased to 120°F.
	6.2.1	The reactor cavity analysis was redone using new support criteria and restraint design.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
	6.2.4	Changes in signals used to actuate the containment isolation system to reflect changes in WNES protection system.
		Addition of bellows expansion joints to the valve isolation tanks.
	6.2.5	The containment liner paint has been changed from a primer and top coat that was zinc free to inorganic zinc as a primer in phenolic topcoat.
		The use of aluminum and zinc inside containment is now permitted.
		Remote thermal conductivity hydrogen analyzers are replaced by in-containment electrochemical sensors and microprocessor based analyzers located in the control room. 36
		The containment free volume has been changed from 2.9×10^6 cubic feet to 2.985×10^6 cubic feet.
	6.5	Containment sump pH limits changed from 9.0 to 8.6 and 9.3 to 9.5.
Missile Shield	3.5	The CRDM roll away missile shield is constructed of steel instead of reinforced concrete.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
Safe Shutdown Impoundment	9.2.5	The presence of fractures in the SSI limestone foundation material was determined to be acceptable if the fractures occur above elevation 769.5 (spillway level), and if the fractures would not cause piping of core material. Clarification of the term "unweathered" to mean material that is firm, hard, and of appropriate strength parameters; the presence of slight oxidation resulting in some color change may not be an indication of weathering.
Shielding	12.1	Deleted the provisions for: 1. Local alarms at entries to Zone IV areas 2. Local and remote alarms at entries to Zone V areas has been deleted. The minimum density for ordinary concrete has been changed from 2.33 g/cc to 2.26 g/cc The secondary shield wall no longer surrounds the pressurizer and the thickness has been increased to 2 ft 9 in.
Tornado Characteristics	3.5	Tornado Missile Characteristics have been changed.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
Piping Systems, Handling Equipment and Tornado Venting Components	3.2 and Appendix 17A	Portions of systems or components are designed as Seismic Category II components.
Non-nuclear Safety Piping Systems or Components	3.2 and Appendix 17A	Portions of systems or components are designed as Class 5 components. 10
Auxiliary Feedwater System	3.6B	The system is identified as high energy fluid system.
II. Reactor Systems		
Fuel	4.2	The reactors are fueled with 17 x 17 fuel assemblies in lieu of 15 x 15 fuel assemblies.
Reactor internals	3.1, 7.7	The use of part length control rods has been removed from the CPSES design.
	4.5.2	The reactor internals have been modified to accept 17 x 17 fuel assemblies.
RCS pressure boundary	5.2	Flux core welding on stainless steel piping components fabricated in the shop are permitted. The allowable heat input range is 15 to 100 KJ/in.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
		The use of the following as valve materials is now permitted for valves 2.5 in. and larger: bodies: SA 351 grade CF8 and CF8M bonnets: SA 351 grade CF8M with nickel plating or stellite hardfacing closure bolting: SA 564 Type 630 and SA 193 grade B6 closure nuts: SA 193 grade B6 and SA 194 grade 8M The use of the following as valve materials is now permitted for valves 2.0 in. and smaller: bodies: SA-351 Gr. CF8M SA-182 F 316 bonnets: SA-182 F 316 discs: A-567 Gr. 1 (Stellite 21) stems: SA-564 Type 416 (Non-pressure retaining applications) The lower heat input range limit for welding processes of austenitic stainless steel have been deleted, and the use of automatic gas tungsten arc-cold wire processes is allowed. The maximum allowable heat input for this process is 45 KJ/in.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes	
	5.7, 7.6	An RCS cold overpressure control system is employed to provide for the mitigation of potential cold overpressurization transients, utilizing existing power operated relief valves with modifications to their actuation logic.	10
Steam Generators	5.4.3	CPSES utilizes model D5 steam generators in Unit 2. Thermal sleeves in reactor coolant loop branch nozzles have been deleted.	
RHR systems	3.6	The RHR system is no longer identified as a high-energy fluid system located outside the containment.	
Safety Injection System	3.1	Change in SIS signal from coincident low pressurizer pressure and water level to low pressurizer pressure.	

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
III. Electrical Systems:		
Reactor Trip System	7.2	The reactor trip on low feedwater flow has been replaced by a low-low steam generator level trip.
		The RTD bypass line T hot and T cold measurements have been replaced by an N-16 power monitor and an inline T cold measurement CPSES utilities 4 section power range neutron detector assemblies. 10
Engineered Safety Feature Systems	7.3	An improved steamline break protection system has been incorporated where safety injection and steamline isolation are initiated from low compensated steamline pressure.
CCW, SSW	7.3	Recirculation valves are now flow controlled, rather than pressure controlled
Post-Accident Monitoring	7.5	SSW and CCW system parameters are not considered post-accident monitors
Hot Shutdown Panels	7.4	Indicators on the hot shutdown panel are no longer considered "Post accident monitors" and are treated as non-Class IE.
Electrical systems	8.1	Non-safety-related loads removed from Class 1E batteries and reassigned to ±125-V non-Class 1E batteries.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
		A static switch has been added to the BOP static uninterrupt. Power supply system inverter to accommodate the 120 V supply.
		138 kV DeCordova line previously was directly fed from DeCordova substation. Now it is transferred through switching station located near DeCordova SES.
		Motor operated disconnect switch is added to the 138 kV line at CPSES to isolate the startup transformer XST1 for maintenance.
	8.3	Added start signal to diesel generator on undervoltage of the preferred offsite power source.
		The condition of diesel generator trip on bus-fault is removed. Now diesel generator does not trip on bus fault condition. However, will trip on overspeed.
		Some events which trip diesel generator excluding accident condition, are added and deleted as follows:
		1. Bus fault signal to trip the generator breaker is deleted and following signal are included to trip the diesel generator.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
		2. Lube oil high temperature
		3. Generator over current
		Diesel generator trips due to following events, during "no accident" conditions have been added 10
		a. Generator Negative Sequence
		b. Generator Ground
		c. Generator Field Ground Trip
		Diesel generator rating modified from a 2000 hour rating to a 2 hour short term rating
		Plant computer removed from DC safeguards bus and added to 125/250 V bus. Both DC safeguard batteries are sized identically.
		Back-up (Alternate) 120 V ac instrument power supply to the 118 V Class IE uninterruptible ac bus is changed from 120 V single phase chassie emergency Lighting System to the 120 V ac supply from Class IE MCC via a 480/120 volt single phase Class 1E bypass transformer. 1
		Static switch is added in each BOP SUP system to allow automatic transfer from inverter to back-up (alternate) 120 V ac power source.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
		The air circuit breakers are molded case type and fusible switches are provided on some switchboard feeders (instead of circuit breakers) for coordination purposes.
		Unit substation transformer was changed from 1500 kVA to 2000/1666 kVA. 33
		The backup supply for the 120-V NSSS instrument buses was originally taken from the 120-V single-phase emergency lighting system. These buses are now fed from the Class 1E motor control centers via a 480/120V single-phase Class 1E bypass transformer.
		Electric penetration centerline spacing changed from 2 ft 6 in. to 2 ft 2 in.
		Total integrated dose that electrical equipment is subjected to is increased to 2.0 x 10^8 rads (1.5 gamma and 0.5 beta).
		Added ±24 volt dc system for the turbine generator systems

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
IV. Mechanical Water Systems:		
Component Cooling	9.2	The following equipment is no longer served by the CCWS: 1. Safety injection pump seal cooler 2. Centrifugal charging pump seal cooler 3. SGBS heat exchanger The following equipment is now served by the CCWS: 1. Instrument air aftercoolers 2. Chilled water system condensers 3. Control room air-conditioning condensers 4. Pump added for component cooling water drain tank in safeguards building 5. Control room alarm added if makeup water added to surge tank The CCWS heat exchangers will utilize Cu-Ni tubes instead of stainless steel. Component Cooling Water pump normal supply temperature is increased to 115 F. 10

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
Condensate Storage Facility	9.2.6	Storage capacity increased from 480,000 gallons to 500,000 gallons.
		Secondary system water storage decreased from 240,000 gallons to 224,000 gallons.
		Reserve auxiliary feedwater capacity increased from 240,000 gallons to 276,000 gallons.
Containment Spray System	3.2	Containment spray system nozzles will not bear an "N" stamp
		Change from one spray header to seven.
	6.5	Minimum spray fall height changed from 117 feet to 115 feet 9 inches.
		Eductor calibrated flow changed from 37.5 gpm to 45 gpm.
		Containment spray delivery lag time changed from 40 seconds to 60 seconds.
		Containment sprayed volume changed from 2.528 x 10^6 cubic feet to 1.725 x 10^6 cubic feet. (From 87% to 57.8%)

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
Piping systems	3.6	The pipe whip analyses are now performed by the ANSYS computer program of Swanson Analysis System, Inc.
	3.9	The addition of stress limits for seismic Category I piping.
Potable and sanitary water system	9.2.4	Sewage plant effluent is discharged to Squaw Creek Reservoir instead of the evaporation ponds.
		Domestic water storage tank is designed to ASME B&PV Code Section III Code Class 3 requirements. 10
		Capacity of the sewage treatment plant is increased from 5000 gpd (normal operation) to 10,000 gpd (normal operation) and to 30,000 gpd during the construction stage.
		Capacity of the hypochlorinator has been increased.
Station Service Water System	9.2.1	The following changes have been made:
		1. The SSI study resulted in an increase of the accident inlet temperature of the service water to 105°F
		2. Revision of Service Water Intake structure

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
		3. Service water pump design temperature increased from 95°F to 120°F
		4. Traveling screen spray piping changed from carbon steel to copper alloy
		5. The SSWS no longer removes heat from the control room air-conditioning system
		6. The SSWS has been added as a backup water supply to the Auxiliary Feedwater System
		7. The non-safety-related train of the SSWS has been deleted
		8. Automatic strainers have been 26
		9. Piping for sizes up to and including 3-1/2 in. is made of stainless steel. For piping 4 in. and larger, carbon steel is used with a plasite 7122 lining instead of all carbon steel piping. 10
Valves: Feedwater Isolation Valves	6.2.4	The feedwater isolation valve will no longer fail closed upon loss of power. Closure will be upon energizing redundant solenoid valves.
Valves: General	3.1 6.2.4	Change from air-operated valves failing closed to air and solenoid-operated valves failing in the direction of greatest safety.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
Materials	6.1B	The control of delta ferrite to reduce the susceptibility of stainless steel welds to hot cracking is no longer considered for ANS Safety Class 3 components. The requirement that the total leachable chloride and fluoride content of clean elastomers and plastics placed over all openings in components fabricated from austenitic stainless steel be limited to 15 and 10 ppm, respectively, has been deleted.
V. Fuel Storage and Handling Systems:		
Fuel storage and handling system	9.1	The following changes were made to the spent fuel storage system: 1. An increase in total spent fuel storage space from 800 to 1116 spent fuel assemblies 2. A decrease in center-to-center spacing from 21 to 16 in. 3. An increase in Keff from 0.90 to 0.95 for spent fuel assemblies if immersed in unborated water. Purification loop was added to the refueling cavity.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes	
		The number of dry storage racks has been increased from 129 to 132.	
		An increase in Keff from 0.90 to 0.98 for new fuel assemblies if flooded with unborated water.	
		The dry cask loading concept is eliminated.	10
		The design of the spent fuel pool cooling heat exchangers is changed from horizontal to vertical.	
VI. Auxiliary Systems:			
Auxiliary Feedwater System	9.2.1	The SSWS has been added as a backup water supply.	
	10.4.9	Manually controlled heating systems have been changed to automatic systems	
		The Auxiliary Feedwater is supplied to the secondary side of the steam generator through a separate upper Auxiliary Feedwater nozzle.	10
Condensers	10.4.1	Design basis for auxiliary condensers was revised.	
		Instrumentation for main condenser was modified to include high differential pressure alarm and temperature recorder in the control room.	

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
		Isolation valves were added to the auxiliary condenser and turbine plant heat exchanger with change from a motor operator to a gear operator.
		The number of condensate pumps in the condenser hotwells has been changed from three to two half-capacity pumps.
Vacuum pumps	10.4.2	The number of mechanical vacuum pumps was changed from two to three. During startup, discharge is to the atmosphere; during normal operation, discharge is through the charcoal filter system.
Circulating water system	10.4.5	Design basis was revised.
		Pressure-differential alarm on the screen is annunciated in control room.
		Butterfly valves on the turbine plant cooling water heat exchangers were added.
		Circulating water pipes are made from coated carbon steel instead of concrete or plastic.
Condensate Cleanup System	10.4.6	Backwash recovery and powdered res[illegible] handling added to system design

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
Feedwater heaters	10.4.7	Separate external drain coolers have been added to the 6A and 6B heaters.
Steam Generator Blowdown System	10.4.8	Treatment capacity of the system has been increased from 50 to 640 gpm.
		Blowdown cooled by condensate or condensate/blowdown instead of component cooling water
		Blowdown now recycled to heater drain tank at high temperature
		The processing of spent resin has been changed from regeneration to flushing into the storage tank prior to drumming.
		Change in bed processing as a result of elimination of the regeneration mode.
		SGBS heat exchanger has been changed from ANS Safety Class 3 to NNS.
Fire Protection System	9.5.1	The Fire Protection System has been completely upgraded to address current NRC Regulatory Positions.
Lighting systems	9.5.3	Lighting system powered from localized motor control centers in lieu of a central power substation

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
		Instrument room and primary plant egress routes have been added to the list of areas provided with DC emergency lighting
		Class IE lighting cables are no longer routed through the turbine building
		Lighting fixtures and lamps of the AC essestial lighting system and DC emergency lighting system are no longer Class IE although the remainder of these systems remain Class IE
		The AC Essential Lighting System and the DC emergency lighting system have been designed independent only with respect to electrical failures, not all failure modes.
Diesel Generator Fuel Oil Day Tank	9.5.4	Fuel oil supply increased to 2160 gallons from 3 hours

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
The Diesel Generator Fuel Oil Storage and Transfer System	9.5.4	The diesel fuel oil storage tank is equipped with: a) A fill line with a shutoff valve. b) A perforated fill line which runs to within 2 feet of the bottom of the tank. c) A dirt and water collector. d) Hold down straps embedded in a concrete foundation. e) A return line from the fuel oil day tank.
Diesel Generator Combustion Air Intake and Exhaust System	9.5.8	Exhaust relief valve discharge changed to vertical. 10
Main Steam Supply System	10.2	Design temperature reduced to 541.5°F from 544.6°F. A vibration trip and alarm have been added. A MSR high level trip and alarm have been added.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
	10.3	The main steam relief valves are relocated upstream from the safety valves to gain accessibility to these valves.
		The main steam isolation valve materials have been changed.
		Each steam line of auxiliary feedwater pump turbine is provided with an air-operated stop valve instead of motorized stop check valve.
		Phosphate was deleted from use and condensate polishing added.
		Design pressure changed to 1200 psig.
		Safety class piping extended to the first moment restraint beyond the Main Steam Stop Valves.
		Power operated relief valves now set to open at 1130 psia.
		The Main Steam Stop Valve integral bypass valve has been upgraded to Code Class 1 from Code Class 2
		Lowest and highest safety valve set pressures changed to 1200 and 1250 psi.
Steam Generators	10.3	Steam generator safety valves rated to pass 105 percent rather than 110 percent of flow.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
Compressed Air System	9.3.1	Air accumulators are upgraded to ANS Safety Class 3 and designed to ASME B&PV Code Section III Class 3 Requirements. 10
VII. Ventilation Systems:		
Containment Ventilation Systems	6.2.6	The hydrogen purge system filter efficiencies have been increased from 90 to 95%.
	9.4	CRDM cooling is accomplished by two 100-percent CRDM exhaust fans per containment instead of three 50-percent exhaust fans. The following changes have been made in the neutron detector well cooling system: 1. Use of chilled water instead of component cooling water as the cooling medium of the neutron detector well cooling system 2. A reduction in airflow from 24,000 ft^3 per containment to 13,100 ft^3 per containment for the neutron detector well cooling system due to the increase in the temperature gradient across the cooling coils

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
		Heating, ventilating, and air-conditioning (HVAC) dampers are manufactured in accordance with ANS Safety Class 3, 10 CFR Part 50 Appendix B, and manufacturers' standards rather than ASME II.
		To provide for adequate cooling during accident conditions, centrifugal chillers are used for the emergency fan coil units instead of service water.
		Increase size of containment coolers to absorb heat load from the CRDM.
		Only the containment purge exhaust ductwork is classified ANS Safety Class 3.
Other ventilation systems	3.1	Chlorine detectors added to the control room ventilation system
	9.4.1	Roof-mounted fans changed to in-duct fans for improved missile protection
		Reduction in outside air intake quantity for control room
		Control room pressurization capacity increased from 150 cfm to 800 cfm.
		Control room air-conditioning units increased to 100 percent redundancy from 50 percent redundancy.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
		Greater than 10 percent of the total air quantity is directed through the control room emergency filtration units.
		Four nuclear safety related vaneaxial fans are utilized in each diesel generator room instead of one.
	9.4.1	The concept of lead system and standby system is deleted. Emergency recirculation will activate both Train A and Train B components. 10
	9.4.2	Auxiliary safeguards buildings ventilation supply distribution systems are now classified as seismic Category I
	9.4.3	Auxiliary building ventilation supply distribution system classified as seismic Category I
	9.4.7	Electrical area ventilation system contains only two 50-percent-capacity fans.
		Uncontrolled access area ventilation system contains two 50-percent-capacity fans.
		Battery rooms have in-duct fans instead of roof-mounted fans.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
		Office and service area ventilation systems have 50-percent redundancy; these systems are not required to satisfy the single-failure criterion and are nonessential to the safe shutdown of the reactor.
	General	Auxiliary, safeguards, fuel-handling, and containment purge supply and exhaust are incorporated into primary plant ventilation system.
		Summer design condition of diesel generator building changed from 104°F to 122°F
		Filter bed thickness increased from 2 inches to 4 inches.
		Air inlet design temperature to the hydrogen purge exhaust unit changed from 120°F to 250°F.
		Instrument air accumulators are now provided for the control room HVAC system.
VIII. Steam & Power Conversion Systems:		
Component and subsystem design	5.4	The steam generators for Unit 2 have been changed from D4's to D5's.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes
Steam and power conversion system	10.1	The steam dump system to the condenser from the steam generator is not a safety-related feature included in the steam and power conversion system.
Turbine-generator	10.2	Added Occupational Safety and Health Act (OSHA) to codes and standards.
		The worst case accident, a failure of the cast stage of the low pressure turbine rotor, is analyzed. 10
		The following have been added to the list of events that initiate a turbine trip: 1. Reactor trip 2. Steam generator high-high level 3. Safety injection 4. Generator trip 5. Moisture separator high level (each MSR) 6. Excessive vibration during speed operation from 900 rpm until unit is synchronized.

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes	
		The following have been deleted from the list of events that initiate a turbine trip: 1. Low flow of stator coolant The following have been added to the list of events that initiate a generator trip: 1. Loss of lube oil pressure 2. Low flow of stator coolant	
IX. Waste Processing Systems:			
Liquid Waste Processing System	11.2	The following equipment has been added to the LWPS: 1. Laundry reverse osmosis system 2. Laundry holdup and monitor tanks 3. Laundry holdup and monitor tank pump 4. 10,000 gallon floor drain tank 2 5. 30,000 gallon floor drain tank 3	
Solid Waste Processing System (SWPS)	11.4	Utilizes an ATCOR proprietary cement solidification system per ATCOR Topical Report No. ATC-132A.	10
X. Sampling & Monitoring Systems:			
Process Sampling System	9.3.2	Phosphate analyzers replaced by sodium ion analyzers. System changed to non-nuclear safety related and non-seismic downstream from the external containment isolation valves.	10
	9.3.2, II.B.3	Inclusion of Post Accident Sampling System.	36

DESIGN CHANGES SINCE PSAR SUBMITTAL

Systems or Components	CPSES/FSAR Section	Changes	
Containment Hydrogen Monitoring System	6.2.5	Remote thermal conductivity analyzers replaced by in-containment electrochemical sensors and microprocessor-based analyzers located in the control room.	36
Steam generator liquid sample monitors	11.5.2	Steam generator liquid sample monitors reduced from four to one to simplify design	
Radiation Monitoring System	11.5	Changed from Analog to digital, microprocessor based system. Process monitors have been added to liquid waste processing and auxiliary condensate systems and to safeguards, auxiliary and fuel building vent ducts. Area monitors have been added to the fuel building to form a criticality alarm system.	
		The auxiliary vent stack is no longer capable of being monitored by the containment air monitor	
		Radiation monitors have been modified and have been added for accident monitoring.	31
	12.3	Change from analog system to digital microprocessor based system	
		Radiation monitors have been added for accident monitoring.	31

TABLE 1.6-1 (Sheet 17 of 17)

Report	Reference Section(s)	Review Status	
"Reactor Vessel Head Drop Analyses," WCAP-9198, January 1978	9.1	U	13
"Westinghouse ECCS Evaluation Model, February 1978 Version," WCAP-9220-P-A (Proprietary Version), WCAP-9221-P-A (Non-Proprietary Version), February 1978.	15.6	AE	
"Westinghouse Emergency Core Cooling System Evaluation Model" - Modified October 1975 Version," WCAP-9168 (Proprietary) and WCAP-9169 (Non-Proprietary), September 1977.		U	36

starting of 12-13 cycles after closure of the diesel generator circuit breaker. However, the objective of the first load group and subsequent load groups is not affected. For details see Section 8.3.

Regulatory Guide 1.10

Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures

Discussion

Testing and sampling of Mechanical (Cadweld) Splices in Reinforcing Bars of the CPSES Concrete Containment Structure complies with the requirements of Revision 1 (1/2/73) of this regulatory guide. For other seismic Category I concrete structures, the testing and sampling of Mechanical (Cadweld) splices complies with the requirements of this guide except that the location of all splices are not recorded and shown in as-built drawings. 36 36

Also refer to Section 3.8. 8

Regulatory Guide 1.11

Instrument Lines Penetrating Primary Reactor Containment

Discussion

The CPSES instrument lines penetrating primary reactor containment comply with the requirements of Safety Guide 11 (3/10/71), as described by Section 7.3.1.1.2 and 6.2.4.1.4.

Also refer to Appendix 1A(N).

Regulatory Guide 1.12

Instrumentation for Earthquakes

Discussion

The installation of instrumentation for earthquakes in the CPSES plant is in conformance with the requirements of Revision 1 (4/74) of this regulatory guide.

8 Also refer to Section 3.7B.4.

Regulatory Guide 1.13

Spent Fuel Storage Facility Design Basis

Discussion

The design of the CPSES spent fuel storage facility complies with Revision 1 (12/75) of this regulatory guide except that the air filtration system is not actuated by a high radiation level alarm. Instead it operates continuously. Refer to Sections 3.5, 3.8.4, 9.1.2, 9.1.3, 9.1.4 and 9.4.2 for details.

Regulatory Guide 1.14

Reactor Coolant Pump Flywheel Integrity

Discussion

Refer to Appendix 1A(N).

Regulatory Guide 1.15

Testing of Reinforcing Bars for Category I Concrete Structures

Regulatory Guide 1.21

Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants

Discussion

Measuring, evaluating and reporting of radioactive materials released from CPSES will meet the recommendations of Revision 1 (6/74) of this regulatory guide.

Also refer to Section 11.5. 8

Regulatory Guide 1.22

Periodic Testing of Protection System Actuation Functions

Discussion

Refer to Appendix 1A(N).

Regulatory Guide 1.23

Onsite Meteorological Programs

Discussion

The meteorological monitoring program at CPSES complies with the recommendations of Safety Guide 23 (2/17/72).

Also refer to Section 2.3. 8

Regulatory Guide 1.24

Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure

Discussion

The analysis of the radiological consequences of the radioactive gas storage tank failure accident presented in Section 15.7.1 complies with the requirements of Safety Guide 24 (3/23/72) except that only gamma radiation contribution is taken into account in the determination of whole body exposures.

Regulatory Guide 1.25

Assumptions used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors

Discussion

The analysis of the radiological consequences of the fuel handling accident inside the Fuel Building presented in Section 15.7.4 complies with the requirements of Safety Guide 25 (3/23/72) except as follows:

1. Instead of iodine adsorber efficiencies of 90 percent for inorganic species and 70 percent for organic species given in the
36 Regulatory Position C.1.J., the efficiencies given in Table 2 of Regulatory Guide 1.52 are used.

2. Only gamma radiation contribution is taken into account in the determination of whole body exposures.

14.2.7 has been revised to delineate tests to be performed on the feedwater control system.

3. Appendix A subparagraph 1.d.3

The accident analysis concerning the inadvertent depressurization of the reactor coolant system is discussed in FSAR Section 3.9N.1.1, Upset Condition number 5. Subsection 5.b lists the condition of the inadvertent opening of one pressurizer power operated relief valve (PORV). The analysis states that the limiting case is the actuation of the pressurizer safety valve. This is classified as an Upset Condition with no operational impairment. The design parameters listed in Table 5.4-16 of the FSAR indicate that the relieving capacity for the pressurizer power operated relief valve is one-half the capacity of the pressurizer safety valve, 210,000 lb/hr vs 420,000 lb/hr. Therefore there is no intention of performing capacity tests of the pressurizer power operated relief valves during the startup phase. 11 Q400.3

Testing of these valves is listed in Table 14.2-2, Sheet 53, of the FSAR as a preoperational test.

The accident analysis covering the opening of a steam generator power operated relief valve is similar. The analysis is described in FSAR Section 15.1.4. The accident analysis uses a value of 964,800 lb/hr @ 1200 psia for the steam flow. Table 10.3-3 in the FSAR lists the steam flow of the steam generator power operated relief valve at 420,000 lb/hr at 1107 psia. Although the design pressures are different, the conservatism allowed by the accident analysis parameters indicates no need to test the capacities of the steam generator power operated relief valves during the startup program.

The testing of the steam generator power operated relief valves is presented in FSAR Table 14.2-2, Sheet 49.

4. Appendix A subparagraph 1.k.2,3

The equipment identified in the above paragraphs of Regulatory Guide
1.68 is calibrated and functionally tested as part of the instrument
calibration program for the TUGCO Chemistry and Health Physics
18 section. The calibration and functional testing is performed and
documented in accordance with approved station calibration
procedures. Therefore, TUGCO Startup will not perform additional
testing, in the form of a preoperational test, on this equipment.

5. Appendix A subparagraph 1.n.11

The design of the Instrument Air System as referenced in FSAR Section 9.3.1.1 is not nuclear safety class because it serves no safety function required by Appendix A of 10 CFR 50. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors" and Regulatory Guide 1.80, "Preoperational Testing of Instrument Air Systems" both specifically state their scope is limited to safety-related systems. Therefore, using the stated design there appears to be no requirement to include the instrument air system in the preoperational testing program. However, as identified by Section 14.2.1, this system is intended to be acceptance tested in accordance with applicable startup administrative procedures.

6. Appendix A subparagraph 1.o.1

36 The vendor has performed applicable load testing on the head lifting
and internals lifting devices for 125% static loads.

36

7. Appendix A subparagraph 4.t

The design differences between Comanche Peak and the reference plants are described in FSAR Section 1.3. Since there are no design differences in the fuel, nuclear, thermal-hydralic or reactor coolant system designs, it is our position that the natural circulation test performed at Trojan demonstrates that the natural circulation capabilities of similar Westinghouse four loop NSSS systems is contained in WCAP-8460, "Natural Circulation Test Report of Zion Station Unit 1." We do not intend to compare any flow (without pumps) and temperature data with that from a reference plant since no design differences exist between Comanche Peak and the reference plant that affect the natural circulation capabilities.

Regulatory Guide 1.68.1

Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants

Discussion

This regulatory guide is not applicable to the CPSES

Regulatory Guide 1.68.2

Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water Cooled Nuclear Power Plants

Discussion

The testing activities conducted as a part of the startup test program
8 will comply with the applicable requirements of Revision 1 (7/78) of this
Q423.10 regulatory guide.
Q400.3

Also refer to Section 14.2.

Regulatory Guide 1.69

Concrete Radiation Shields for Nuclear Power Plants

Dicussion

The regulatory guide states that ANSI N101.6-1972, Concrete Radiation Shields, is considered applicable to shielding structures for nuclear power plants, subject to certain conditions. These conditions are stated in Regulatory Positions Nos. 1-8 of the regulatory guide. The guidance provided in this regulatory guide, dated December 1973, is adhered to as follows:

The seismic instrumentation described previously is provided for Unit 1 of the CPSES, as allowed by Section 4.4 of ANSI N18.5. All seismic instrumentation conforms to the requirements of ANSI N18.5-1974 and is calibrated to measure and record accelerations in the range applicable to the earthquake conditions of the CPSES.

3.7B.4.2 Location and Description of Instrumentation

The seismic instruments enumerated in Subsection 3.7B.4.1 are situated at the following locations:

1. The triaxial time history accelerograph has three triaxial acceleration sensors. The first is located at the top of the Containment Building mat, the second is located on the exterior face of the Containment Building wall at elevation 1000 ft 6 in., and the third is located in the "free field." These sensors have the function of sensing the absolute seismic accelerations in two horizontal orthogonal directions and in the vertical direction at the Containment foundation, on the Containment structure, and in the "free field." The data collected by the sensors are transmitted to the recorder.

   In addition, a seismic trigger is installed on the Containment Building foundation and is connected to all three sensors and the recorder. The function of the seismic trigger is to start the time history accelerograph whenever a preset threshold is exceeded for any of the three directions. A time delay device keeps the entire system operating for five seconds after the last motion above the threshold of the trigger.

   The triaxial time history accelerograph also includes a magnetic tape recorder and a playback unit which records the signals for accelerations versus time in the three orthogonal directions and provides immediate visual display of the recorded time histories on a strip chart.

2. The triaxial peak accelerograph is a passive instrument that requires no power source to sense motion and to record data. It is used at one of the steam generators, on the reactor piping, and on one of the component cooling water heat exchangers in the Auxiliary Building for recording the peak seismic accelerations on the equipment and systems.

36

3. The passive response spectrum recorder is used on the Containment Building foundation, on a floor slab adjacent to one of the steam generator compartment walls, and near one of the safety injection pumps in the Safeguards Building for recording seismic responses at these locations for different preset frequencies.

4. The response spectrum switch is located on the Containment Building foundation and is used for transmitting to the Control Room a signal whenever the response in any of the three orthogonal directions exceeds a preset value.

5. The seismic switch is located on the Containment Building foundation and is used for transmitting to the Control Room a signal whenever the acceleration in one of the three orthogonal directions exceeds a preset value.

A schematic diagram indicating the locations of all seismic instrumentation is presented on Figure 3.7B-54.

3.7B.4.3 Control Room Operator Notification

In case of any seismic activity of sufficient intensity to activate the seismic instrumentation, the Control Room operator is alerted by means

3.10B SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10B.1 SEISMIC QUALIFICATION CRITERIA

The seismic Category I instrumentation and electrical equipment which require seismic qualification are as follows: 3

1. 6900-V switchgear (nuclear-safety-related)

2. 6900-V to 480-V transformers (associated with nuclear-safety-related buses)

3. 480-V switchgear (nuclear-safety-related)

4. 480-V motor control and motor control centers (nuclear-safety-related)

5. 125-V station batteries and racks

6. 480-VAC to 125-VDC battery chargers

7. 125-VDC panels and switchboards (nuclear-safety-related) 3

8. 125-VDC to 120-VAC inverters (nuclear-safety-related instrument buses)

9. Nuclear-safety-related instrument bus panels

10. Containment penetration assemblies

11. Emergency lighting system (See FSAR Section 9.5.3.2.2) 36

12. Diesel generator and accessories

13. Diesel generator control panels

14. Relay boards and racks (nuclear-safety-related)

15. Instrument racks (nuclear-safety-related)

16. Hot shutdown panel (used in event of Control Room evacuation)

17. Main Control boards

18. Verticle panels (including solid state sequencer and isolation equipment)

19. Wire and cable raceway system (nuclear-safety-related)

20. Electrical supports (nuclear-safety-related)

21. Containment particulate, iodine, and gas radiation monitors

22. Motors (nuclear-safety-related)

23. 120/208 VAC power distribution and lighting panels

3 24. Miscellaneous 3-phase 480-208/120v and 1-phase 480-120v transformers

27 25. Control Room ventilation radiation monitors

3 26. Post-Accident Monitoring System including sensors

27. Electronic Transmitters-pressure and differential pressure (nuclear safety related).

3 Q032.3 28. Analog Control Cabinets (Train A and Train B)

3.10B.3 METHODS AND PROCEDURES OF ANALYSIS OR TESTING OF SUPPORTS OF INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Supports of instrumentation and electrical equipment such as battery racks, instrument racks, control consoles, cabinets, and panels are analyzed or tested, or both, by their suppliers in accordance with the methods and procedures described in Section 3.7B.2.1.3.

Such supports are generally required to have overall natural frequencies greater than 33 Hz. Where this requirement cannot be met, suppliers are required to qualify their products by performing full dynamic analysis or testing, or both, to demonstrate their structural integrity during and after the SSE, and to generate response spectra or derive maximum amplification factors at all equipment and instrument mounting locations. The equipment and instruments to be mounted on these supports are then analyzed or tested by their suppliers on the basis of the response spectra or maximum accelerations furnished by support manufacturers.

The supplier accounts for possible amplification through equipment supports by analysis or testing, as described in Section 3.7B.2. Verification is by documentation based on either actual tests or analytical methods.

Documentation pertaining to the seismic qualification of all seismic Category I electrical equipment, instrumentation, and supports is reviewed for compliance with the requirements set forth in Section 3.7B.2, NRC regulatory guides, and applicable codes and standards.

The following is a description of the analysis and testing procedures used for qualification of Seismic Category I cable trays: 36

Two types of cable trays, solid bottom and ladder type, are used. Trays are supported on structural steel frames anchored to the ceiling, walls, or both, and braced transversely and longitudinally.

Under normal conditions of loading and support spacing, trays may have natural frequencies for lateral and normal bending which are lower than 33 Hz; therefore, they are required to sustain amplified accelerations which correspond to the peak values of the applicable floor response spectra.

Maximum equivalent static loads for SSE and five percent equipment damping are determined from the peaks of floor response spectra curves and based on the full mass of the tray (weight of the tray plus cable). Suppliers are required to determine by static testing on representative trays of each type being used to determine the strength of the trays in each of the three principal directions of load application, to compare the equivalent static loads in each direction with the respective strengths of the trays, and to combine the results using an interaction formula to account for the simultaneous application of dead load and earthquake excitation in the three principal directions.

36 Supports for the Seismic Category I electrical cable tray and conduit system are designed to be rigid (eliminating additional amplification of seismic accelerations) and to resist the gravity and seismic loads imposed on them by the cable trays. Supports are made of structural steel members, braced in two orthogonal directions (transverse and parallel to the direction of tray runs).

3.10B.4 OPERATING LICENSE REVIEW

The available seismic qualification documentation for seismic Category I instrumentation and electrical equipment is retained in the plant
33 records.

3.11N ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

The mechanical and electrical portions of the Engineered Safety Features and the Reactor Protection Systems are designed to ensure acceptable performance in all environments anticipated under normal, test, and design basis accident conditions. This section presents information on the design basis and qualification verifications for mechanical and electrical equipment in the Engineered Safety Features and the Reactor Protection System that are within the scope of the Westinghouse Nuclear Steam Supply System (NSSS). Section 3.7N presents the seismic design requirements and Section 3.10N presents the seismic qualification of electrical equipment.

3.11N.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

A list of safety-related equipment located inside and outside Containment, required to function during and subsequent to an accident, is given in Appendix 3A, Table 5-2. Plant normal environments are listed 36
in Table 3.11B-2. Plant abnormal and accident environments for this equipment are provided in Appendix 3A, Table 5-2. Environmental qualification conditions of this equipment in the Westinghouse scope of supply will be based upon the approach delineated in References [1] and [3].

3.11N.2 QUALIFICATION TESTS AND ANALYSES

For Westinghouse NSSS Class 1E equipment, Westinghouse will meet the Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," including IEEE Standard 323a-1975, the Nuclear Power Engineering Committee (NPEC) Position Statement of July 24, 1975, by an appropriate combination of any or all of the following: type testing, operating experience, qualification by analysis, and ongoing qualification. 33
Topical report, WCAP-8587, "Environmental

33 Qualification of Westinghouse Class 1E Equipment" (October 1975) has
been revised and resubmitted to the staff as WCAP-8587, Revision 1,
"Methodology For Qualifying Westinghouse PWR-SD Supplied NSSS Safety
Related Electrical Equipment" (September, 1977). This revision to
WCAP-8587 provided additional information requested by the staff
concerning the methodology employed by Westinghouse to qualify safety
2 related electrical equipment. In response to NRC staff requests for
Q032.2 further information on the details of the qualification program,
Westinghouse submitted Supplement 1 to WCAP-8587, "Equipment
Qualification Data Packages". This supplement describes the
performance specifications and requirements and the proposed test plan
for each piece of safety related electrical equipment. This material
is applicable to, and should be reviewed on the Comanche Peak docket.
Table 3.11N-3 identifies the Westinghouse supplied safety-related
equipment and the corresponding Equipment Qualification Data Package
33 contained in Supplement 1.

The overall Class 1E Westinghouse equipment qualification program
includes generic environmental conditions, e.g., temperature, pressure,
relative humidity, chemistry, radiation, which are established for the
4 various pieces of Westinghouse supplied Class 1E equipment. The
conditions vary according to location of the equipment. The generic
environmental conditions for which the equipment is qualified are
reported in the specific Equipment Qualification Data Package. The
postulated environmental conditions used for equipment qualification
36 are provided in Appendix 3A, Table 5-2.

How the requirements of the General Design Criteria (GDC) 1, 4, 23, and
50 are met is addressed in Section 3.1. Specific information
2 concerning how GDC 1 and 4 are met is reported in the applicable
Equipment Qualification Data Packages (Reference 3). Specific
information concerning how GDC 23 is met can be found in Section
7.2.2.2. Specific information concerning how GDC 50 is met is provided
4 in Section 6.2. Specific information concerning how Appendix B of 10

CFR Part 50 is met is located in Chapter 17. Regulatory Guides 1.30, 1.40, 1.73 and 1.89 are addressed in Appendix 1A(N). 4

3.11N.3 QUALIFICATION TEST RESULTS

A summary of the qualification test results are provided in Appendix 3A, Table 5-2. 33

3.11N.4 LOSS OF VENTILATION

Refer to Section 3.11B.4

3.11N.5 ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT

Chemical environments in the primary containment as result of an accident condition are shown in Appendix 3A, Table 5-2 and discussed in Section 3.11B.5.1. The postulated integrated radiation environments for normal and accident conditions are shown in Appendix 3A, Table 5-2. 36 Radiation and chemical environments for which the NSSS scope equipment is qualified are provided in the appropriate Equipment Qualification Data Packages listed in Table 3.11N-3 [3] and in Appendix 3A, Table 5-2.

REFERENCES

1. Jordan, W. G., Lorentz, D. G. and Miller, R. B., "Methodology For Qualifying Westinghouse PWR-SD Supplied NSSS Safety Related Electrical Equipment", WCAP-8587, Revision 1, September, 1977. 2

2. Wilson, J. F., "Electric Hydrogen Recombiner for PWR Containments," WCAP-7709L, Supplements 1 through 7 (Proprietary) and WCAP-7820, Supplements 1 through 7 (Non-Proprietary), 1971 through 1977.

2 3. "Equipment Qualification Data Packages", Supplement 1 to WCAP-8587, Revision 1, April, 1978.

TABLE 3.11N-1

DELETED

SEE FSAR TABLE 3.11B-2 36

AND

FSAR APPENDIX 3A EQR TABLE 5-2

TABLE 3.11N-2

DELETED

SEE FSAR APPENDIX 3A EQR TABLE 5-2 36

3.11B ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

To ensure proper performance during normal and design basis accident conditions, instrumentation and electrical components of the Engineered Safety Features (ESF) and other safety-related systems are designed for the environmental conditions and bases described in Section 3.11B.1. Specific information is also included in Sections 3.11B.2 and 3.11B.5 4
to demonstrate that the safety-related components have the capability to function as required in the combined temperature, pressure, humidity, chemistry, and radiation dose of the postaccident environment. In addition, an evaluation is made in Section 3.11B.4 of the environmental effects that would follow the loss of plant ventilation system used for cooling Class 1E electrical equipment.

Section 3.7B presents the seismic design requirements, and Section 3.10B presents the seismic qualification of electrical equipment.

LOCAs and steam line breaks are accidents that will cause environmental changes within the Containment. Pressure and temperature 4
transients for these design basis events are presented in Section 6.2.

3.11B.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

Safety-related equipment and components are designed to function during and subsequent to the DBA and are located within the Containment as well as in other seismic Category I structures (Safeguards Building, Auxiliary Building, Fuel Building, Electrical and Control Building and 4
Service Water Intake Structure).

The ESF systems and components required to function during and after an accident are shown in Table 3.11B-1. Plant normal (operating) conditions (pressure, temperature, and relative humidity) are given in 36
Table 3.11B-2. Plant accident environmental conditions (pressure, temperature, chemistry, relative humidity, and integrated Q032.31
normal/accident radiation dose) are given in Appendix 3A Table 5-1.

33 Environmental qualification documentation for Class 1E equipment is
Q032.31 retained in the plant records.

The ESF and other safety-related equipment which must remain operable during and after the DBA are further discussed in appropriate chapters of the FSAR as follows:

1. ESF and other Safety-Related Equipment Located Inside Containment

   4 a. Mechanical equipment (which contains Class 1E electrical components) in Chapter 6

   b. Class 1E equipment in Chapter 8

   c. Instrumentation and controls in Chapter 7

2. ESF and other Safety-Related Equipment Located Outside Containment

   4 a. Mechanical equipment (which contains Class 1E electrical components) is described in Chapters 6, 9, and 10.

   b. Instrumentation and controls are described in Chapter 7.

   c. Class 1E equipment is described in Chapter 8.

3.11B.2 QUALIFICATION TESTS AND ANALYSES

Qualification tests and analyses are performed on all ESF equipment and components as necessary to ensure their availability during and after
4 a DBA. These tests consist of simulation of actual physical conditions on the equipment or a prototype on a generic basis, or analysis, or a combination of prototype tests and analysis as applicable. Qualification testing is performed under simulated

conditions of temperature, pressure, relative humidity, chemistry, and radiation dose in excess of those expected for post-DBA conditions. The testing period is sufficient to ensure the capability to function during and after a DBA. 4

In order to provide assurance that all ESF and other Safety Related
equipment has capability to meet environmental conditions as required
the appropriate quality aassurance programs are established and 4
implemented.

Class 1E instrumentation and electrical equipment is capable of
operating in the worst expected environmental conditions as specified
for each component and its location in Appendix 3A, Table 5-1. The 33
Class 1E electrical equipment is specified for manufacture in
accordance with the criteria listed in Section 8.3. All class 1E 8
equipment will be qualified per IEEE 323 [11] and other applicable Q032.31
standards per Section 8.3 and below.

IEEE-323-1974 requirements include the need to establish the qualified
life of Class IE equipment. This requirement has, in many cases, 9
represented a state-of-the-art challenge in assessing the longevity of
equipment under normal and accident environments. Many of the
equipment qualification test reports have identified relatively short 33
lives of certain components. In some cases, these estimated, seemingly
short, qualified lives may be the result of testing or accelerated
aging limitations, rather than due to intrinsic equipment limitations.
TUGCO, as the first operating license applicant with an IEEE-323-1974
commitment has been confronted with an enormous developmental program 9
in meeting these new requirements.

Consistent with TUGCO's commitment to meet the requirements of IEEE-323-1974 subject to state-of-the-art limitations, TUGCO will modify any of the currently listed qualified lives as additional information and better testing and analytical techniques are developed. Therefore, the qualified lives currently listed should be regarded as decision points with regard to an ongoing aging evaluation, rather than a fixed component replacement interval. Changes from those qualified lives currently indicated will be documented by revisions to the appropriate qualification documents and/or approving additional qualification documents.

Instrumentation and control equipment mounted in the Control Room area is not subject to damaging vibration in either normal or postaccident modes, since no rotating equipment or fluid system components which could induce vibration are in proximity to the Control Room areas.

For Class 1E instrumentation and control equipment, including sensors, mounted in seismic Category 1 areas outside of the Control Room, the following design practices ensure that non-seismic vibration does not degrade the equipment:

1. Pressure and differential pressure transmitters and switches are mounted "off line," on secure and rigid (fn>33 hertz) instrument racks and supports. The instruments are supported independently of their connection to the process.

2. In-line temperature switches (thermostats) are not used.

3. In-line flow meters (such as target meters, rotameters, turbine meters, or paddle switches) are not used.

4. For equipment such as RTDs, level switches and valve limit switches, which are subject to non-seismic vibration, the qualification reports account for expected vibration, including the OBE tests required by IEEE 344-1975.

In addition to these design features, assurance that damaging vibration effects do not occur in service are provided by the preoperational tests and inspections as described in Section 14.2 as well as by the periodic on-line testing performed in accordance with Chapter 16.

Electrical equipment and components located outside of the Containment
33 Q032.31 are qualified to meet the plant environmental conditions as listed in Appendix 3A, Table 5-1.

Qualification tests for all safety related electric valve operators outside as well as inside containment are designed in accordance with References [8, 11, 13 and 15] to demonstrate their capability to function during and after accident environmental conditions. In addition, safety related electric valve operators installed inside the containment are also qualified in accordance with R.G. 1.73 [6]. 6 Q040.33

The class 1E instrumentation and control equipment and designated Accident Monitoring equipment (see FSAR Section 7.5) have been qualified in accordance with IEEE 344-1975 and IEEE 323-1974 criteria. 36

Because the Containment has no Class 1E motors which are required to operate on a continuous basis during or following a DBA, NRC Regulatory Guide 1.40 [3] is not applicable to CPSES. 4 Q022.4

Safety related equipment located inside the Containment that must remain operable during a DBA are listed in Appendix 3A, Table 5-1. As indicated in the list referenced previously, the only motors in the Containment required to operate in the event of a DBA are those motor-operated isolation valves which operate at the onset of a DBA. 33 Q032.31

The Class 1E motors located outside the Containment are type-tested in accordance with the intent of IEEE 334-1974 [12]. 4 Q040.32

The initial preoperational test procedures incorporate the recommendations of NRC Regulatory Guide 1.41 [4] for verifying proper load group assignment.

Provisions are integrated into the design criteria for required test circuitry, undervoltage-sensing relays, and simulated accident test signals to permit testing as outlined in Regulatory Guide 1.41.

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Subsequent to the initial service date, the station standard maintenance and test procedures require that any major modification or repair to the onsite power systems or redundant load groups necessitates retesting of these systems to verify independence among redundant systems.

The design basis for the maintenance, testing, and replacement of plant
4 batteries is in compliance with the recommendations set forth in IEEE 450-1972 [17], as discussed in Section 8.3.2.

4 Inspections of the batteries are described in Section 8.3.

It is not proposed to test equipment of a passive nature, such as primary and secondary shielding.

6 Q040.32 Qualification tests for electric penetration assemblies are based on References [10 and 11].

Qualification tests for electric cables are based on References [11 and 16].

6 How the requirements of GDC 1, 24, 23 and 50 are met is addressed in Section 3.1. Specific information concerning how GDC 1 and 4 are met is reported in Appendix A of Reference [18]. Specific information concerning how GDC 23 is met can be found in Sections 7.2 and 7.3.

6 Q040.34 The quality assurance program to be applied to the design, fabrication and testing of all BOP safety related equipment conform to the requirements of 10CFR part 50 Appendix B. The QA program is described in Chapter 17.

Specific information concerning how Appendix B of 10 CFR Part 50 is met
4 is discussed in Chapter 17. Level of compliance with NRC Regulatory
6 Q040.39 Guides 1.30, 1.40, 1.63, 1.73, 1.89 and 1.131 is addressed in Appendix 1A(B).

3.11B.3 QUALIFICATION TEST RESULTS

All safety-related equipment and components will be demonstrated to perform their designed safety function under all normal, abnormal and accident conditions, by appropriate testing and analyses. The detailed qualification information and test results when completed, will be available for an NRC audit. A summary of the test results is provided in Appendix 3A, Table 5-1. 33 Q040.36

3.11B.4 LOSS OF VENTILATION

3.11B.4.1 Environmental Design Basis

The following design features preclude the possibility of a total ventilation system failure, which insures that room ambient temperatures do not exceed the maximum operational temperature limit for instrumentation and electrical equipment. A forced shutdown of the reactor due to high temperature is therefore not considered credible. 4

1. Control Room Ventilation System

   The Control Room air-conditioning system is designed to maintain its ambient temperature at 75°F and 50 percent relative humidity. The system is of seismic Category I design and includes the following design features:

   a. Sufficient redundancy in equipment and power supplies is provided to enable the system to sustain a single active component failure without loss of function.

   b. Redundant fans are connected to separate Class 1E buses as described . Section 8.3.

c. Instrumentation and controls which incorporate audible and visual alarms enable the operator to continuously monitor system performance and alert him in the event of system malfunction.

d. Failure modes for isolation valves and dampers (as described in Section 9.4.1) are set so that their failure does not render the system inoperable.

2. Ventilation Systems of Other Areas

The ventilation systems which serve areas containing safety-related equipment are provided with design features which are similar to those of the Control Room ventilation system.

3.11B.4.2 Limiting Environmental Conditions

1. UPPER LIMITS

The upper limit environmental conditions for the Control Room are as follows:

36	Temperature, ^{o}F	80
Q032.31	Pressure, in. wg	+0.1
	Relative humidity %	50
	Chemistry	N/A
	Radiation environment	Refer to Appendix 3A, Table 5-1

All other areas upper limit environmental conditions for safety related equipment are shown in Appendix 3A, Table 5-1.

2. LOWER LIMITS

The lower limit operating environment that Class 1E equipment is expected to operate in is specified in Table 9.4-2 and augmented by the following:

a. The lower limit operating temperature:
$40^{o}F$ for all areas except Control Room
$70^{o}F$ for the Control Room. 6 Q040.38

b. The lower limit operating relative humidity:
None

c. The lower limit operating pressure:
atmospheric for all areas except containment
-1.5 psig for containment

3.11B.4.3 Testing of Control and Electrical Equipment

Factory testing of protective system equipment is performed as stated in Section 3.11B.2. Periodic testing is provided in accordance with IEEE 338, Criteria for the Periodic Testing of Nuclear Power Generation Station Protection Systems [14], and IEEE 279, Criteria for Protection Systems [8].

The previously stated criteria for onsite testing are applied to all components of protection systems and are performed under ambient environmental conditions.

Overall testing activities are described in quality assurance programs identified in Chapter 17.

3.11B.5 ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT

3.11B.5.1 Chemical Environment

The majority of pressure boundary components are constructed principally of stainless steel. Containment structures are principally constructed of concrete and carbon steel (galvanized or coating).
4 These materials are compatible with, and do not suffer significant degradation in the containment spray environment. Other pressure boundary component (i.e. gaskets) materials are also selected for their compatibility with the spray environment.

Details concerning the chemicals used in and the chemistry of the containment spray system is presented in Sections 6.2.2 and 6.5.2.
33 Accident chemistry qualification of the safety-related electrical and
Q032.31 instrumentation components is shown in Appendix 3A, Table 5-1.

The atmosphere inside the containment building after a LOCA consists of
4 steam-air-water mixture with 2300 ppm boric acid bufered with sodium hydroxide (NaOH) to increase the final spray solution ph to between 8.6 to 10.

3.11B.5.2 Radiation Environment

The radiation environment for normal operation and for the DBA for each
33 safety-related component has been determined. The accident exposure
Q032.31 criteria are based on the TID-14844 [18] release model and, accordingly, are consistent with NRC Regulatory Guide 1.4 [1]. Additional conservative assumptions used in determining these upper-bound postaccident dose levels inside the Containment are as follows:

1. No consideration is given to the removal of radioactivity from the Containment atmosphere by sprays, filters, or fission product plateout. The only fission product removal is through decay.

2. No consideration is given to equipment, floor slabs, or secondary shielding inside the Containment Building.

36

The postulated integrated normal/accident dose is given in Appendix 3A, Table 5-1. 36

Q032.31

REFERENCES

1. NRC Regulatory Guide 1.4, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors, Revision 2, 6/74.

2. NRC Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electric Equipment, 8/11/72.

3. NRC Regulatory Guide 1.40, Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants, 3/16/73.

4. NRC Regulatory Guide 1.41, Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments, 3/16/73.

5. NRC Regulatory Guide 1.63, Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants, Revision 2, 7/78. 8

6. NRC Regulatory Guide 1.73, Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants, 1/74.

7. NRC Regulatory Guide 1.89, Qualification of Class 1E Equipment for Nuclear Power Plants, 11/74.

8. IEEE 279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations, sponsored by the IEEE Joint Committee on Nuclear Power Standards.

9. IEEE 308-1971, Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations.

4 10. IEEE 317-1976, Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations.

11. IEEE 323-1974, Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

12. IEEE 334-1974, Standard for Type Tests of Continuous Duty Class 1E Motors for Nuclear Power Generating Stations.

13. IEEE 336-1971, Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations.

14. IEEE 338-1971, Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems, prepared by Subcommittee 3 of the IEEE Joint Committee on Nuclear Power Standards, Revision No. 15, May 15, 1971.

15. IEEE 382-1972, IEEE Trial-Use Guide for Type Test of Class 1E Electric Valve Operators for Nuclear Power Generating Stations.

16. IEEE 383-1974, Standard for Type Test of Class 1E Electric Cables, Field Splices and Connections for Nuclear Power Generating Stations.

17. IEEE 450-1972, Recommended Practice for Maintenance, Testing, and Replacement of Large Stationary Type Power Plant and Substation Lead Storage Batteries.

18. TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites, by J.J. DiNunno, F.D. Anderson, R.E. Bauer, and R.L. Waterfield, Division of Licensing and Regulation, United States Nuclear Regulatory Commission.

TABLE 3.11B-2
NORMAL ENVIRONMENTAL CONDITIONS
FOR SAFETY-RELATED EQUIPMENT

BUILDING	AREA	NORMAL AMBIENT TEMP	NORMAL AMBIENT PRESS	NORMAL AMBIENT REL. HUMID
Containment	Outside the missile shield	120°F	Atmos	70%
Containment	Inside the missile shield	140°F	Atmos	70%
Containment	Excore detectors	135°F	Atmos	70%
Containment	N-16 detectors	200°F	Atmos	70%
Service Water Intake Structure	Various	122°F	Atmos	95%
Safeguards	Diesel generator compartments	122°F	Atmos	95%
Electrical and Control	Control Room and HVAC areas	75°F	0.1" wg	50%
Electrical and Control	Battery Room	87°F	Atmos	70%
Safeguards, Auxiliary, Electrical and Control, and Fuel Handling	All (except as noted above)	104°F	Atmos	70%
Turbine	Various	115°F	Atmos	95%

36

A status of the specific reviews of each type of equipment is provided 20 in Section 4.0 where the review forms are provided. The criteria used for these reviews is provided in Section 6.0. The reviews are conducted in accordance with approved procedure and are subject to audit and review by Quality Assurance.

The final step in the qualification of each type of Class 1E electrical equipment is the acceptance of that equipment with its individual qualification program for CPSES. This acceptance is documented on the certification forms as statused in Section 4.0. The criteria for this 20 acceptance is provided in Section 7.0.

The review and certification process documents the CPSES qualification program and demonstrates the level of involvement by the applicant.

Section 5 provides a summary of the environmental qualification data in an easily auditable format.

2.4 CPSES QUALIFICATION MAINTENANCE PROGRAM

The Qualification Maintenance Program at CPSES is designed to be integrated into the existing maintenance program while assuring that all qualified life contingencies and qualification requirements are fulfilled. The Qualification Maintenance Program consists of maintenance activities which in many cases, occur normally at a power 36 plant, but that are formalized in the case of class 1E electrical equipment.

All maintenance activities that are required by the equipment qualification reports shall be performed within the time intervals specified. In addition, this equipment shall be replaced before their qualified lives expire. As an alternative to replacement, it may be possible to extend qualified lives through additional testing or analysis. Any requalification of equipment will be controlled and documented in a process similar to the original qualification effort.

A required maintenance or replacement interval of 1 year shall be considered to be equivalent to every refueling outage for class 1E equipment which is either unavailable or inaccessible for maintenance during normal operations.

Although the methods used to qualify class 1E equipment are considered to be very conservative a program to identify qualification deficiencies shall be implemented at CPSES. A periodic surveillance of class 1E equipment shall be performed to identify signs of degradation. If degradation is discovered or if a piece of class 1E equipment should fail, the equipment shall be repaired and a failure analysis performed to determine the mode of failure. Particular attention will be placed to detect multiple failures attributable to a common cause. Any of these common-mode failures identified shall result in a revision to the qualification report, which shall include action to prevent this type of failure. If the failure analysis determines that the failure can be classified as a random failure, it shall be documented in an attempt to recognize failure trends. If a failure trend is recognized, indicating a common mode failure, a qualification report revision shall be 36
required.

Whenever possible, exact replacement spare parts and replacements shall be specified for class 1E equipment. An exact replacement shall be considered to be a component with the same part number as the original. If an exact replacement is not available, an engineering analysis shall be performed to assure that the proposed substitution does not violate the equipment's qualification. Parts with expired shelf lives will not be used on class 1E equipment at CPSES.

The beginning of life date for class 1E electrical equipment at CPSES shall be considered to be the date of the initial fuel load. For equipment with short qualified lives, an engineering analysis shall be performed to determine if a portion of a qualified life has been used before the initial fuel load date. If this is determined, the initial qualified life of this equipment shall be adjusted to account for this age related degradation or the equipment shall be refurbished.

TABLE 4-1

LIST OF TYPES OF CLASS 1E EQUIPMENT

(1 OF 5)

* Includes equipment located inside containment

NUMBER	DESCRIPTION	STATUS OF QUALIFICATION/REVIEW	
ES1D	Medium Pump Motors	Certified	24
ES1D.1	Service Water Pump Motors	Certified	
ES1D.2	Fuel Oil Transfer Pump Motors	Reviewed	36
ES2D	Lighting and Miscellaneous Transformers	Certified	33
ES5	7.2KV Metal Clad Switchgear	Reviewed	29
ES6 (1 of 2)	480V Transformers	Certified	24
ES6 (2 of 2)	480V Switchgear	Received	
ES7	Motor Control Centers	Received	33
ES8A	Batteries	Received	
ES8B (1 of 2)	Battery Chargers	Certified	24
ES8B (2 of 2)	Isolation Transformers	Received	33
ES9	Static Power Supply System	Reviewed	36
ES10	AC Distribution Panels	Reviewed	33
ES11 (1 of 2)	DC Distribution Panels	Reviewed	
ES11 (2 of 2)	Transfer Switch	In progress	36
ES12*	Electrical Penetration Assemblies	Certified	
ES12A*	Electrical Penetration Assemblies	--	33
ES13A	8KV Power Cable	Certified	24
ES13B.1*	600V Control Cable	Certified	
ES13B.1 (2 of 2)*	600V Cable and Switchgear Wire	Certified	33
ES13B.2 (1 of 2)*	600V Power and Lighting Cable	Certified	24
ES13B.2 (2 of 2)	Silicone Rubber Insulated Cable	Certified	33
ES13C*	Instrument Cable	Certified	24
ES13C.1	Instrumentation and Thermocouple Cable	Certified	33
ES13D (1 of 4)	Silicone Rubber Control Cable	Certified	31

TABLE 4-1
(2 of 5)

ES13D (2 of 4)	Tefzel Coax & Triax	Certified	
ES13D (3 of 4)	EPR Single & Multiconductor	Certified	31
ES13D (4 of 4)	XLPE Coax & Triax	Certified	
ES13D.1	Class 1E Connectors	Received	36
ES13E	600V Power and Control with Silicone	Certified	33
ES16A	Containment Hi Range Rad. Monitor	Received	31
ES18	Relay Panels, Boards and Racks	Received	33
ES22	Solid State Sequencers	Certified	24
ES22A	Solid State Sequencers	In progress	36
ES24	Solid State Isolation Equipment	Certified	33
ES28*	Conduit Seals	Certified	
ES29 (1 of 4)	Limit Switches	Certified	36
ES29 (2 of 4)	Limit Switches	Reviewed	
ES29 (3 of 4)	Limit Switches	Received	
ES29 (4 of 4)	Limit Switches	Received	
ES100 (1 of 4)	8KV Termination Assemblies and Splices	Certified	33
ES100 (2 of 4)*	600V Heat Shrink & Splice kits	Certified	
ES100 (3 of 4)	8KV Motor Splices	Received	
ES100 (4 of 4)*	Terminal Blocks	Certified	
SS15*	Airlock Electrical Penetration Assemblies	Certified	24
MS13	Medium Pump Motors	Certified	33
MS15A	RWM Pump Motors	Certified	31
MS15B	Sump Pump Motors	Certified	24
MS15C	Small and Medium Pump Motors - CW	Certified	33
MS20B (1 of 2)	Motor Operated Valves - Operator	Certified	
MS20B (2 of 2)	Motor Operated Valves - Limit Switches	See ES29	36
MS20B.1 (1 of 3)	Pneumatic Hydraulic Operator	Certified	25
MS20B.1 (2 of 3)*	Motor Operated Valves - Operator	Certified	33
MS20B.1 (3 of 3)*	Motor Operated Valves - Limit Switches	See ES29	36
MS34 (1 of 3)	Emergency Generator	Received	
MS34 (2 of 3)	Emergency Generator Control Panel	In progress	29
MS34 (3 of 3)	Emergency Diesel Control Panel	Received	36

TABLE 4-1
(3 of 5)

MS76 (1 of 3)	MSIV's - Bypass Operator	Received	33
MS76 (2 of 3)	MSIV's - Main Actuator	Received	24
MS76 (3 of 3)	MSIV's - Limit Switches	See ES29	
MS78 (1 of 2)	Main Steam Relief Valves - Limit Switches	See ES29	36
MS78 (2 of 2)	Fisher Transducer	Received	
MS80B (1 of 5)	Centrifugal Water Chillers, MCC'S	Certified	24
MS80B (2 of 5)	Chillers, Electrical Components	In progress	
MS80B (3 of 5)	Chillers, Electrical Devices	Deleted	
MS80B (4 of 5)	Chillers, Time-Delay Relays	Deleted	36
MS80B (5 of 5)	Chillers, Control Panel	Deleted	
MS81 (1 of 2)	HVAC Coolers, Motors	Certified	24
MS81 (2 of 2)	Manual Control Station's	Certified	33
MS82 (1 of 2)	HVAC Cleanup Trains - Control Devices	Received	36
MS82 (2 of 2)	Control Panel, Fire Protection	Received	33
MS83B	HVAC Fan Motors	Certified	24
MS84 (1 of 3)	MVAC Dampers and Valves - Operators	Certified	33
MS84 (2 of 3)	HVAC Dampers and Valves - Solenoids	Certified	
MS84 (3 of 3)	HVAC Dampers and Valves - Limit Switches	See ES29	36
MS86 (1 of 3)*	HVAC Isolation Valves - Operators	Certified	33
MS86 (2 of 3)*	HVAC Isolation Valves - Solenoids	Certified	
MS86 (3 of 3)*	HVAC Isolation Valves - Limit Switches	See ES29	36
MS87 (1 of 2)	Control Room Air Conditioning Motors	Certified	24
MS87 (2 of 2)	Control Room Air Conditioning Units	Received	33
MS87A (1 of 2)	UPS Air Conditioning Motors	Certified	36
MS87A (2 of 2)	UPS Air Conditioning Units	In progress	
MS92B	Vaneaxial Fan Motors	Certified	24
MS160A*	Hydrogen Monitor System	Received	33
MS600 (1 of 4)	Valve Actuators - Solenoids	Reviewed	31
MS600 (2 of 4)	Valve Actuators - Motor Operator	Reviewed	
MS600 (3 of 4)*	Valve Actuators - Limit Switches	See ES29	36
MS600 (4 of 4)	Valve Actuators - I/P Converter	Reviewed	
MS603 (1 of 2)*	Solenoid Operated Valves	Reviewed	33
MS603 (2 of 2)*	Solenoid Operated Valves - add on	Reviewed	31

TABLE 4-1
(4 of 5)

MS604 (1 of 2)*	Diaphragm Valves - Solenoids	Reviewed	31
MS604 (2 of 2)*	Diaphragm Valves - Limit Switches	See ES29	36
MS605	Control Boards and Panels	Reviewed	
MS611A	Electronic Transmitters	Reviewed	31
MS611B	Analog Control System	In progress	
MS613	Miniature Recorders	In progress	
MS616	Differential Pressure Switches	Received	33
MS620	Level Switches	Certified	
MS622*	RTD Assemblies	Certified	24
MS630A*	Containment Water Level (WR)	In progress	
MS631A	ICC Monitor	--	33
MS632	Solenoid Valves	Received	
CPF-585-S	Multiplexors and Isolators	Received	
ESE1 (1 of 2)*	Pressure Transmitters Group A-Barton	Certified	36
ESE1 (2 of 2)*	Pressure Transmitters Group A-Veritrak	In progress	29
ESE2	Pressure Transmitters Group B-Barton	Certified	
ESE3*	Differential Pressure Transmitters Group A	Certified	
ESE4 (1 of 2)	Differential Pressure Transmitters-Group B - Barton	Certified	
ESE4 (2 of 2)	Differential Pressure Transmitters Group B - Veritrak	Certified	
ESE6*	Well Mounted RTD'S	Certified	
ESE7*	Fast Response RTD'S	Certified	36
ESE10	Nuclear Instrumentation Cabinets	Certified	
ESE13	Process Protection System	Certified	
ESE14	PAM Indicators	Certified	
ESE15	PAM Recorders	Certified	
ESE16	Solid State Protection System	Certified	
ESE18	Static Inverters	Certified	
ESE20	Reactor Trip Switchgear	Certified	
ESE21*	Containment Pressure Sensors	Certified	
ESE22*	Four Section Power Range Detectors	Certified	

TABLE 4-1

ESE27*	N16 Detectors	Received	
ESE43	Incore Thermocouple Connectors	In progress	33
ESE44	Thermocouple Junction Box	In progress	
ESE47	Boron Dilution	In progress	
HE1*	Motor Operators Group A	In progress	24
HE2*	Solenoid Operators Group A	Certified	36
HE3*	Limit Switches Group A	Certified	31
HE4	Motor Operators Group B	Certified	36
HE-7	RCS Safety Valves - Position Switches	In progress	
HE-8*	Conax Connectors	In progress	33
HE-10*	Solenoid Valves - RCS Vent	In progress	
AE2	Large Pump Motors	Certified	
AE3	Canned Pump Motors	Certified	36
SP1*	Hydrogen Recombiners	Received	

Notes

1. IN PROGRESS - Vendor's qualification program is in progress but all qualification documentation not yet submitted.

   RECEIVED - All qualification documentation has been received but the review per Section 6 is not complete.

   REVIEWED - The qualification documentation has been reviewed but certification per Section 7 is not complete.

   CERTIFIED - The equipment qualification has been certified for use at CPSES in the specified manner.

BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-1

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
Valve motor operators Containment spray, Aux. Feedwater, Component Cooling, & Chilled Water category a	Containment bldg./ outside missile barrier	Limitorque Type SMB Class RH Insulated motors	Temp Pressure Rel. Humid Radiation Chemistry	268^oF(4) 48.1 psig 100% $1.9x10^8R$(g&b) spray 2300 ppm H_3BO_3 & NaOH to pH 8.6-10	300^oF 70 psig 100% $2.04x10^8R$(g) spray 3000 ppm Boron .064 molar $Na_2S_2O_3$ 10.5 pH	Acc Dur 30 days	Acc Dur 30 days	N/A	N/A	40 YR	Comb Test Anl Past- Hist		MS-208.1 2 of 3	
Valve motor operators Aux. FW., Com. Cool. W., Chill. W, & Cntmt Spray Category d	Safeguards Building	Limitorque Type SMB Class B Insulated motors	Temp Pressure Rel. Humid Radiation Chemistry	122^oF(5) Atmos. 95% $8.11x10^6R$(g) None	250^oF 25 psig 100% $2.04x10^8R$(g) None	Acc Dur 1 yr	Acc Dur >1 yr	N/A	N/A	40 YR	Comb Test Anl Past- Hist		MS-208.1 2 of 3	36
Valve Limit Switches Aux. FW, Com. Cool, Chill W., Cntmt spray category d	Containment & Safeguards/ various	NAMCO EA180	Temp Pressure Rel. Humid. Radiation Chemistry	268^oF 48.1 psig 100% $1.9x10^8R$ (g&b) spray 2300 ppm H_3BO_3 & NaOH to pH 8.6-10				N/A	N/A				MS-208.1 3 of 3 (ES29)	

CPSES/EQR

BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-1

(Sheet 2 of 37)

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
Valve motor	Safeguards	Limitorque	Temp	$122^{o}F$(5)	$385^{o}F$	Acc	Acc	N/A	N/A	40 yr	Comb		MS-20B
operators	bldg./valve	motor	Pressure	Atmos	70 psig	Dur	Dur				Test		1 of 2
Containment	isolation	actuator	Rel. Humid	95%	100%	(1 yr)					Anl		
spray	tank	SMB	Radiation	$2.24x10^{4}R(g)$	$2.04 X 10^{8}R(g)$						Past-		
system	compartments		Chemistry	None	spray N/A						Hist		
category d													
Valve limit	Safeguards	NAMCO	Temp	$122^{0}F$(5)		Acc		N/A	N/A				MS-20B
switches	bldg./	limit	Pressure	Atmos		Dur							2 of 2
Containment	valve	switch	Rel. Humid	95%		1 yr							(ES29)
spray	isolation	model	Radiation	$2.24X10^{4}R(g)$									
system	tank		Chemistry	None									
category d	compartments												
Main Steam	Safeguard	Namco limit	Temp	$122^{o}F$(5)				N/A	N/A				MS-78
Relief Valve	bldg./main	switches	Pressure	Atmos									(ES29)
operators	steam	model	Rel. Humid	95%									
Main steam	compartment	EA170	Radiation	$1.15x10^{3}R(g)$									
supply			Chemistry	None									
system													
category d													
Transducer		Fisher	Temp			Acc							MS-78
Main Steam			Pressure			Dur							2 of 2
Relief Valves			Rel. Humid			1 yr							
Main Steam			Radiation										
Supply			Chemistry										
System													
category d													

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BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-1

(Sheet 3 of 37)

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED[2]	OPERABILITY[1] REQ[3]	OPERABILITY DEM	ACCURACY REQ	ACCURACY DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
Main Steam	Safeguards	-Greer	Temp	122°F(5)		Acc		N/A	N/A				MS-76
Isol. Valve	bldg./main	Actuator	Pressure	Atmos		Dur							2 of 3
operators	steam	model	Rel. Humid	95%									
Main steam	compartment	OPC-03928-	Radiation	1.15×10^3R(g)									
supply system		CSVA	Chemistry	None									
category d													
Main Steam	Safeguards	Paul Monroe	Temperature	122°F(5)		Acc		N/A	N/A				MS-76
Isol. Valve	bldg./main	Valve	Pressure	Atmos.		Dur							1 of 3
Bypass	steam	Operator	Rel. Humid.	95%									
Operator	compartment	PD18860	Radiation	1.15×10^3R(g)									
Main Steam			Chemistry	None									
Supply													
System													
category d													
Main Steam	Safeguards	NAMCO	Temperature	122°F(5)		Acc		N/A	N/A				MS-76 36
Isol. Valve	bldg./main	Model	Pressure	Atmos.		Dur							3 of 3
Limit	steam	EA740	Rel. Humid.	95%									(ES29)
Switch	compartment		Radiation	1.15×10^3R(g)									
Main Steam			Chemistry	None									
Supply													
System													
category d													
Feedwater	Safeguards	Borg-Warner	Temp	200°F(5)	215°F	Acc	Acc	N/A	N/A	40YR	Test		MS-208.1
Isolation	bldg./	Nuclear	Pressure	Atmos	110 psig	Dur	Dur						1 of 3
Valves	Feedwater	Valve	Rel. Humid	100%	100%								
operator	Piping	Div.	Radiation	1.18×10^4R(g)	3×10^6R(g)								
FW system	Area	Model	Chemistry	None	None								

CPSES/EOR

BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 6-1

(Sheet 4 of 37)

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
category a		P/N 38991												
Valve motor Operators Component cooling water & reactor makeup water systems category d	Auxiliary & Safeguards bldg./outside shielded compartments & SWIS	Limitorque SMB-000 15H3BC, 2HOBC SMB-00/ 10H2BC, 10H3BC	Temp Pressure Rel. Humid Radiation Chemistry	$122^{o}F$(5) Atmos 95% $8.11x10^{6}R(g)$ None	$250^{o}F$ 25 psig 100% $2.04x10^{8}R(g)$ None	Acc Dur 1 yr	Acc Dur 1 yr	N/A	N/A	40 YR	Comb Test Anl Past-Hist		MS-600 2 of 4	
Valve Solenoids for Class 1E Air Operated Valves categories a&d	Containment, Safeguards, & Auxiliary bldgs. and SWIS	ASCO/ Catalog NP-1	Temp Pressure Rel. Humid. Radiation Chemistry	$268^{o}F$ 48.1 psig 100% $1.92x10^{8}R(b\&g)$ spray 2300 ppm H_3BO_3 & NaOH to pH 8.6-10	$346^{o}F$ 110 psig 100% $2x10^{8}R(g)$ spray 3000 ppm H_3BO_3 0.064 molar $Na_2S_3O_3$ & NaOH to 9.5-10.5 pH	Acc Dur 30 days	Acc Dur 30 days	N/A	N/A	8.5 YR (In cntmt) 19 YR (Outside cntmt)	Test		MS-600 1 of 4	36
Valve motor operator category d	Safeguards/ Valve iso. tank compartments	Limitorque Type SMB-00-10	Temp Pressure Rel. Humid Radiation Chemistry	$122^{o}F$ Atmos 95% $8.11x10^{6}R(g)$ None	$300^{o}F$ 70 psig 100% $2.04x10^{8}R(g)$ None	Acc Dur 1 yr.	Acc Dur 1 yr.	N/A	N/A	40 YR	Comb Test Anl Past-Hist		MS-600 2 of 4	

BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-1

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
Valve Limit Switches category a	Containment	NAMCO/ EA180	Temp Pressure Rel. Humid. Radiation Chemistry	$268^{\circ}F$ 48.1 psig 100% $1.92x10^{8}R(b\&g)$ spray 2300 ppm H_3BO_3 & NaOH to pH 8.6-10				N/A	N/A				MS-600 3 of 4 (ES29)
Valve Limit Switches category d	Auxiliary, Safeguards, and SWIS	NAMCO/ EA170	Temperature Pressure Rel. Humid. Radiation Chemistry	$122^{\circ}F(5)$ Atmos 95% $8.11x10^{6}R(g)$ None		Acc Dur		N/A	N/A				MS-600 3 of 4
Transducer & Positioner Class 1E Air operated valves category d	Safeguards bldg and outside/ normal access areas outside shielded compartments	Fisher Type 546 Transducer & Type 3582 Positioner	Temperature Pressure Rel. Humid. Radiation Chemistry	$122^{\circ}F(5)$ Atmos 100% $8.11x10^{6}R(g)$ None	$122^{\circ}F$ Atmos. 75% $1x10^{7}R(g)$ None	Acc Dur 1 yr	Acc Dur 3 yr	0.1"	0.1"	6 YR	Comb Test Anl		MS-600 4 of 4
Spent Fuel Pool Cooling Water Pump Motor	Fuel bldg./ inside shielded compartment	Siemens-Allis Type F-VPI Drip proof	Temp Pressure Rel. Humid Radiation	$122^{\circ}F(5)$ Atmos 95% $9x10^{3}R(g)$	$122^{\circ}F$ Atmos 100% $2x10^{8}R(g)$	Acc Dur (1 yr)	Acc Dur 1 yr	N/A	N/A	27 YR	Comb Test Past- Hist		MS-13

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
category d	El. 810'-6"		Chemistry	None	None								
Safeguards Sump Pump Motor category d	Safeguards bldg./normal access areas outside of shielded compt.	Crane-Deming/ Reliance motors Type RH, Class H, Insulation System	Temp Pressure Rel. Humid Radiation Chemistry	122^oF(5) Atmos 95% $1.91x10^4R(g)$ None	165^oF Atmos 100% $2.04x10^8R(g)$ None	Post-Acc 1 yr	Con-tin-uous	N/A	N/A	40YR	Comb Test Anl		MS-15B
Reactor Makeup Water Pump Motor category d	Aux. bldg.	Seimens-Allis Open drip proof, Class B, 130^oC	Temp Pressure Rel. Humid Radiation Chemistry	122^oF(7) Atmos 95% $9x10^2R(g)$ None	140^oF Atmos 95% $2x10^8R(g)$ None	Post seis 1 yr	Post seis	N/A	N/A	16 YR	Comb Test Anl Past-Hist		MS-15A
Local control station reactor makeup category d	Auxiliary bldg.	Reliance Electric Corp./ Plant Specific	Temp Pressure Rel. Humid. Radiation Chemistry	122^oF(5) Atmos 95% 900R(g) None	122^oF Atmos 95% $4.4x10^4R(g)$ None	Post seis 1 yr	Post seis 4.4 yr	N/A	N/A	40 YR	Comb Test Anl		MS-605
Wall mounted control stations electrical control	Safeguards bldg./El. 810'-6"	Reliance Electric Co./ Plant Specific	Temp Pressure Rel. Humid Radiation Chemistry	122^oF(5) Atmos 95% $1.47x10^3R(g)$ None	122^oF Atmos 95% $4.4x10^4R(g)$ None	Acc Dur 1 yr	Acc Dur 1.3 yr	N/A	N/A	40 YR	Comb Test Anl		MS-605

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BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-1

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
category d														
RTD	Containment	Conax	Temp	$268^{o}F$(4)	$310^{o}F$	Post	Post	$\pm1^{o}F$	$\pm1^{o}F$	40YR	Test		MS-622	
Assemblies	bldg./	Corp.	Pressure	48.1 psig	53 psig	Acc	Acc	or	or					
Accident	various	Dwg. Nos.	Rel. Humid	100%	100%	30	30	0.5%	0.5%					
Monitoring		2323-9028	Radiation	$2.01x10^{8}$R(b&g)	$1x10^{9}$R(g)	days	days							
system		-08 &	Chemistry	spray	spray									
category b		2323-9033		2300 ppm	2% H_3BO_3									
		-01		H_3BO_3	2% Na_2 S_2O_3 &									
				& NaOH to	0.3% NaOH									
				pH 8.6-10										
Indicators &	Electrical &	Westinghouse	Temp	$80^{o}F$(6)									MS-613	
Miniature	Control bldg./	ISD	Pressure	+0.1"wg										
Recorders	control		Rel. Humid	50%										36
Accident	room		Radiation	138R(g)										
Monitoring			Chemistry	None										
system														
category d														
Control	Electrical &	Reliance	Temp	$80^{o}F$(6)	$80^{o}F$	Acc	Acc	N/A	N/A	40 YR	Comb		MS-605	
Boards	Control bldg./	Electric Co.	Pressure	+0.1"wg	Atmos	Dur	Dur				Test			
Electrical	control	Main Control	Rel. Humid	50%	90%	1 yr	1 yr				Anl			
Control	room	Board &	Radiation	140R(g)	$4.4x10^{4}$R(g)									
System		Vertical	Chemistry	None	None									
category d		Panels												
Safety	Electrical	Borg-Warner/	Temp	$122^{o}F$(5)		Acc		N/A	N/A				MS-808	
Related	and Control	York Div.	Pressure	Atmos		Dur							2 of 5	
Chiller	bldg./	Chiller	Rel. Humid	95%										

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
electrical equipment category d	uncontrolled access area	units HTK3D1-EABS HTJ3B2-BCBS HTC2A3-ABBS	Radiation Chemistry	150R(g) None										
Safety related chiller MCC chilled water system category d	Electrical and Control bldg./ uncontrolled access area	Borg-Warner/ York Div. - Gould Inc. Motor Control Centers	Temp Pressure Rel. Humid Radiation Chemistry	122^0F(5) Atmos 95% 150R(g) None	122^0F Atmos 100% $10^4R(g)$ None	Acc Dur 1 yr	Acc Dur	N/A	N/A	40 YR	Comb Test Anl Past-Hist		MS-80B 1 of 5	
Safety Related Chilled Water pump motor Chilled water system category d	Electrical & Control bldg./ uncontrolled access area	Seimans-Allis motor TEFC, 155^0C, Class F Insulation	Temp Pressure Rel. Humid Radiation Chemistry	122^0F(5) Atmos 95% 150R(g) None	140^0F Atmos 95% $2x10^8R(g)$ None	Acc Dur 1 yr	Acc Dur 1 yr	N/A	N/A	16 YR	Comb Test Anl Past-Hist		MS-15C	36
Emergency Fan Coil unit motors Chilled water sys. and ventilation	Auxiliary, Fuel Handling, & Safeguards bldgs/pump rooms and electrical	AAF/ Reliance Elec. Company Class H Type RH Insulation	Temp Pressure Rel. Humid Radiation Chemistry	122^0F(5) Atmos 95% $2.75x10^6R(g)$ None	165^0F Atmos 100% $2.04x10^8R(g)$ None	Acc Dur 1 yr	Con-tin-uous	N/A	N/A	40 YR	Comb Test Anl		MS-81 1 of 2	

BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-1

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	OPERABILITY DEM	ACCURACY REQ	ACCURACY DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
category d	equip. areas												
Local Control Stations For Emergency Fan Coil Units, Chilled water system and ventilation category d	Auxiliary Fuel Handling, Safeguards bldg./ various areas	AAF Dwg. R107P-1124841	Temp	$122^{0}F(5)$	$137^{0}F$	Acc	Acc	N/A	N/A	40 YR	Comb		MS-81
			Pressure	Atmos	Atmos	Dur	Dur				Test		2 of 2
			Rel. Humid	95%	100%	1 yr	1 yr				Anl		
			Radiation	$1.91x10^{4}R(g)$	$1x10^{6}R(g)$								
			Chemistry	None	None								
ESF HVAC Atmospheric cleanup filt-ration unit electrical equipment. Primary plant exhaust sys. category d	Auxiliary bldg.	C.V.I. Corp.	Temp	$122^{0}F(5)$		Acc		N/A	N/A				MS-82
			Pressure	Atmos		Dur							1 of 2
			Rel. Humid	95%									
			Radiation	$9x10^{3}R(g)$									
			Chemistry	None									
Control Room Atmos pheric clean-up filt-ration unit Electrical equipment control room HVAC sys.	Electrical & Control bldg./HVAC equipment room	C.V.I. Corp.	Temp	$122^{0}F(5)$		Acc		N/A	N/A				MS-82
			Pressure	Atmos		Dur							1 of 2
			Rel. Humid	95%									
			Radiation	150R(g)									
			Chemistry	None									

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BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-1

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
category d													
H_2 Purge	Auxiliary	C.V.I. Corp.	Temp	$122^0F(5)$				N/A	N/A				MS-82
Exhaust Filt-	bldg/outside		Pressure	Atmos									1 of 2
ration unit	shielded		Rel. Humid	95%									
electrical	compartments		Radiation	$9x10^3R(g)$									
equip			Chemistry	None									
category d													
Control			Temp.										MS-82
Panel, Fire			Pressure										2 of 2
Protection			Rel. Humid										
E!ectrical			Radiation										
con^rol			Chemistry										
system													
category													
HVAC Fan	Outside	Buffalo	Temp	$122^0F(5)$	266^0F	Acc	Acc	N/A	N/A	40 YR	Test		MS-83B
motors	Containment	Forge/	Pressure	Atmos	Atmos	Dur	Dur						
HVAC System	various	Westinghouse	Rel. Humid	95%	100%	(1 yr)	1 yr						
category d	areas		Radiation	$9x10^3R(g)$	$2x10^8R(g)$								
			Chemistry	None	None								
Vaneaxial	Electrical	Reliance	Temp	$122^0F(5)$	165^0F	Acc	Con-	N/A	N/A	40 YR	Comb		MS-92B
Fan motors	Control &	Electric	Pressure	Atmos	Atmos	Dur	tin-				Test		
ventilation	Safeguards	Company	Rel. Humid	95%	100%	(1 yr)	ous				Anl		
systems	bldg/D.G.	Class H type	Radiation	900R(g)	$2.04x10^8R(g)$								
category d	compartments	RH insulation	Chemistry	None	None								
	& HVAC equip.												
	room												

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BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-1

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EQUIPMENT	LOCATION	MANUFACTURER	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES			OPERABILITY(1)		ACCURACY		QUAL	QUAL		PURCHASE
TYPE/CATEGORY	STRUCTURE/AREA	TYPE/MODEL	PARAMETER	POSTULATED	QUALIFIED(2)	REQ(3)	DEM	REQ	DEM	LIFE	METHOD	REF	SPEC
HVAC Damper	outside con-	PAPCO	Temp	$122^{o}F$(5)	$250^{o}F$	Acc	Acc	N/A	N/A	40 YR	Comb		MS-84
motor	tainment/	-Limitorque	Pressure	Atmos	25 psig	Dur	Dur				Test		1 of 3
operators	various	motor	Rel. Humid	95%	100%	1 yr	1 yr				Anl		
Ventilation		actuator	Radiation	$2.33x10^{5}R(g)$	$2.04x10^{8}R(g)$						Past-		
systems		model SMC-04	Chemistry	None	spray N/A						Hist		
category d													
HVAC Damper	outside	PAPCO	Temp	$122^{o}F$(5)	$346^{o}F$	Acc	Acc	N/A	N/A	40 YR	Test		MS-84
solenoid	containment/	-ASCO NP-1	Pressure	Atmos	110 psig	Dur	Dur						2 of 3
actuators	various	Solenoid	Rel. Humid	95%	100%	1 yr	1 yr						
Ventilation		valves	Radiation	$2.33x10^{5}R(g)$	$2x10^{8}R(g)$								
systems			Chemistry	None	spray N/A								
Category d													
HVAC Dampers	outside	PAPCO	Temp	$122^{o}F$(5)		Acc		N/A	N/A				MS-84
& valves-	containment/	Acme-	Pressure	+0.1" wg		Dur							3 of 3
limit	various	Cleveland	Rel. Humid	95%									(ES29)
switches		Development	Radiation	$2.33x10^{5}R(g)$									
Ventilation		Co.	Chemistry	None									
systems		-NAMCO Limit											
category d		switches											
		Model EA750											
Containment	Containment	POSI-SEAL	Temp	$268^{o}F$(4)	$346^{o}F$	Acc	Acc	N/A	N/A	19 YR	Test		MS-86
ventilation	bldg/outside	-ASCO	Pressure	48.1 psig	110 psig	Dur	Dur			(Inside			2 of 3
isolation	missile	Automatic	Rel. Humid	100%	100%	30	30			cntmt)			
valves	shield &	switch	Radiation	$1.9x10^{8}R(b\&g)$	$2x10^{8}R(g)$	days	days			40 YR			
electrical	Auxiliary	company	Chemistry	spray	spray					(outside			
equipment-	bldg.	solenoid		2300 ppm	3000 ppm					cntmt)			

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
Solenoid valve Containment ventilation system category a&d		valve NP-1		H_3BO_3 & NaOH to pH 8.6 to 10	H_3BO_3, 0.064 M $Na_2S_2O_3$ & NaOH to pH 9.5 to 10.5								
Containment ventil. iso- lation valves electrical equip. -motor actuators Containment ventil. sys. category a	Containment bldg/ outside missile shield	POSI-SEAL -Limitorque motor actuator Type SMB-0 Class RH Insulated motors	Temp Pressure Rel. Humid Radiation Chemistry	268^oF(4) 48.1 psig 100% $1.9x10^8R$(b&g) spray 2300 ppm H_3BO_3 & NaOH to pH 8.6 to 10	300^oF 70 psig 100% $2.04x10^8R$(g) spray NaOH 10.5 pH .28 molar H_3BO_3 3000 ppm .064 molar $Na_2S_2O_3$	Acc Dur 30 days	Acc Dur 30 days	N/A	N/A	40 YR	Comb Test Anl Past- Hist		MS-86 1 of 3
Containment Ventilation isolation valves motor actuators category d	Auxiliary Bldg./ various	POSI-SEAL Limitorque Type SMB-0 Class B Insulated motors	Temp Pressure Rel. Humid Radiation Chemistry	122^oF Atmos. 95% $9x10^3R$(g) None	250^oF 25 psig 100% $2.04x10^8R$(g) None	Acc Dur 1 yr	Acc Dur 1 yr	N/A	N/A	40 YR	Comb Test Anl Past- Hist		MS-86 1 of 3
Containment ventilation	Containment bldg./outside	POSI-SEAL -Namco limit	Temp Pressure	268^oF(4) 48.1 psig				N/A	N/A				MS-86 3 of 3

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EQUIPMENT	LOCATION	MANUFACTURER	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES			OPERABILITY(1)		ACCURACY		QUAL	QUAL		PURCHASE
TYPE/CATEGORY	STRUCTURE/AREA	TYPE/MODEL	PARAMETER	POSTULATED	QUALIFIED(2)	REQ(3)	DEM	REQ	DEM	LIFE	METHOD	REF	SPEC
isolation	primary,	switch model	Rel. Humid	100%									(ES29)
valves elec.	secondary &	EA740	Radiation	$1.9x10^8$R(b&g)									
equip.	compartment		Chemistry	spray									
-limit switch	shield			2300 ppm									
Containment	walls			H_3BO_3 &									
ventilation				NaOH tc									
systems				pH 8.6 to 10									
category a													
Control	Electrical &	C.V.I. Corp.	Temp	122°F(5)	266°F	Acc	Acc	N/A	N/A	40YR	Test		MS-87
Room air	Control bldg./	-Westinghouse	Pressure	Atmos	Atmos	Dur	Dur						1 of 2
conditioning	HVAC equip.		Rel. Humid	95%	100%	(1 yr)	1 yr						
units elec.	room		Radiation	150R(g)	$2x10^8$R(g)								
equipment			Chemistry	None	None								
-AC fan and													
compressor													
drive motors													
Ventilation													
system													
category d													
Process	Auxiliary	Valcor	Temp	122°F	346°F	Acc	Acc	N/A	N/A	40 YR	Comb		MS-603
Solenoid	bldg/	Engineering	Pressure	Atmos	113 psig	Dur	Dur				Test		2 of 2
Valves	shielded	1" Solenoid	Rel. Humid	100%	100%	1 yr	6.2 YR				Anl		
CVCS	compartment	Valves/	Radiation	$9x10^3$R(g)	$2x10^8$R(g)								
category d		V526-5295-	Chemistry	None	spray N/A								
		53											
Control room	Electrical &	Pennwalt	Temp	122°F(5)		Acc		N/A	N/A				MS-87
air cond-	Control bldg/	C.V.I. Corp.	Pressure	Atmos		Dur							2 of 2

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
itioning units elec. equip. Ventilation sys. category d	HVAC equip. room		Rel. Humid Radiation Chemistry	95% 150R(g) None									
UPS Room Air Conditioning Drive Motors Ventilation category d	Electrical & Control Bldg./ El. 778'-0"	Westinghouse Class H Insulation Integral HP AC Polyphase Induction Motor	Temperature Pressure Rel. Humid Radiation Chemistry	$122^{o}F$ Atmos 95% $1x10^{3}$(g) None	$122^{o}F$ Atmos 100% $2x10^{8}R(g)$ None	Acc Dur 1 yr	Con-tin-uous	N/A	N/A	40 YR	Comb Test Anl Past-Hist		MS-87A 1 of 2
UPS Air Conditioning Units Ventilation System category d	Electrical & Control bldg.		Temperature Pressure Rel. Humid Radiation Chemistry	$122^{o}F$ Atmos. 95% None				N/A	N/A				MS-87A 2 of 2
Process Solenoid Valves Containment Isolation Valves category a	Containment bldg./outside missle barrier Safeguards bldg./ shielded compartments	Valcor Engineering Corp. models V52600-5292-7 and V52600-5950-1 1" solenoid	Temp Pressure Rel. Humid Radiation Chemistry	$268^{o}F$(4) 48.1 psig 100% $1.92x10^{8}R$(b&g) spray 2300 ppm H_3BO_3 &	$346^{o}F$ 113 psig 100% $2x10^{8}R(g)$ spray 1720-2200 ppm H_3BO_3,	Acc Dur 30 days	Acc Dur 6.2 yrs.	N/A	N/A	40 YR	Comb Test Anl		MS-603 1 of 2

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
		valves		NaOH to pH 8.6 to 10	0.064 M $Na_2S_2O_3$ & NaOH to pH 10.5									
Differential Pressure Switches Electrical Controls category d	Auxiliary Safeguards, and Control buildings & SWIS/ normal access areas	ITT Barton series 580	Temp Pressure Rel. Humid Radiation Chemistry	122^{0}F(5) +0.1" wg 95% $2.24x10^{4}$R(g) None		Acc Dur		±5% set- point ±15% full scale					MS-616	
Power operated Diaphragm Valves -Limit Switch Containment Isolation category a	Containment & safeguards bldg./shielded compartments	ITT Grinnell -Namco Limit Switch Model EA-180	Temp Pressure Rel. Humid Radiation Chemistry	268^{0}F(4) 48.1 psig 100% $1.92x10^{8}$R(b&g) spray 2300 ppm H_3BO_3 & NaOH to pH 8.6 to 10		Acc Dur		N/A	N/A				MS-604 2 of 2 (ES29)	36
Air Operated Diaphragm Valves -Solenoid Valve Containment isolation	Containment & safeguards bldg./shielded compartments & outside missile shield	ITT Grinnell ASCO Solenoid Valve model NP831655E	Temp Pressure Rel. Humid Radiation Chemistry	268^{0}F(4) 48.1 psig 100% $1.92x10^{8}$R(b&g) spray 2300 ppm H_3BO_3	346^{0}F 110 psig 100% $2x10^{8}$R(g) spray 3000 ppm H_3BO_3, 0.064M	Acc Dur 30 days	Acc Dur 30 Day	N/A	N/A	8.5 YR (In cntmt) 30 YR (outside cntmt)	Test		MS-604 1 of 2	

BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-1

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
category a				& NaOH pH 8.6-10	$Na_2S_2O_3$ & NaOH to pH 9.5 to 10.5									
Level Switches Electrical Controls category d	Safeguards & Auxiliary bldg/outside shielded compartments	Magnetrol A-153F-TDM -X-MPG-Y- SIMD4DC- SIMD4DC & A-153F-MPG -X-Y-TDM- SIMD4DC- SIMD4DC	Temp Pressure Rel. Humid Radiation Chemistry	122°F(5) Atmos 100% 1.91×10^4R(g) None	350°F Atmos 100% 4.4×10^4R(g) None	Acc Dur	Acc Dur	N/A	N/A	15 YR	Comb Test Anl		MS-620	
Analog Control System Accident Monitoring System category d	Electrical & Control bldg/control room	Westinghouse, ISD	Temp Pressure Rel. Humid Radiation Chemistry	80°F(6) +0.1" wg 50% 138R(g) None		Acc Dur							MS-611B	36
Local Panel Spent fuel pool cooling cleanup sys. category d	Fuel bldg/ normal access area outside shielded compartments	Reliance Elec Corp./ Plant specific	Temp Pressure Rel. Humid Radiation Chemistry	122°F(5) Atmos 95% 7×10^3R(g) None	122°F Atmos. 95% 4.4×10^4R(g) None	Acc Dur 1 yr	Acc Dur 4.4 yr	N/A	N/A	40 YR	Comb Test Anl		MS-605	
Electronic Tranmitters	various areas outside	Rosemount 1153 DB	Temp Pressure	122°F(5) Atmos	318°F 73 psig	Acc Dur	Acc Dur	$\pm$.25%	$\pm$.[illegible]%	10 YR	Comb Test		MS-611A	

CPSES/EQR

BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-1

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
(pressure	containment	and	Rel. Humid	100%	100%	1 yr	2 yr				Anl			
and		1153GB	Radiation	$1.61x10^6$R(g)	$2.21x10^7$R(g)									
differential			Chemistry	None	None									
pressure)														
Analog Control														
Accident														
Monitoring														
System														
category d														
Local Panel	Safeguards	Reliance	Temp	$122^oF(7)$	122^oF	Post-	Post-	N/A	N/A	40 YR	Comb		MS-605	
Electrical	building/	Electric	Pressure	Atmos	Atmos.	seis	seis				Test			
Control Sys	switchgear	Corporation	Rel. Humid	95%	100%	1 yr	1.3 yr				Anl			
hot shutdown	area	Plant	Radiation	$1.01x10^3$R(g)	$4.4x10^4$R(g)									
panel		specific	Chemistry	None	None									36
category d														
Local Panel	Safeguards	Reliance	Temp	$122^oF(5)$	122^oF	Acc	Acc	N/A	N/A	40 YR	Comb		MS-605	
Primary	bldg/sampling	Electric	Pressure	Atmos	Atmos.	Dur	Dur				Test			
Sampling and	room	Corporation	Rel. Humid	95%	95%	1 yr	1.3 yr				Anl			
monitoring		Plant	Radiation	$1.89x10^3$R(g)	$4.4x10^4$R(g)									
system valve		specific	Chemistry	None	None									
control panel														
category d														
Shutdown	Safeguards	Reliance	Temp	$122^oF(5)$		Acc		N/A	N/A				MS-605	
Transfer	building/	Electric	Pressure	Atmos		Dur								
Panel	Switchgear	Corporation	Rel. Humid	70%		1 Yr								
Electrical	room		Radiation	$1.47x10^3$(g)										31
Control			Chemistry	None										

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
category d														31
7.2KV Switchgear & Accessories Electrical power system category d	Safeguards bldg. (Switchgear) Electrical & Control Bldg. (potential transformers)	Brown Boveri Electric	Temp Pressure Rel. Humid Radiation Chemistry	$122^{\circ}F$(5) Atmos 70% 1.47×10^{3}R(g) None	$125^{\circ}F$ Atmos 90% 10^{5}R(g) None	Acc Dur 1 yr	Acc Dur 1 yr	N/A	N/A	40 YR	Comb Test Anl Past- Hist		ES-5	
480V load center unit sub- station transformer Electrical power system category d	Safeguards building/ switchgear room	Westinghouse 2000 KVA Dry Type ASL	Temp Presure Rel. Humid Radiation Chemistry	$122^{\circ}F$(5) Atmos 70% 1.47×10^{3}R(g) None	$122^{\circ}F$ Atmos 95% 10^{4}R(g) None	Acc Dur (1 YR)	Acc Dur (1 YR)	N/A	N/A	40 YR	Comb Test Anl		ES-6 1 of 2	36
480 V load center unit substation switchgear Electrical power system category d	Safeguards building/ switchgear room	Westinghouse	Temp Pressure Rel. Humid Radiation Chemistry	$122^{\circ}F$(5) Atmos 70% 1.47×10^{3}R(g) None		Acc Dur		N/A	N/A				ES-6 2 of 2	

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TABLE 5-1

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
Auxiliary Relay Racks Electrical control system category d	Electrical & Control bldg./control room	York Electo- Panel	Temp Pressure Rel. Humid Radiation Chemistry	80^{0}F(6) +0.1" wg 50% 140R(g) None		Acc Dur		N/A	N/A				ES-18	
Solid State Isolation Equipment Electrical control sys. category d	Electrical & Control bldg/control room	Forney Engineering Company	Temp Pressure Rel. Humid Radiation Chemistry	80^{0}F(6) +0.1" wg 50% 138R(g) None	131^{0}F Atmos 90% 10^{4}R(g) None	Acc Dur 1 yr	Acc Dur	N/A	N/A	40 YR	Comb Test Anl		ES-24	36
AC & DC panel boards and DC switchboards Electrical power System category d	various locations outside containment	General Electric Co./AV line swbd & CCB & NAB panel boards	Temp Pressure Rel. Humid Radiation Chemistry	122^{0}F(5) Atmos 95% $1.37x10^{4}$R(g) None	122^{0}F Atmos 95% $5x10^{4}$R(g) None	Acc Dur 1 yr	Acc Dur 1 yr	N/A	N/A	40 YR	Comb Test Anl Past- Hist		ES-10/11 1 of 2	
Distribution Panel Transfer Switch	Various locations outside containment	ASCO	Temp Pressure Rel. Humid. Radiation	122^{0}F(5) Atmos 95% $1.37x10^{4}$R(g)		Acc Dur							ES-11 2 of 2	33

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
Electrical			Chemistry	None										
Power														33
System														
category d														
Batteries	Electrical &	Gould Inc.	Temp	$122^oF(5)$		Acc		N/A	N/A				ES-8A	
& Accessories	Control		Pressure	Atmos		Dur								
Electrical	bldg./		Rel. Humid	90%										
power	battery		Radiation	140R(g)										
system	room		Chemistry	None										
category d														36
Battery	Electrical	Power	Temp	122^oF	122^oF	Acc		N/A	N/A	40 YR	Comb		ES-8B	
Chargers &	& Control	Conversion	Pressure	Atmos	Atmos	Dur					Test		1 of 2	
Accessories	bldg/dis-	Products	Rel. Humid	90%	95%	1 yr					Anl			
Electrical	tribution	Model	Radiation	140R(g)	$10^4R(g)$									
power	room	3SD-130-300	Chemistry	None	None									
system														
category d														
Isolation	Electrical	Power	Temp	$122^oF(5)$		Acc		N/A	N/A				ES-8B	
transformer	& Control	Conversion	Pressure	Atmos		Dur							2 of 2	
Electrical	bldg./	Products	Rel. Humid	90%										
power	distribution		Radiation	421R(g)										33
system	room		Chemistry	None										
category d														
Static	Electrical &	Elgar Corp.	Temp	122^oF	122^oF	Acc	Acc	N/A	N/A	40 YR	Comb		ES-9	
uninteruptible	Control bldg/	Model UPS	Pressure	Atmos	Atmos	Dur	Dur				Test			36
power supply	distribution	UPS-103-1-	Rel. Humid	70%	85%	1 yr	27 yr				Anl			

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
Electrical & control system category d	room	132 10KVA Unit	Radiation Chemistry	140R(g) None	10^4R(g) None								
Cable Termination racks Electrical & Control system category d	Electrical & Control bldg/cable spreading area	Reliance Electric Corp./ Plant specific	Temp Pressure Rel. Humid Radiation Chemistry	115^0F Atmos 95% 140R(g) None	122^0F Atmos 95% $4.4x10^4$R(g) None	Acc Dur 1 yr	Acc Dur 1.6 yr	N/A	N/A	40 YR	Comb Test Anl		MS-605
Solid state sequencer Electrical control system category d	Electrical & control bldg/control room El. 830'-0"	Automation Industries Inc. Vitro Lab Division	Temp Pressure Rel. Humid Radiation Chemistry	80^0F(6) +0.1"wg 50% 138R(g) None	120^0F Atmos 90% 10^5R(g) None	Acc Dur (1 yr)	Acc Dur	N/A	N/A	40YR	Ongo		ES-22
Solid State sequencers Electrical control system category d	Electrical & control bldg/control room El. 830'-0"	Automation Industries Inc./Vitro Lab Division	Temp Pressure Rel. Humid Radiation Chemistry	80^0F(6) +0.1"wg 50% 138R(g) None		Acc Dur 1 yr		N/A	N/A				ES22A
Motor Control Centers	Various locations outside	General Electric Co. Model	Temp Pressure Rel. Humid	122^0F(5) Atmos 95%		Acc Dur		N/A	N/A				ES-7

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BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA
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EQUIPMENT	LOCATION	MANUFACTURER	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES			OPERABILITY(1)		ACCURACY		QUAL	QUAL		PURCHASE
TYPE/CATEGORY	STRUCTURE/AREA	TYPE/MODEL	PARAMETER	POSTULATED	QUALIFIED(2)	REQ(3)	DEM	REQ	DEM	LIFE	METHOD	REF	SPEC
Electric	containment	IC7700	Radiation	1.37×10^{4}R(g)									
power			Chemistry	None									
System													
category d													
Electrical	Containment	Bunker	Temp	268^{o}F(4)	340^{o}F	Acc	Acc	N/A	N/A	40 YR	Comb		ES-12
Penetration	bldg/	Ramo Corp/	Pressure	48.1 psig	105 psig	Dur	Dur				Test		
assemblies	penetration	Header	Rel. Humid	100%	100%	1 yr	4.5 yr				Anl		
Electrical	area	Plate	Radiation	1.92×10^{8}R(b&g)	2.112×10^{8}R(g)								
power &		Type	Chemistry	spray	3000 ppm								
control		Design		2300 ppm	H_3BO_3 &								
system				H_3BO_3 &	NaOH to								
category a				NaOH to	11 pH								
				8.6-10 pH									
Electrical	Containment		Temperature	268^{o}F(4)				N/A	N/A				ES-12A
Penetration	bldg./		Pressure	48.1 psig									
Assemblies	penetration		Rel. Humid.	100%									
Electrical	area		Radiation	1.92×10^{8}R(b&g)									
Power &			Chemistry	spray									
Control				2300 ppm									
System				H_3BO_3 &									
category a				NaOH to									
				ph 8.6-10									
Airlock	Containment	Chicago	Temp	268^{o}F	340^{o}F	Acc	Acc	N/A	N/A	40 YR	Comb		SS-15
Electrical	bldg.& Personnel	Bridge	Pressure	48.1 psig	62.5 psig	Dur	Dur				Test		
Penetration	airlock and	and Iron Co.	Rel. Humid	100%	100%	and	and				Anl		
Assemblies	equipment	-Conax Corp.	Radiation	1.9×10^{8}R(b&g)	2.2×10^{8}R(g)	Post-	Post-						
Low Voltage	hatch/outside	Model P/N	Chemistry	spray	spray	Acc	Acc						

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
Power and Instrumentation category b	missile barrier	7414-10000- 01 thru 06		2300 ppm H_3BO_3 & NaOH to 8.6-10 pH	0.15 gal/ min/sq ft H_3BO_3 & NaOH pH = 10.5									
8KV power cables Electrical power system category d	Various locations outside containment & fuel bldg.	Okonite Co. Okoguard 8KV insulated cable	Temp Pressure Rel. Humid Radiation Chemistry	$122^{o}F$(5) Atmos 95% $1.61x10^{6}R(g)$ None	$345^{o}F$ 114 psig 100% $2x10^{8}R(g)$ spray N/A	Acc Dur (1 yr)	Acc Dur >1 yr	N/A	N/A	40 YR	Test		ES-13A	
Instrumen- tation cable Electrical control system categories a & d	Various locations including inside containment	Rockbestos Company Firewall III	Temp Pressure Rel. Humid Radiation Chemistry	$268^{o}F$(4) 48.1 psig 100% $1.92x10^{8}R$(g&b) spray 2300 ppm H_3BO_3 & NaOH to 8.6-10 pH	$346^{o}F$ 113 psig 100% $2.01x10^{8}R(g)$ spray 0.28 M H_3BO_3, 3000 ppm & NaOH to pH 9-11	Acc Dur (30 days)	Acc Dur (130 days)	N/A	N/A	40YR	Test		ES-13C	36
Instrument- ation and thermo- couple extension	All buildings except containment	Anaconda 300V Flame guard. Flame Retardant	Temp Press Rel. Humid Radiation Chemistry	$122^{o}F$ Atmos 95% $8.11x10^{6}R(g)$ None	$312^{o}F$ 65 psig 100% $2x10^{8}R(g)$ spray N/A	Acc Dur (1 yr)	Acc Dur (40 yr)	N/A	N/A	40 YR	Test		ES-13C.1	

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	OPERABILITY(1) DEM	ACCURACY REQ	ACCURACY DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
wire category d		EPR											
600 V control cable Electrical control system categories a&d	Various locations including inside containment	Rockbestos Co. Firewall III	Temp Pressure Rel. Humid Radiation Chemistry	$268^{o}F$(4) 48.1 psig 100% $1.92x10^{8}R$(g&b) spray 2300 ppm H_3BO_3 & NaOH to 8.6-10 pH	$346^{o}F$ 113 psig 100% $2.01x10^{8}R$(g) spray 0.28M (3000 ppm) H_3BO_3 0.064 M $Na_2S_2O_3$ NaOH to pH 9-11	Acc Dur (30 days)	Acc Dur (130 days)	N/A	N/A	40 YR	Test		ES-13B.1 36
600v power & Lighting cable Electrical power & lighting system categories a&d	Various locations including inside containment	Okonite Co.	Temp Pressure Rel. Humid Radiation Chemistry	$268^{o}F$(4) 48.1 psig 100% $1.92x10^{8}R$(b&g) spray 2300 ppm H_3BO_3 & NaOH to 8.6-10pH	$345^{o}F$ 114 psig 100% $2.1x10^{8}R$(g) spray .28M (3000 ppm) H_3BO_3, 0.064 M $Na_2S_2O_3$ & NaOH to pH 10.5	Acc Dur (30 days)	Acc Dur (1[illegible] days)	N/A	N/A	40 YR	Test		ES-13B.2 1 of 2
600V	Various	Okonite/	Temp	$122^{o}F$(5)	$343^{o}F$	Acc	Acc	N/A	N/A	40 YR	Test		ES-13B.2

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BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
Power & Lighting Cable Electrical Power & Lighting category d	locations outside containment	Okotherm with Silicone Rubber Insulation	Pressure Rel. Humid Radiation Chemistry	Atmos. 100% 8.11×10^6R(g) None	110 psig 100% 5×10^7R(g) None	Dur 1 yr	Dur 1 yr						2 of 2
600V Switch- board wire Electircal control system categories a&d	Various locations including inside Containment	Rockbestos Type SISF 600 V Firewall III Switchboard wire 90°C	Temp Pressure Rel. Humid Radiation Chemistry	268°F(4) 48.1 psig 100% 1.92×10^8R(b&g) spray 2300 ppm H_3BO_3 & NaOH to 8.6-10 pH	346°F 113 psig 100% 2.04×10^8R(g) spray 0.28 molar H_3BO_3 (3000 ppm boron) 0.064 molar $Na_2S_2O_3$ & NaOH to 9-11 pH	Acc Dur 30 days	Acc Dur 1 yr	N/A	N/A	40 YR	Test		ES-13B.1 2 of 2
Pre- fabricated 600V Control cable Electrical control system	Various locations including inside containment	Boston Ins. Wire/ Silicone Rubber Insulated cable	Temp Pressure Rel. Humid Radiation Chemistry	268°F(4) 48.1 psig 100% 2×10^8R(g&b) spray 2300 ppm H_3BO_3	346°F 113 psig 100% 2.2×10^8R(g) spray 3000 ppm H_3BO_3	Acc Dur 30 days	Acc Dur 110 days	N/A	N/A	40 YR	Test		ES-13D 1 of 4

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
categories				& NaOH to	.064 M								
a&d				8.6-10 pH	$Na_2S_2O_3$								
					NaOH to								
					9.5-11 pH								
Pre-	Various	Boston Ins.	Temp	268^oF(4)	340^oF	Acc	Acc	N/A	N/A	40 YR	Test		ES-13D
fabricated	locations	Wire/	Pressure	48.1 psig	105 psig	Dur	Dur						2 of 4
Coaxial &	including	Coax &	Rel. Humid	100%	100%	30 days	159						
Triaxial	inside	Triax with	Radiation	$2x10^8R$(b&g)	$2x10^8R$(g)		days						
Cable	containment	TEFZEL	Chemistry	spray	spray								
Electrical		Insul.		2300 ppm	.28 molar								
Control				H_3BO_3	H_3BO_3								
categories				& NaOH to	NaOH to								
a&d				8.6-10 pH	10.5 pH								
Pre-	Various	Boston Ins.	Temp	268^oF(4)	340^oF	Acc	Acc	N/A	N/A	40 YR	Test		ES-13D
fabricated	locations	Wire/	Pressure	48.1 psig	105 psig	Dur	Dur						3 of 4
single &	including	Bostrad	Rel. Humid	100%	100%	30 days	100						
multi-	inside	with EPR	Radiation	$2x10^8R$(b&g)	$2x10^8R$(g)		days						
conductor	Containment	Insul.	Chemistry	spray	spray								
Control				2300 ppm	.28 molar								
Cable				H_3BO_3	H_3BO_3								
Electrical				& NaOH to	NaOH to								
Control				8.6-10 pH	10.5 pH								
categories													
a&d													
Pre-	Various	Boston Ins.	Temp	268^oF(4)	300^oF	Acc	Acc	N/A	N/A	40 YR	Test		ES-13D
fabricated	locations	Wire/	Pressure	48.1 psig	60 psig	Dur	Dur						4 of 4
Coaxial &	including	Bostrad 7	Rel. Humid	100%	100%	30 days	104						

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
Triaxial	inside	XLPE Ins.	Radiation	$2x10^8$R(b&g)	$2x10^8$R(g)		days							
cable	Containment	CSPE	Chemistry	spray	spray									
Electrical		jacket		2300 ppm	.28 molar									
Control				H_3BO_3	H_3BO_3									
categories				& NaOH to	NaOH to									
a & d				8.6-10 pH	10.5 pH									
Electrical			Temp			Acc		N/A	N/A				ES13.D.1	
Connectors			Pressure			Dur								
Electrical			Rel. Humid			1 yr								
Power and			Radiation											
Control			Chemistry											
Systems														
category														36
600V	Various	Anaconda	Temp	268^oF(4)	385^oF	Acc	Acc	N/A	N/A	40 YR	Test		ES-13E	
Power &	locations	Silicone	Pressure	48.1 psig	113 psig	Dur	Dur							
Control	including	rubber	Rel. Humid.	100%	100%	30	30							
Cable	inside	insulated	Radiation	$1.92x10^8$R(b&g)	$2x10^8$R(g)	days	days							
with	containment	cable	Chemistry	spray	spray									
Silicone				2300 ppm	4000 ppm									
Rubber				H_3BO_3	H_3BO_3 &									
Electrical				& NaOH to	NaOH to									
Power				8.6 to 10pH	ph 10.5									
category a & d														
Terminal	Inside	Weidmuller	Temp	268^oF(4)	475^oF	Acc	Acc	N/A	N/A	40 YR	Test		ES-100,	
Blocks	containment	Type SAK-	Pressure	48.1 psig	70 psig	Dur	Dur						App B	
Electrical		6N, 4 & 10	Rel. Humid	100%	100%	30 days	30 days						4 of 4	
power		Glass	Radiation	$1.92x10^8$R(b&g)	$2.07x10^8$R(g)									

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
system category a		filled phenolic material	Chemistry	spray 2300 ppm H_3BO_3 & NaOH to 8.6-10 pH	spray 3000 ppm H_3BO_3 & NaOH to 10.5 pH									
Terminal Blocks Electrical power system category d	Various locations outside containment	Weidmuller Type SAKR, SAK-C10, RSF2, SAK- 35, R-SF1, SAK-10, SAK- 2.5; Cellulose filled melamine	Temp Pressure Rel. Humid. Radiation Chemistry	$122^{o}F$(5) Atmos 95% $4x10^{7}R$(b&g) None	$266^{o}F$ Atmos 95% $7x10^{7}R$(g) None	Acc Dur 1 yr	Acc Dur 1 yr	N/A	N/A	40 YR	Test		ES-100 App. B 4 of 4	36
8KV nuclear terminations & bus connections Electrical power system category d	Various locations except containment building	Raychem/ NHVT high volt term. & NHVBC high volt Bus conn kit	Temp Pressure Rel. Humid Radiation Chemistry	$122^{o}F$(5) Atmos 95% $1.47x10^{3}R$(g) None	$360^{o}F$ 70 psig 100% $2x10^{8}R$(g) spray	Acc Dur 1 yr	Acc Dur 1 yr	N/A	N/A	40 YR	Test		ES-100, App B 1 of 4	
8KV motor splice joints Electrical	Safeguards bldg.		Temp Pressure Rel. Humid Radiation	$122^{o}F$(5) Atmos 95% $1.61x10^{6}R$(g)				N/A	N/A				ES-100 App. B. 3 of 4	33

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
power system category d			Chemistry	None										33
600V	Various	Raychem	Temp	268^oF(4)	314^oF	Acc	Acc	N/A	N/A	40 YR	Test		ES-100,	
in line	locations	WCSF	Pressure	48.1 psig	66 psig	Dur	Dur						App B	
cable splice	including	compound	Rel. Humid	100%	100%	30	30						2 of 4	
assemblies	inside	& tubing	Radiation	$1.92x10^8R$(g&b)	$2.9x10^8R$(g)	days	days							
Electrical power & control system categories a&d	containment	N-MCK connection kits	Chemistry	spray 2300 ppm H_3BO_3 & NaOH to 8.6-10 pH	spray 6200 ppm H_3BO_3 & NaOH to 10.5 pH w/ $Na_2S_2O_3$									
Conduit seals	Containment	Conax	Temp	268^oF(4)	342^oF	Acc	Acc	N/A	N/A	40YR	Comb		ES-28	36
Electrical	bldg.	Corp./ECSA	Pressure	48.1 psig	128 psig	Dur	Dur				Test			
power &			Rel. Humid	100%	100%	30	30				Anl			
control			Radiation	$1.92x10^8R$(g&b)	$2.2x10^8R$(g)	days	days							
systems category a			Chemistry	spray 2300 ppm H_3BO_3 & NaOH to 8.6-10 pH	spray 3000 ppm H_3BO_3 & NaOH to pH 10.57									
Diesel engine	Safeguards	Delaval	Temp	122^oF(5)		Acc		N/A	N/A				MS-34	
generator	bldg./D.G.		Pressure	atmos		Dur							1 of 3	
set	compartment		Rel. Humid	95%										
Diesel			Radiation	$9x10^2R$(g)										
generator			Chemistry	None										

AMENDMENT 36
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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
sets & auxiliary system category d													
Generator & exciter regulator Diesel generator sets & auxiliary system category d	Safeguards bldg/D.G. compartment	Portec	Temp Pressure Rel. Humid Radiation Chemistry	$122^{o}F(5)$ Atmos 95% $9x10^{2}R(g)$ None		Acc Dur		N/A	N/A				MS-34 1 of 3
Generator control panel Diesel generator sets & auxiliary system category d	Safeguards bldg/D.G. compartment	RTE-DELTA	Temp Pressure Rel. Humid Radiation Chemistry	$122^{o}F(5)$ Atmos 95% $9x10^{2}R(g)$ None		Acc Dur		N/A	N/A				MS-34 2 of 3
Engine Control panel Diesel Generator sets &	Safeguards bldg/D.G. compartment	Delaval	Temp Pressre Rel. Humid Radiation Chemistry	$122^{o}F(5)$ Atmos 95% $9x10^{2}R(g)$ None		Acc Dur		N/A	N/A				MS-34 3 of 3

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
auxiliary system category d													
Fuel Oil Transfer Pump Motors Diesel Generators category d	Safeguards bldg./ D.G. compartments El. 810'-6"	Reliance Electric/ Type RH Class H Insulation 3 H.P.	Temp Pressure Rel. Humid Radiation Chemistry	$122^{o}F$ Atmos. 95% $9x10^{2}R(g)$ None	$167^{o}F$ Atmos 100% $2.04x10^{8}R(g)$ None	Acc Dur 1 yr	Con- tin- uous	N/A	N/A	40 YR	Comb Test Anl		ES10.2
Component Cooling water pump motor Component cooling water system category d	Auxiliary bldg/pump compartment	Westinghouse Therma- lastic Epoxy Insul.	Temp Pressure Rel. Humid Radiation Chemistry	$122^{o}F$(6) Atmos 95% $9x10^{2}R(g)$ None	$122^{o}F$ Atmos 100% $2x10^{8}R(g)$ None	Acc Dur 1 yr	Acc Dur 1 yr	N/A	N/A	40 YR	Comb Test Anl		ES-1D 36
Auxiliary FW pump motor Aux. F.W. system category d	Safeguards bldg/pump compartment	Westinghouse Therma- lastic Epoxy Insul.	Temp pressure Rel. Humid Radiation Chemistry	$122^{o}F$(6) Atmos 95% $2.77x10^{3}R(g)$ None	$122^{o}F$ Atmos. 100% $2x10^{8}R(g)$ None	Acc Dur 1 yr	Acc Dur 1 yr	N/A	N/A	40 YR	Comb Test Anl		ES-1D
Containment spray pump motor	Safeguards bldg/ pump	Westinghouse Therma- lastic	Temp pressure Rel. Humid	$122^{o}F$(6) Atmos 95%	$122^{o}F$ Atmos 100%	Acc Dur 1 yr	Acc Dur 1 yr	N/A	N/A	40 YR	Comb Test Anl		ES-1D

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
Containment	compartment	Epoxy	Radiation	1.61×10^6R(g)	2×10^8R(g)									
spray sys.		Insul.	Chemistry	None	None									
category d														
Service water	SWIS/	Seimens-Allis	Temp	122^oF(5)	122^oF	Acc	Acc	N/A	N/A	22.5Yr	Comb		ES-1D.1	
pump motor	pump	NEMA Type 1	Pressure	Atmos	Atmos	Dur	Dur				Anl			
Service	compartment	Form	Rel. Humid	95%	100%	(1 yr)	1 yr				Past-			
water		wound motor	Radiation	200R(g)	2×10^8R(g)						Hist			
system		F-VPI epoxy	Chemistry	None	None						Ongo			
category d		Insul												
Lighting &	Various	Sorgel	Temp	122^oF(5)	122^oF	Acc	Acc	N/A	N/A	59 YR	Test		ES-2D	
miscellaneous	locations	Division	Pressure	Atmos	Atmos	Dur	Dur							36
transformers	except	of Square D	Rel. Humid	95%	100%	(1 Yr)	(1 yr)							
Electrical	containment	480-208/210	Radiation	9×10^3R(g)	1×10^6									
power &		VAC	Chemistry	None	None									
lighting		45KVA												
system		80^oC Temp												
category d		Rise												
Hydrogen	Electrical	EXO SENSORS	Temp	122^oF(5)									MS-160A	
Monitor	and Control	INC	Pressure	Atmos										
Processors	Bldg./	Hydrogen	Rel. Humid	95%										
category d		Analyser	Radiation	138R(g)										
		system	Chemistry	None										
Level	Containment/		Temp	268^oF(4)									MS-630A	
Transmitters			Pressure	48.1 psig										
Containment			Rel. Humid	100%										
Water Level			Radiation	1.92×10^8R(g&b)										

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
(WR)			Chemistry	spray										
Monitor				2300 ppm										
category a				H_3BO_3 &										
				NaOH to										
				8.6-10 pH										
														36
Level	Auxiliary		Temp	122^oF(5)									MS-630A	
electronic	Bldg./		Pressure	Atmos										
receivers			Rel. Humid	95%										
Cntmt sump			Radiation	138R(g)										
monitor			Chemistry	None										
category d														
Sub Cooling	Electrical		Temp										MS-631A	
Margin	and Control		Pressure											
Monitoring	Bldg./		Rel. Humid											
category d			Radiation											
			Chemistry											33
Multiplexors	Outside		Temp										CPF-	
and Isolators	Containment/		Pressure										585-S	
category d	various		Rel. Humid											
			Radiation											
			Chemistry											
Containment	Containment/		Temp										ES-16A	
High Range	El 905' 6"		Pressure											
Radiation			Rel. Humid											27
Monitor			Radiation											
category a			Chemistry											

BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-1

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
Solenoid	Various	ASCO	Temp	$268^{o}F$(4)									MS-632	
valves	locations	NP-1	Pressure	48.1 psig										
category a	including		Rel. Humid	100%										
	containment		Radiation	$1.92x10^{8}R$(b&g)										
			Chemistry	spray										
				2300 ppm										
				H_3BO_3 &										
				NaOH to										
				ph 8.6-10										
Position	Various	NAMCO	Temperature	$268^{o}F$(4)	$391^{o}F$	Acc	Acc	N/A	N/A	7.5 YR	Test		ES-29	
Switches	locations	EA180	Pressure	48.1 psig	119 psig	Dur	Dur						1 of 4	
category a	including		Rel. Humid.	100%	100%	30	30							36
	inside		Radiation	$2x10^{8}R$(b&g)	$2.3x10^{8}R$(g)	days	days							
	containment		Chemistry	spray	spray									
				2300 ppm	2300 ppm									
				H_3BO_3 &	H_3BO_3 &									
				NaOH to	NaOH to									
				pH 8.6-10	pH 10-11									
Position	Various	NAMCO	Temperature	$122^{o}F$(5)	$194^{o}F$	Acc	Acc	N/A	N/A	18 YR	Test		ES-29	
Switches	locations	EA170	Pressure	Atmos	Atmos	Dur	Dur						2 of 4	
category d	outside		Rel. Humid.	100%	100%	1 yr	1 yr							
	containment		Radiation	$8.11x10^{6}R$(g)	$2.04x10^{8}R$(g)									
			Chemistry	None	None									
Position	Various	NAMCO	Temperature	$122^{o}F$(5)		Acc		N/A	N/A				ES-29	
Switches	locations	EA750	Pressure	Atmos.		Dur							3 of 4	
category d			Rel. Humid.	95%		1 yr								

BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-1

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
			Radiation	8.11×10^6 R(g)										
			Chemistry	None										
Position	Various	NAMCO	Temperature	268^0F(4)		Acc		N/A	N/A				ES-29	
Switches	locations	EA740	Pressure	48.1 psig		Dur							4 of 4	36
category a	including		Rel. Humid.	100%		30								
	inside		Radiation	2×10^8 R(g)		days								
	containment		Chemistry	spray										
				2300 ppm										
				H_3BO_3 &										
				NaOH to										
				ph 8.6-10										

CPSES/EQR

BOP ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-1

(Sheet 36 of 37)

NOTES:

(1.) Operability is described for those accidents that directly affect the equipment being described. For example, equipment located inside containment would be directly affected by LOCA, MSLB, FLB, or a seismic event. Equipment in the control room, however, would only be directly affected by a seismic event.

(2.) Figures for the actual test conditions/qualification envelopes are provided or referenced in the qualification reports. 36

(3.) All equipment is required to remain undamaged and/or able to perform its intended safety function (without a specified accuracy) during and following a seismic event (SSE) unless specifically noted otherwise.

(4.) Temperature and pressure transients inside containment for LOCA and SLB are presented in FSAR Section 6.2. Containment sprays are operated for the duration of LOCA or SLB. The calculated vapor temperature resulting from postulated steam line breaks inside Containment Building exceeds 268^{o}F for a short time and reaches a peak value of 334^{o}F.

(5.) Environmental conditions may rise to the extreme parameters listed during 60 day period after an accident due to the indirect effect of a possible loss of non-safety-related HVAC systems.

(6.) Extreme environmental conditions vary little from normal conditions. Area is supported by safety-related HVAC systems as described in FSAR Section 9.4, or area has no HVAC system support (eg SWIS)

(7.) Equipment is not required to function after LOCA or SLB. The extreme parameters listed are due to the indirect effect of a possible loss of non-safety-related HVAC systems, or are normal conditions.

(8.) Postulated radiation valves are based on 40 year integrated dose, normal and accident, unless noted otherwise. 33

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ABBREVIATIONS:

b, g, n refer to beta gamma, or neutron radiation, respectively.

ACC DUR = Equipment operability requirement is for the duration of an accident (ie before, during, and after).

POST-ACC = Equipment is required to operate following an accident, either indefinitely or for the time period specified. POST-ACC includes POST-SEIS unless specifically noted otherwise.

POST-SEIS = Equipment is required to operate with the accuracy indicated following a seismic event.

NOT REQ FOR ACC = Equipment is not required to operate for an accident.

HI RAD SIG = Equipment is required to operate for an accident only on receipt of an initial high radiation signal.

TEST = Equipment qualification is by type test.

ANL = Equipment qualification is by analysis.

ONGO = Equipment qualification is by ongoing qualification program.

PAST-HIST = Equipment qualification is by past operating history.

COMB = Equipment qualification is by a combination of the methods indicated.

NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-2

(Sheet 1 of 19)

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
Wide-range	Containment	Barton	Temperature	268^0F(4)	420^0F	Post-	Post	±11%	±10%	6 Yr	Test	ESE-1		
reactor	Bldg./outside	763	Pressure	48.1 psig	57 psig	Acc	Acc							
coolant	missile shield	Veritrak	Rel. Humid	100%	100%	4	4							
pressure		76PH2	Radiation	$1.54x10^7R(g)$	$5x10^7R(g)$	months	months							
transmitter			Chemistry	$1.2x10^8R(b)$	$9x10^8R(b)$									
Accident				spray	2500 ppm	Post-	Post	±1.0%	±1.0%					
Monitoring				2300 ppm H_3BO_3		seis	seis							
System				NaOH to pH	10.7 ph									
category a				8.6-10										
Pressurizer	Containment	Barton	Temperature	268^0F(4)	420^0F	<5 min.	5	+11%	+10%	6 Yr	Test	ESE-1		
pressure	Bldg./outside	763	Pressure	48.1 psig	57 psig		min.	-16%	-15%					
transmitter	missile		Rel. Humid	100%	100%									36
Reactor	shield		Radiation	$6.7x10^5R(g)$	$5x10^7R(g)$	During	During	±11%	±7%					
Protection			Chemistry	$1.4x10^6R(b)$	$9x10^8R(b)$	Seis	Seis							
System				spray	2500 ppm									
category a				2300 ppm H_3BO_3										
				NaOH to pH	10.7									
				8.6-10										
Pressurizer	Containment	Barton	Temperature	268^0F(4)	420^0F	Post-	Post-	±15%	±15%	6 Yr	Test	ESE-3		
Level	Bldg./outside	764	Pressure	48.1 psig	57 psig	Acc	Acc							
transmitter	missile		Rel. Humid	100%	100%	4	4							
Accident	shield		Radiation	$1.54x10^7R(g)$	$5x10^7R(g)$	months	months							
Monitoring			Chemistry	$1.2x10^8R(b)$	$9x10^8R(b)$									
System				spray	2500 ppm									
category a				2300 ppm										
				H_3BO_3	10.7									
				NaOH to										

NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-2

(Sheet 2 of 19)

EQUIPMENT	LOCATION	MANUFACTURER	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES			OPERABILITY(1)		ACCURACY		QUAL	QUAL		PURCHASE
TYPE/CATEGORY	STRUCTURE/AREA	TYPE/MODEL	PARAMETER	POSTULATED	QUALIFIED(2)	REQ(3)	DEM	REQ	DEM	LIFE	METHOD	REF	SPEC
				pH 8.6-10									
Steam	Containment	Barton	Temperature	268^0F(4)	420^0F	Post-	Post-	±15%	±15%	6 Yr	Test	ESE-3	
generator	Bldg./outside	764	Pressure	48.1 psig	57 psig	Acc	Acc						
level	missile		Rel. Humid	100%	100%	4	4						
transmitter	shield		Radiation	$1.54x10^7R(g)$	$5x10^7R(g)$	Month	Month						
Reactor			Chemistry	$1.2x10^8R(b)$	$9x10^8R(b)$								
Protection				spray	2500 ppm	Trip	Trip	±10%	±10%				
System				2300 ppm H_3BO_3		<5 min	<5 min						
Accident				NaOH to pH	10.7								
Monitoring				8.6-10									
System													
category a													36
Reactor	Containment	Barton	Temperature	120^0F([illegible])	420^0F	During	During	±11%	±10%	6 Yr	Test	ESE-3	
coolant flow	Bldg./outside	764	Pressure	Atmos	57 psig	seis	seis						
transmitter	missile		Rel. Humid	70%	100%								
Reactor	shield		Radiation	$2x10^6R(g)$	$5x10^7R(g)$	Post-	Post-	±1%	±0.5%				
Protection			Chemistry	None	$9x10^8R(b)$	seis	seis						
System					2500 ppm H_3BO_3								
category c					NaOH to								
					10.7 pH								
Wide range	Containment	RdF	Temperature	268^0F(4)	420^0F	Post-	Post-	$\pm2.0^0F$	$\pm2.0^0F$	5.9 Yr	Test	ESE-6	
reactor	Bldg./Inside	Model N^0.	Pressure	48.1 psig	75 psig	Acc	Acc						
coolant	missile	21205	Rel. Humid	100%	100%	4	4						
temperature	shield		Radiation*	2.4x10[illegible] R(g&b)	$2.47x10^8R$(g&b)	Month	Month						
detector			Chemistry	spray	2750 ppm								
Accident			*Based on	2300 ppm									

NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-2

(Sheet 3 of 19)

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
Monitoring System category a			10 year normal & accident dose	H_3BO_3 NaOH to pH 8.6-10	10.7									
Excore Detectors Power Range Reactor Protection System category c	Containment Bldg./inside missile shield	W IGTD 24045	Temperature Pressure Rel. Humid Radiation* Chemistry *Based on 10 year normal dose	$200^{0}F$ 5 psig 95% $1.62x10^{9}R(g)$ $1.61x10^{11}$ $n/cm^{2}(n)$ spray 2300 ppm H_3BO_3 NaOH to 8.6-10	$200^{0}F$ 5 psig 95% $2.2x10^{9}R(g)$ $4.59x10^{17}$ $n/cm^{2}(n)$ spray 2750 ppm H_3BO_3 NaOH to 10.7 pH	5 min	5 min	No change in sensi- tivity	No change in sensi- tivity	5 Yr	Comb Test Anl	ESE-22		36
Excore detectors source range reactor protection system category c	Containment Bldg./inside missile shield		Temperature Pressure Rel. Humid Radiation* Chemistry *Based on 10 year normal dose	$150^{0}F(7)$ Atmos 70% $1.62x10^{9}R(g)$ $1.61x10^{11}$ $n/cm^{2}(n)$ None								ESE-47	Boron Dilution Fix	33
Fast response reactor	Containment Bldg./inside	RdF Model N^{0}	Temperature Pressure	$268^{0}F(4)$ 48.1 psig	$420^{0}F$ 75 psig	<5 min	5 min	$\pm 2.0^{0}F$	$\pm 2.0^{0}F$	11.8 Yr	Test	ESE-7		36

NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-2

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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED[2]	OPERABILITY[1] REQ[3]	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
coolant	missile	21232	Rel. Humid	100%	100%									
temperature	shield		Radiation*	$1.5X10^8$R(g)	$1.5x10^8$R(b&g)									
detector			Chemistry	spray	2750 ppm									
Reactor			*Based on	2300 ppm										
protection			20 year	H_3BO_3	10.7									
system			normal	NaOH to pH										
category a			dose	8.6-10										
Containment	Containment	Barton	Temperature	268^oF(4)	420^oF	Post-	Post-	N/A	N/A	6 Yr	Test	ESE-21		
pressure	Bldg./outside	351	Pressure	55 psig	75 psig	Acc	Acc							
sensor	missile		Rel. Humid	100%	100%	4	4							
Reactor	shield		Radiation	$1.54x10^7$R(g)	$5x10^7$R(g)	month	month							36
protection			Chemistry	$1.75x10^8$R(b)	$9x10^8$R(b)									
system &				spray										
Accident				2300 ppm										
Monitor				H_3BO_3	2750 ppm									
System				NaOH to										
category a				pH 8.6-10	10.7									
Nitrogen-16	Containment	W IGTD	Temperature	350^oF		Trip		$\pm$1%				ESE-27		
Detectors	Bldg./outside	WL24076	pressure	48.1		<5 min								
Reactor	missile		Rel. Humid	100%		During &								
protection	shield		Radiation	$2.3x10^7$R(g&b)		Post-								
System			Chemistry	spray		Seis								
category c				2300 ppm										
				H_3BO_3										
				NaOH to										
				ph 8.6-10										

NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA
TABLE 5-2
(Sheet 5 of 19)

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
N16 Detector Junction Box Reactor Protection System	Containment Bldg./outside missile shield	W WL20451	Temperature Pressure Rel. Humid. Radiation Chemistry	$268^{o}F$(4) 48.1 psig 100% $2.07x10^{6}R$(b&g) spray 2300 ppm H_3BO_3 NaOH to ph 8.6-10		Trip <5 min		N/A	N/A			ESE-27	
Electric hydrogen recombiners Engineered safeguard features category a	Containment Bldg./outside missile shield	W Sturtevant Model B	Temperature Pressure Rel. Humid Radiation Chemistry	$268^{o}F$(4) 48.1 psig 100% $1.65x10^{8}R$(g&b) spray 2300 ppm H_3BO_3 NaOH to pH 8.6-10		1 year Post LOCA		N/A	N/A			SP-1	
Hydrogen Recombiner Power Supply & Control Panel Engineered Safeguard Feature	Safeguards building/	W	Temperature Pressure Rel. Humid. Radiation Chemistry	$122^{o}F$(5) Atmos 95% 400R(g) None		1 yr Post LOCA		N/A	N/A			SP-1	

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NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA
TABLE 5-2
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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
category d													
Valve	Containment	Limitorque	Temperature	268^{o}F(4)		1 Yr		N/A	N/A			HE-1	
electric	Bldg./outside	later	Pressure	48.1 psig		Post							
motor	missile		Rel. Humid	100%		DBA							
operators	shield		Radiation	2x10^{8}R(g&b)		During							
Chemical			Chemistry	spray		&							
volume				2300 ppm		Post							
control,				H_3BO_3		Seis							
safety				NaOH to									
injection,				pH 8.6-10									
and													
Residual													36
heat removal													
systems													
category a													
Solenoid	Containment	Asco	Temperature	268^{o}F(4)	420^{o}F	1 Yr	1 Yr	N/A	N/A	8 Yr	Test	HE-2	
operated	Bldg./outside	NP-8316	Pressure	48.1 psig	57 psig	Post	Post						
valves	missile	NP-8320	Rel. Humid	100%	100%	DBA	DBA						
Chemical	shield	NP-210-036	Radiation	2x10^{8}R(g&b)	? 10^{8}R(g)								
Volume control,		NP-206-381	Chemistry	spray	2500 ppm								
safety				2300 ppm to									
injection,				H_3BO_3	H_3BO_3								
Residual				NaOH to	NaOH to								
heat removal				pH 8.6-10	10.7 pH								
and Cntmt													
isolation													
systems													

NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-2

(Sheet 7 of 19)

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED[2]	OPERABILITY[1] REQ[3]	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
category a														36
Valve	Containment	NAMCO	Temperature	$268^{o}F$(4)	$450^{o}F$	1 Yr	1 Yr	N/A	N/A	10 Yr	Test	HE-3		
operator	Bldg./outside	EA180	Pressure	48.1 psig	70 psig									
external	missile	&	Rel. Humid	100%	100%									
limit switches	shield	EA740	Radiation	$2x10^{8}R$(g&b)	$2.04x10^{8}R$(g)									
Chemical volume		Series	Chemistry	spray	spray									
control, safety				2300 ppm	2500 ppm									
injection, and				H_3BO_3	H_3BO_3									
Residual heat				NaOH to pH	NaOH									
removal				8.6-10	pH 10.5									
and Cntmt														
isolation														
systems														
category a														
Steamline	Safeguards	Barton	Temperature	$122^{o}F$(5)	$420^{o}F$	During	Con-	±11%	±10%	6 Yr	Test	ESE-1		
pressure	Bldg./	763	Pressure	Atmos	57 psig	seis	tin-							
transmitters	Steam		Rel. Humid	95%	100%		uous							
Reactor	Tunnel		Radiation	$1.15x10^{3}R$(g)	$5x10^{7}R$(g)	Post-seis		±1%	±1%					
protection &			Chemistry	None	spray									
Accident														
Monitoring														36
Systems														
category d														
Turbine	Turbine	Barton	Temperature	$115^{o}F$	$130^{o}F$	During	Con-	±1%	±1%	5 Yr	Test	ESE-2		
impulse	Bldg	753	Pressure	Atmos	Atmos	Seis	tin-							
pressure			Rel. Humid	95%	95%		uous							

NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA
TABLE 5-2
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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
transmitter Reactor protection system category d			Radiation Chemistry	200R(g) None	10^4R(g) None								
Containment pressure transmitter Reactor protection and Accident Monitoring Systems category d	Safeguards Bldg.	Barton 752	Temperature Pressure Rel. Humid Radiation Chemistry	122^oF(5) Atmos 95% $2.44x10^4$R(g) None	130^oF Atmos 95% 10^5R(g) None	Post-Acc	Post-Acc	$\pm$1%	$\pm$1%	5 Yr	Test	ESE-4	
Boric Acid tank level transmitter category d	Auxiliary Bldg./ normal access areas outside shielded compartments	Barton 752	Temperature Pressure Rel. Humid Radiation* Chemistry *Based on 5 year normal dose	122^oF(5) Atmos 95% $8.75x10^4$R(g) None	130^oF Atmos 95% 10^5R(g) None	Post-Seis	Post-Seis	$\pm$1%	$\pm$1%	5 Yr	Test	ESE-4	
Refueling water storage	Auxiliary Bldg./outside	Barton 752	Temperature Pressure	122^oF(5) Atmos	130^oF Atmos	Post-Acc	Post-Acc	$\pm$1%	$\pm$1%	5 Yr	Test	ESE-4	

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NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA
TABLE 5-2
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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
tank level	sheilded	Veritrak	Rel. Humid	95%	95%	4	4							
transmitter	compartment	76DF1	Radiation	200R(g)	10^5R(g)	months	months							
Accident			Chemistry	None	None									36
monitoring														
system														
category d														
NIS Source	Electrical		Temperature	$80^oF(6)$								ESE-47	Boron	
Range	& Control		Pressure	±0.1 in wg									Dilution	
Drawer	Bldg/control		Rel. Humid	50%									Fix	33
RPS	room		Radiation	138R(g)										
category d			Chemistry	None										
Nuclear	Electrical	W NICD	Temperature	$80^oF(6)$	120^oF	Con-	Con-	Normal	Normal	5 Yr	Test	ESE-10		
instrumentation	& Control	Single	Pressure	+0.1 in wg	Atmos	tin-	tin-	ac-	ac-					
system	Bldg/control	Bay	Rel. Humid	50%	92%	uous	uous	curacy	curacy					36
Reactor	room	Console	Radiation	138R(g)	10^4R(g)									
protection			Chemistry	None	None									
system														
category d														
Source		W-ID	Temperature									ESE-47	Boron	
range			Pressure										Dilution	
Pre-			Rel. Humid										Fix	
amplifier			Radiation											33
NIS			Chemistry											
category d														
Process	Electrical	W-ISD	Temperature	$80^oF(6)$	120^oF	Contin	Con-	$\pm0.5\%$	$\pm0.1\%$	5 Yr	Test	ESE-13		36

NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA
TABLE 5-2
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EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
protection	& Control	7300	Pressure	±0.1 in wg	Atmos	-uous	tin-							
sets	Bldg./control	Series	Rel. Humid	50%	95%		uous							
Reactor	room	2&3 Bay	Radiation	138R(g)	10^4R(g)									
protection			Chemistry	None	None									
system														
category d														
Accident	Electrical	W-RID/	Temperature	80^0F(6)	120^0F	Contin	Con-	±1.5%	±1.25%	5 Yr	Test	ESE-14		
Monitoring	& Control	VX-252	Pressure	±0.1 in wg	Atmos	-uous	tin-							
system	Bldg./control		Rel. Humid	50%	95%		uous							
indicators	room		Radiation	138R(g)	10^4R(g)									
Accident			Chemistry	None	None									
Monitoring														36
system														
category d														
Accident	Electrical	W CID	Temperature	80^0F(6)	120^0F	Contin	Con-	±4%	±2%	5 Yr	Test	ESE-15		
Monitoring	& Control	Optimac	Pressure	±0.1 in wg	Atmos	-uous	tin-	span	span					
system	Bldg./control	100	Rel. Humid	50%	95%		uous							
recorders	room		Radiation	138R(g)	10^4R(g)									
Accident			Chemistry	None	None									
Monitoring														
system														
category d														
Solid state	Electrical	W NICD	Temperature	80^0F(6)	120^0F	Contin	Con-	N/A	N/A	5 Yr	Test	ESE-16		
protection	& Control	Two	Pressure	±0.1 in wg	Atmos	-uous	tin-							
system logic	Bldg./control	Train	Rel. Humid	50%	88%		uous							
Reactor	room		Radiation	138R(g)	10^4R(g)									

NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA
TABLE 5-2
(Sheet 11 of 19)

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
protection system & Engineered safeguard features category d			Chemistry	None	None								
Static inverter (instrument power supply) Reactor protection system & Engineered safeguard features category d	Electrical & Control Bldg./ distribution room	W PED 7.5 kva	Temperature	104^oF(6)	120^oF	Contin -uous	Con- tin- uous	N/A	N/A	5 Yr	Test	ESE-18	
			Pressure	Atmos	Atmos								
			Rel. Humid	70%	95%								
			Radiation	140R(g)	10^4R(g)								
			Chemistry	None	None								
Reactor trip switchgear Reactor protection system & Engineered safeguard features category d	Safeguard Bldg./ Electrical Equipment area	W LVSD type DS-416 bkrs	Temperature	122^oF(5)	120^oF	30 days	30 days	N/A	N/A	5 Yr	Test	ESE-20	
			Pressure	Atmos	Atmos								
			Rel. Humid	95%	95%								
			Radiation	$1.01x10^3$R(g)	10^4R(g)								
			Chemistry	None	None								

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NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA
TABLE 5-2
(Sheet 12 of 19)

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
Valve	Safeguards	Limitorque	Temperature	$122^{o}F(5)$	$130^{o}F$	1 yr	1 yr	N/A	N/A	40 YR	Test	HE-4	
electric	Bldg./	SMB-000-2	Pressure	Atmos	Atmos								
motor	various	through	Rel. Humid	95%	95%								
operators		SMB-5-500	Radiation	$8.11x10^{6}R(g)$	$2x10^{7}R(g)$								
SIS,		and	Chemistry	None	None								
CVCS,		SB-000-2											
& RHRS		through											
category d		SB-3-175											
Valve	Auxiliary	Limitorque	Temperature	$122^{o}F(5)$	$130^{o}F$	1 yr	1 yr	N/A	N/A	40 YR	Test	HE-4	
electric	Bldg./	SMB-000-2	Pressure	Atmos	Atmos								
motor	various	through	Rel. Humid	95%	95%								
operators		SMB-5-500	Radiation	$3.15x10^{6}R(g)$	$2x10^{7}R(g)$								
CVCS &		and	Chemistry	None	None								
RHRS		SB-000-2											
category d		through											
		SB-3-175											
Solenoid	Safeguards	ASCO	Temperature	$122^{o}F(5)$	$420^{o}F$	1 yr	1 yr	N/A	N/A	8 YR	Test	HE-2	
operated	Bldg.	NP-8316	Pressure	Atmos	57 psig								
valves		NP-8320	Rel. Humid	95%	100%								
SIS,		NP-210-036	Radiation	$8.11x10^{6}R(g)$	$2x10^{8}R(g)$								
CVCS, RCS,		NP-206-381	Chemistry	None	spray								
RHRS &													
Containment													
isolation													
category d													
Solenoid	Auxiliary	ASCO	Temperature	$122^{o}F(5)$	$420^{o}F$	1 yr	1 yr	N/A	N/A	8 YR	Test	HE-2	

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NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA
TABLE 5-2
(Sheet 13 of 19)

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
operated	Bldg.	NP-8316	Pressure	Atmos	57 psig								
valves		NP-8320	Rel. Humid	95%	100%								
SIS		NP-210-036	Radiation	$2.25x10^7R(g)$	$2x10^8R(g)$								
category d		NP-206-381	Chemistry	None	spray								
Valve	S feguards	NAMCO	Temperature	$122^oF(5)$	450^oF	1 yr	1 yr	N/A	N/A	10 YR	Test	HE-3	
operator	Bl g.	EA180	Pressure	Atmos	70 psig								
external		&	Rel. Humid	95%	100%								
limit		EA740	Radiation	$8.11x10^6R(g)$	$2.04x10^8R(g)$								
switches		Series	Chemistry	None	spray								
RHRS,					2500 ppm								
CVCS, SIS,					H_3BO_3								
RCS, &					NaOH to								
Containment					ph 10.5								
isolation													
category d													
Valve operator	Auxiliary	NAMCO	Temperature	$122^oF(5)$	450^oF	1 yr	1 yr	N/A	N/A	10 YR	Test	HE-3	
external limit	Bldg	EA180	Pressure	Atmos	70 psig								
switches		&	Rel. Humid	95%	100%								
CVCS, SIS,		EA740	Radiation	$2.3x10^7R(g)$	$2.04x10^8R(g)$								
& RHRS		Series	Chemistry	None	spray								
category d					2500 ppm								
					H_3BO_3								
					NaOH to								
					pH 10.5								
Safety	Safeguards	W LMD	Temperature	122^oF	122^oF	1 yr	1 yr	N/A	N/A	3.8 Yr	Comb	AE-2	
injection	Bldg./	Buffalo	Pressure	Atmos	Atmos						Test		

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NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA
TABLE 5-2
(Sheet 14 of 19)

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
pump motors Safety injection system category d	safety injection pump compartments		Rel. Humid Radiation Chemistry	95% $2.25x10^5$R(g) None	95% $5x10^7$R(g) None						Anl		
Residual heat removal pump motors Residual heat removal system category d	Safeguards Bldg./RHR pump compartment	W LMD Buffalo	Temperature Pressure Rel. Humid Radiation Chemistry	122^oF Atmos 95% $1.8x10^6$R(g) None	122^oF Atmos 95% $5x10^7$R(g) None	1 yr	1 yr	N/A	N/A	3.8 Yr	Comb Test Anl	AE-2	
Centrifugal charging pump motors Chemical volume control system category d	Safeguards Bldg./ shielded compartments	W LMD Buffalo	Temperature Pressure Rel. Humid Radiaton Chemistry	122^oF Atmos 95% $2.75x10^6$R(g) None	122^oF Atmos 95% $5x10^7$R(g) None	1 yr	1 yr	N/A	N/A	3.8 Yr	Comb Test Anl	AE-2	
Boric Acid transfer pump motor Chemical volume control system category d	Auxiliary Bldg.	W Chem Pump	Temperature Pressure Rel. Humid Radiation Chemistry	122^oF Atmos 95% $3.46x10^3$R(g) None	122^oF Atmos 95% $1x10^4$R(g) None	6 hr	6 hr	N/A	N/A	40 Yr	Comb Test Anl	AE-3	

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NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-2

(Sheet 15 of 19)

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
Wide	Safeguards	Barton	Temperature	122°F(5)	130°F	Post	Post	±1%	±1%	5 YR	Test	ESE-4		
Range	bldg.	752	Pressure	Atmos.	Atmos.	Acc	Acc							
Containment			Rel. Humid	95%	95%	1 yr	1 yr							
Pressure			Radiation	$2.44x10^{4}$R(g)	10^{5}R(g)									36
Transmitter			Chemistry	None	None									
Accident														
Monitoring														
category a														
RCS	Containment/	NAMCO/	Temperature	268°F(4)	450°F	Acc	Acc	N/A	N/A	10 YR	Test	HE-3		
PORV	Outside	EA180&	Pressure	48.1 psig	70 psig	Dur	Dur							
Limit	missile	EA740	Rel. Humid	100%	100%	1 yr	1 yr							
Switches	shield	series	Radiation	$2x10^{8}$R(g)	$2.04x10^{8}$R(g)									
category a			Chemistry	spray	spray									
				2300 ppm	2500 ppm									
				H_3BO_3	H_3BO_3									
				NaOH to	NaOH to									
				pH 8.6-10	pH 10.5									
RCS Code	Containment		Temperature	268°F(4)								HE-7		33
Safety Vlvs			Pressure	48.1 psig										
Position			Rel. Humid	100%										
Switches			Radiation	spray										
category a			Chemistry	2300 ppm										
				H_3BO_3										
				NaOH to										
				pH 8.6-10										
Solenoid	Containment	Target	Temperature	268°F								HE-10		

NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA
TABLE 5-2
(Sheet 16 of 19)

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED(2)	OPERABILITY(1) REQ(3)	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC
valves RCS high Point Vent category a		Rock Solenoid	Pressure Rel. Humid Radiation Chemistry	48.1 psig 100% spray 2300 ppm H_3BO_3 NaOH to pH 8.6-10									
Electrical Connectors Solenoid v. & Limit Switches Control category a		Conax	Temperature Pressure Rel. Humid. Radiation Chemistry	268^{0}F(4) 43.1 psig 100% 2x10^{8}R(b&g) spray 2300 ppm H_3BO_3 NaOH to pH 8.6-10		Post LOCA 1 yr		N/A	N/A			HE-8	
Connectors & adaptors Incore Thermocouples Accident Monitoring category a		CKB Industries	Temperature Pressure Rel. Humid. Radiation Chemistry	268^{0}F(4) 48.1 psig 100% 2x10^{8}R(b&g) spray 2300 ppm H_3BO_3 NaOH to pH 8.6-10		Post LOCA 1 YR		N/A	N/A			ESE-43	
Reference Junction		W-IGTD	Temperature Pressure	268^{0}F(4) 48.1 psig		Post LOCA						ESE-44	

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NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA

TABLE 5-2

(Sheet 17 of 19)

EQUIPMENT TYPE/CATEGORY	LOCATION STRUCTURE/AREA	MANUFACTURER TYPE/MODEL	ABNORMAL/ACCIDENT ENVIRONMENTAL EXTREMES PARAMETER	POSTULATED	QUALIFIED[2]	OPERABILITY[1] REQ[3]	DEM	ACCURACY REQ	DEM	QUAL LIFE	QUAL METHOD	REF	PURCHASE SPEC	
Box			Rel. Humid.	100%		1 YR								
Incore			Radiation	$2x10^8$R(g)										
Thermocouples			Chemistry	spray										
Accident				2300 ppm										33
Monitoring				H_3BO_3										
category a				NaOH to										
				pH 8.6-10										

NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA
TABLE 5-2
(Sheet 18 of 19)

NOTES:

(1.) Operability is described for those accidents that directly affect the equipment being described. For example, equipment located inside containment would be directly affected by LOCA, MSLB, FLB, or a seismic event. Equipment in the control room, however, would only be directly affected by a seismic event.

(2.) Figures for the actual test conditions/qualification envelopes are provided in the referenced documents.

(3.) All equipment is required to remain undamaged and/or able to perform its intended safety function (without a specified accuracy) during and following a seismic event (SSE) unless specifically noted otherwise.

(4.) Temperature and pressure transients inside containment for LOCA and SLB are presented in FSAR Section 6.2. Containment sprays are operated for the duration of LOCA or SLB. The calculated vapor temperature resulting from postulated steam line breaks inside Containment Building exceeds 268°F for a short time and reaches a peak value of 334°F.

(5.) Environmental conditions may rise to the extreme parameters listed during 60 day period after an accident due to the indirect effect of a possible loss of non-safety-related HVAC systems.

(6.) Extreme environmental conditions vary little from normal conditions. Area is supported by safety-related HVAC systems as described in FSAR Section 9.4, or area has no HVAC system support (eg SWIS)

(7.) Equipment is not required to function after LOCA or SLB. The extreme parameters listed are due to the indirect effect of a possible loss of non-safety-related HVAC systems, or are normal conditions.

(8.) Postulated radiation values are based on 40 year integrated dose, normal plus accident, unless noted otherwise. 33

NSSS ENVIRONMENTAL QUALIFICATION SUMMARY DATA
TABLE 5-2
(Sheet 19 of 19)

ABBREVIATIONS:

b, g, n refer to beta, gamma, or neutron radiation, respectively.

Acc Dur = Equipment operability requirement is for the duration of an accident.

POST-ACC = Equipment is required to operate following an accident, either indefinitely or for the time period specified.
POST-ACC includes POST-SEIS unless specifically noted otherwise.

POST-SEIS = Equipment is required to operate with the accuracy indicated following a seismic event.

NOT REQ FOR ACC = Equipment is not required to operate for an accident.

HI RAD SIG = Equipment is required to operate for an accident only on receipt of an initial high radiation signal.

TEST = Equipment qualification is by type test

ANL = Equipment qualification is by analysis

COMB = Equipment qualification is by a combination of the methods indicated.

4.2.4.6 Onsite Inspection

Detailed written procedures are used by the station staff for the post shipment inspection of all new fuel and associated components such as control rods, plugs, and inserts. Fuel handling procedures specify the sequence in which handling and inspection takes place.

Loaded fuel containers, when received onsite, are externally inspected to ensure that labels and markings are intact and seals are unbroken. 36
After the containers are opened, the shock indicators attached to the suspended internals are inspected to determine if movement during transit exceeded design limitations.

Following removal of the fuel assembly from the container in accordance with detailed procedures, the fuel assembly polyethylene wrapper is examined for evidence of damage. The polyethylene wrapper is then removed and a visual inspection of the entire bundle is performed.

Control rod assemblies are shipped in fuel assemblies and are inspected during fuel receipt operations. The control rod assembly is withdrawn 36
from the fuel assembly to ensure free and unrestricted movement. The exposed section is then visibly inspected for mechanical integrity, replaced in the fuel assembly and stored with the fuel assembly.

REFERENCES

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(Non-Proprietary), July 1978 and Appendix A, "Hafnium."

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5. Miller, J. V. (Ed.), "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8720 (Proprietary) and WCAP-8785 (Non-Proprietary), October 1976.

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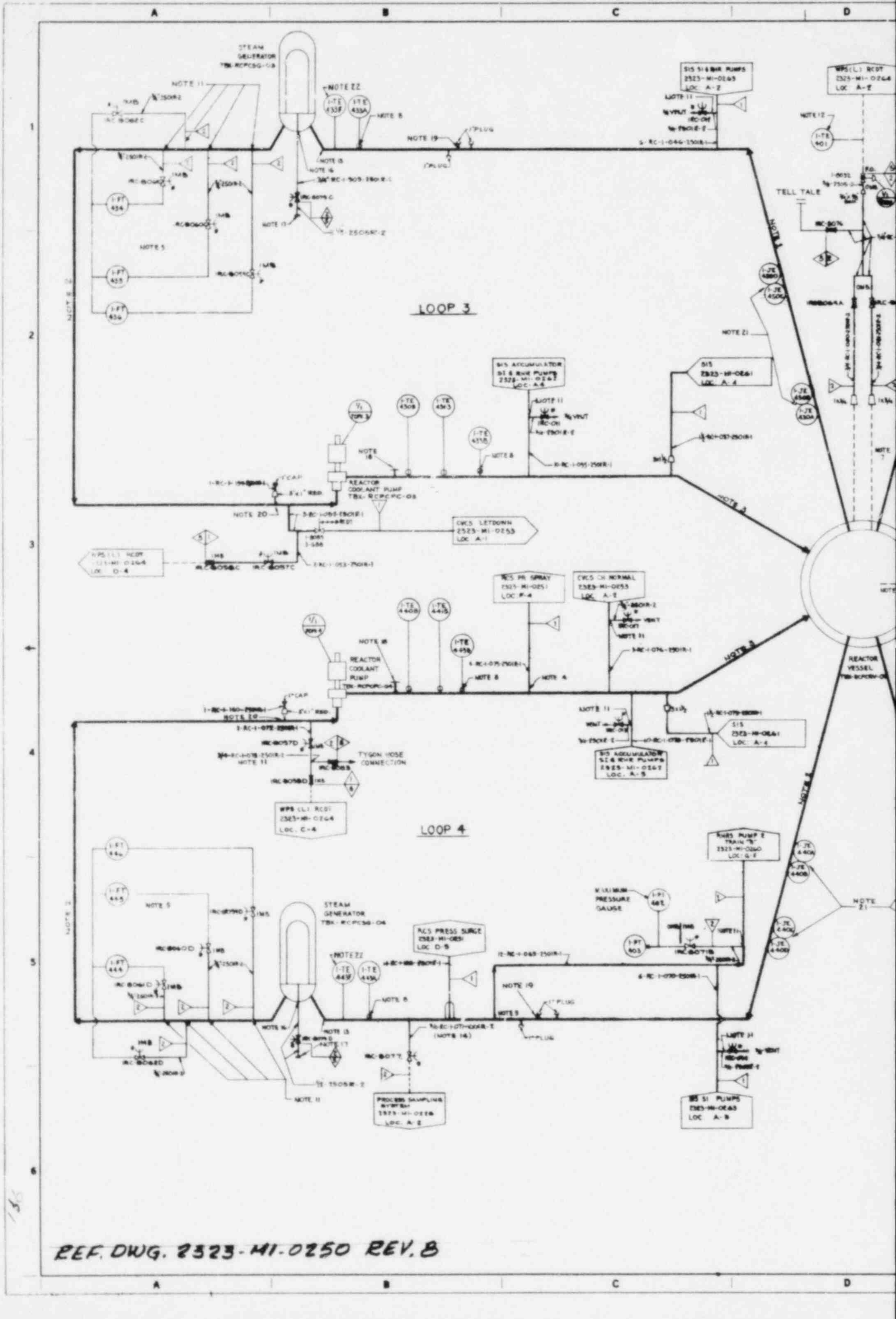
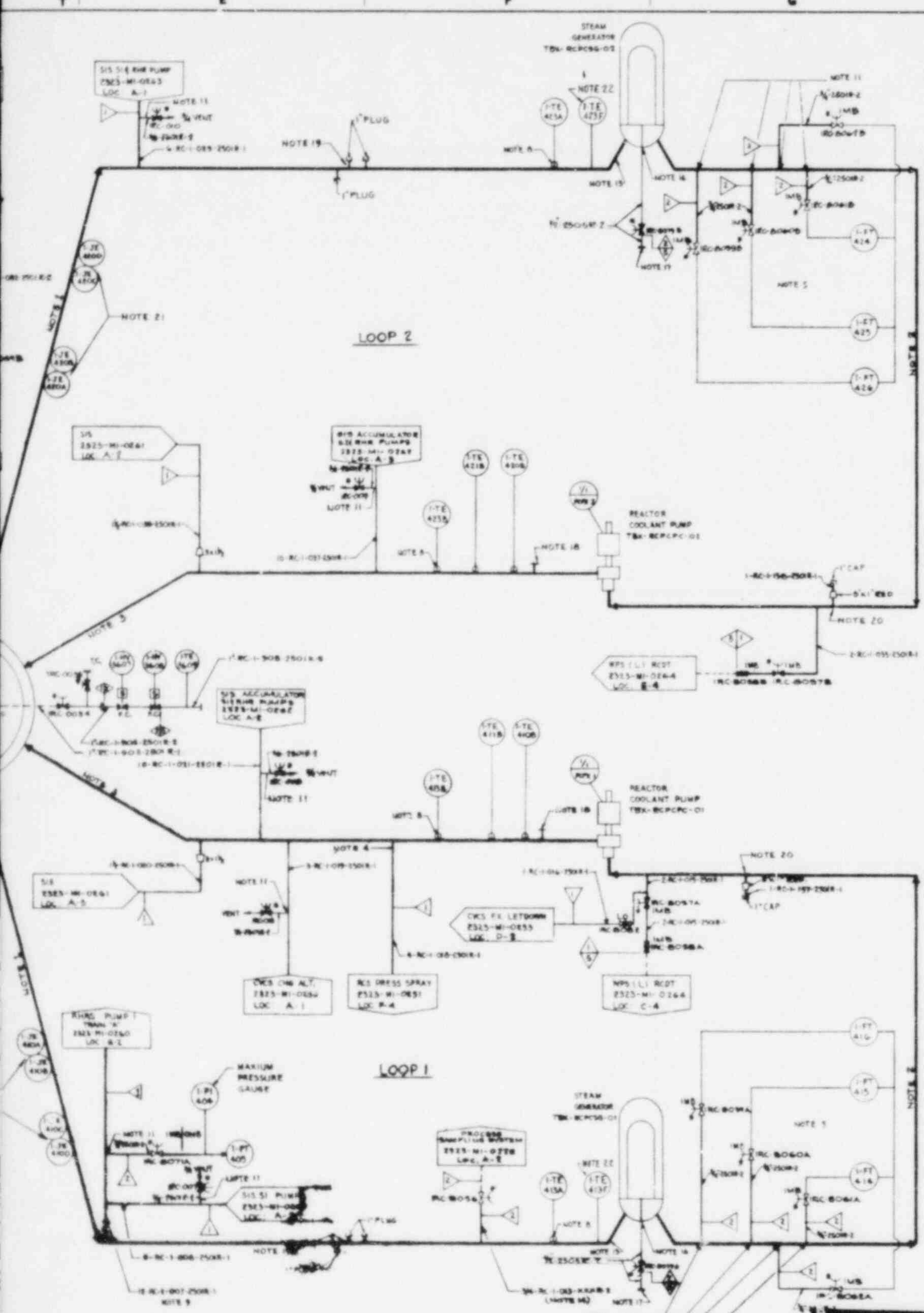
9. Skaritka, J. (Ed.), "Hybrid B_4C Absorber Control Rod Evaluation Report," WCAP-8846-A, October 1977.

10. Eggleston, F. T., "Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries," WCAP- 8768, latest revision.

11. Demario, E. E., "Hydraulic Flow Test of the 17 x 17 Fuel Assembly," WCAP-8278 (Proprietary) and WCAP-8279 (Non-Proprietary), February 1974.



REF. DWG. 2323-MI-0250 REV. 8



NOTES

1. 29" INSIDE DIAMETER. (BY WESTINGHOUSE)
2. 31" INSIDE DIAMETER. (BY WESTINGHOUSE)
3. 27.5" INSIDE DIAMETER. (BY WESTINGHOUSE)
4. SPRAY LINE SCOOP.
5. ELBOW FLOW METERS INSTALLATION-SEE REF. G. ON DWG. 2323-MI-0200.
6. VENT PIPE FURNISHED WITH REACTOR VESSEL HEAD.
7. HEAD GASKET MONITORING CONNECTIONS FURNISHED WITH REACTOR VESSEL.
8. RTD INSTALLED IN WELL.
9. CONNECTION LOCATED IN BOTTOM HALF OF REACTOR COOLANT PIPING ON 45° ANGLE TO VERTICAL.
10. LOOP IDENTIFICATION AS SHOWN.
11. 3/8 IN. ID FLOW RESTRICTOR PROVIDED (SEE MECHANICAL SYMBOLS & NOTES DWG 2323-MI-0200, NOTE 15)
12. STRAP ON (SURFACE MOUNTED) RTD LOCATED AT BOTTOM OF PIPE.
13. 31" ID x 29" REDUCING ELBOW (BY WESTINGHOUSE)
14. SPECIAL PIPE 1.050 O.D. x .434" ID. ALL OTHER REQUIREMENTS IN ACCORDANCE WITH PIPING CATEGORY 2501R.
15. [illegible] INDICATES HERMETICALLY SEALED VALVE.
16. 3/8 I.D. FLOW RESTRICTOR SUPPLIED WITH STEAM GENERATOR FOR CLASS 1 TO CLASS 2 TRANSITION SIMILAR TO ARRANGEMENT SHOWN BY NOTE 15 ON MECHANICAL SYMBOLS & NOTES DWG. Nº 2323-MI-0200.
17. TYGON HOSE TO BE INSTALLED AT THIS CONNECTION FOR DRAINING BOTTOM OF CHANNEL HEAD TO CONTAINMENT SUMP AFTER REACTOR COOLANT SYSTEM HAS BEEN DRAINED BELOW STEAM GENERATOR NOZZLES.
18. 2" NOZZLE TO BE PLUGGED IN FIELD.
19. HOT LEG BYPASS LINE SCOOPS. FOR LAYOUT SEE DWG. 2323-MI-0252
20. LOCATE CONNECTION ON UPPER 90° OF PIPE CIRCUMFERENCE.
21. LOCATE N-16 MONITOR AS CLOSE AS POSSIBLE TO BIOLOGICAL SHIELD.
22. STRAP-ON RTD'S. THESE INSTRUMENTS ARE TO BE LOCATED IN PROXIMITY OF EXISTING IN LINE RTD'S FOR CROSS-CALIBRATED PURPOSES.

REVISED PER THE FOLLOWING DOCUMENTS

CMC #	INCORP. ON DWG. REV.	DCA #	INCORP. ON DWG. REV.
82701 R0	7	10245 R0	7
78,504 R1	7	10,912 R2	7
78,505 R0	7	10,246 R1	7
78,509 R2	7	11,161 R1	7
78,510 R1	7	12,884 R0	7
78,511 R1	7		

REDRAWN WESTINGHOUSE FLOW DIAGRAM

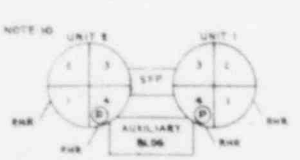
GIBBS & HILL RESPONSIBILITY FOR THIS DRAWING IS LIMITED TO ADDITIONS TO THE WESTINGHOUSE FLOW DIAGRAMS AND TO ACCURATE IMPLEMENTATION OF CHANGES CONCURRED IN BY WESTINGHOUSE.

AMENDMENT 36
DECEMBER 10, 1982

REFERENCE NOTE:

THIS FLOW DIAGRAM HAS BEEN REDRAWN FROM W DWG. Nº 1382F58 SH. 1 OF 7 REV. 6 WITH EXCEPTIONS AS FOLLOWS:

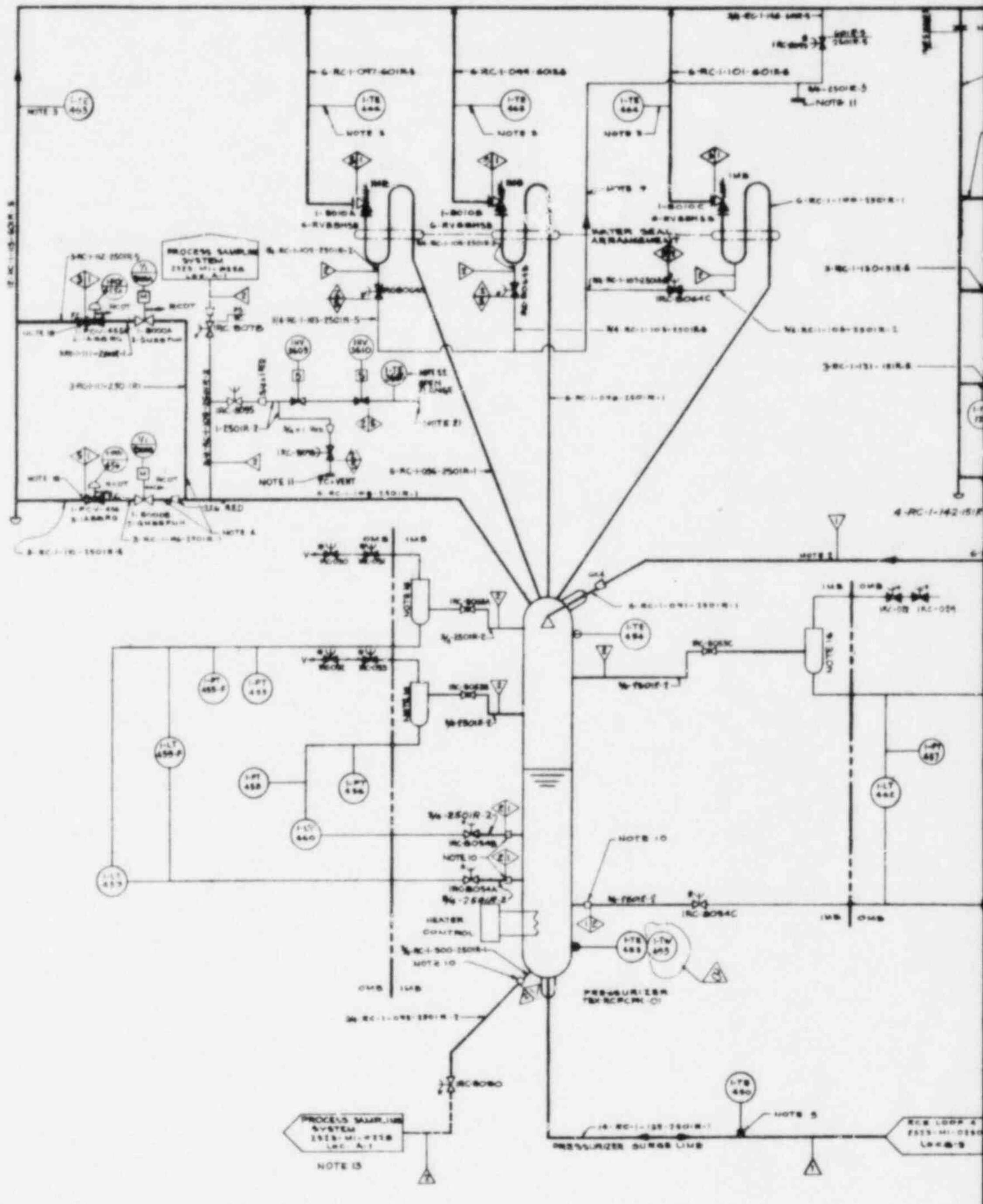
a. VALVES AND LINE NUMBERS HAVE BEEN ADDED.
b. CONTROL LOOPS HAVE BEEN DELETED EXCEPT FOR THE PRIMARY AND FINAL ELEMENTS. THE DETAILS OF THE CONTROL LOOPS WILL BE SHOWN ON INSTRUMENTATION AND CONTROL DIAGRAM.



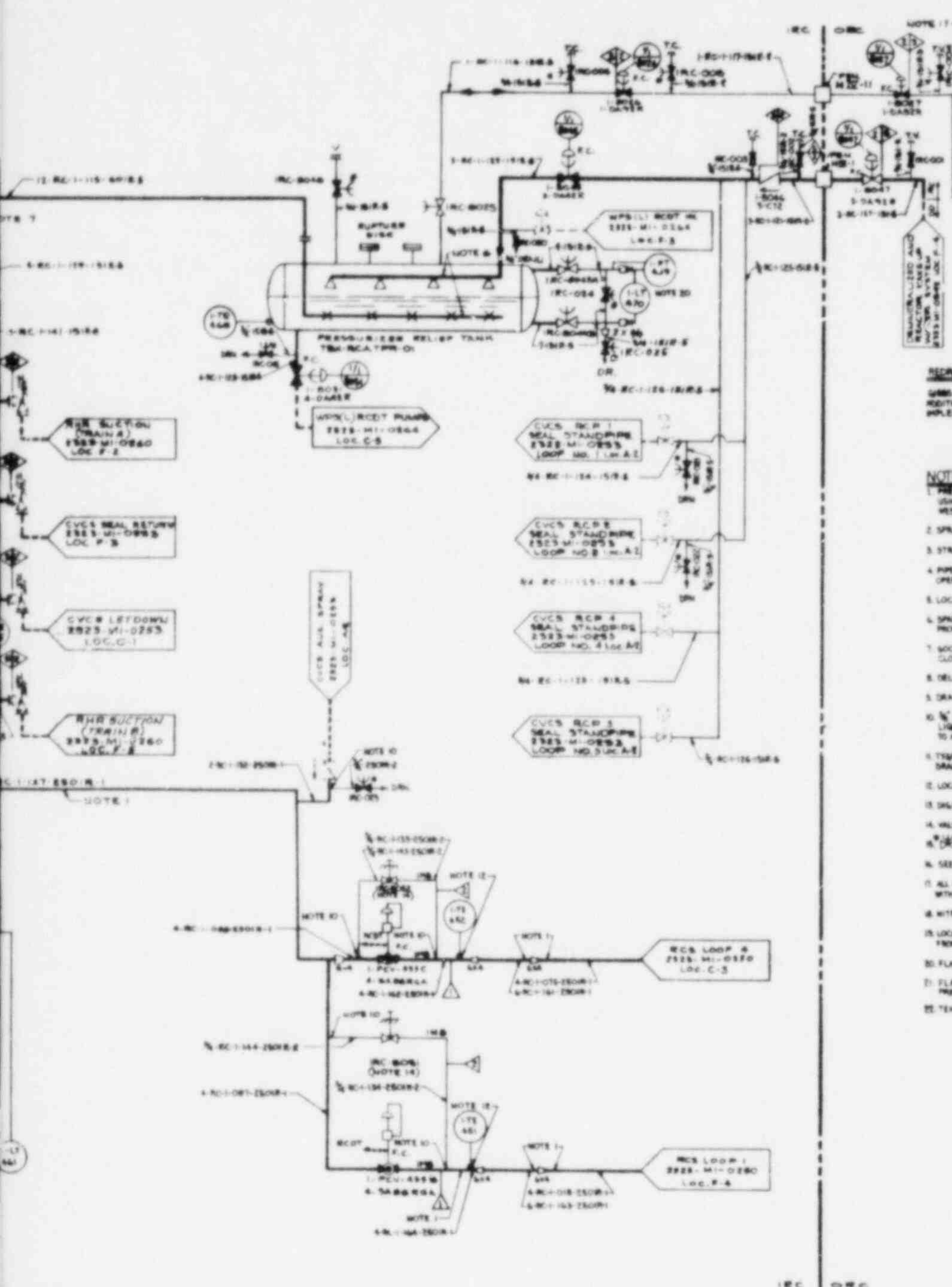
COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

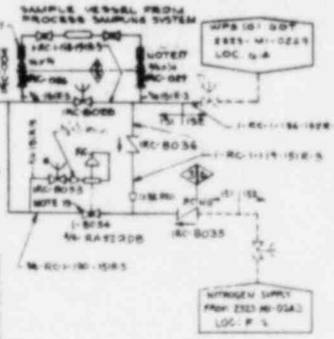
FLOW DIAGRAM
REACTOR COOLANT SYSTEM

FIGURE 5.1-1 sheet 1 of 3



REF. DWG. 2323-M1-0251 REV. 8





REDRAWN WESTINGHOUSE FLOW DIAGRAM:

GIBBS & HILL RESPONSIBILITY FOR THIS DRAWING IS LIMITED TO ADDITIONS TO THE WESTINGHOUSE FLOW DIAGRAMS, AND TO ACCURATE IMPLEMENTATION OF CHANGES CONCURRED IN BY WESTINGHOUSE.

NOTES:

1. PRESSURIZER SPRAY LINES FROM REACTOR COOLANT LOOPS TO PRESSURIZER ARE RUN USING LONG RADIUS BENDS TO COMPLY WITH MAXIMUM ALLOWABLE PRESSURE DROP PER WESTINGHOUSE DOCUMENT PFS-RD-735-4.
2. SPRAY PIPE SLOPED TO PROVIDE STEAM SEAL BETWEEN PRESSURIZER AND SPRAY VALVES.
3. STRAP ON (SURFACE MOUNTED) RTD'S LOCATED AT BOTTOM OF PIPE.
4. PIPE SLOPED DOWNWARD TO PROVIDE WATER SEAL BETWEEN PRESSURIZER AND MOTOR OPERATED VALVES.
5. LOCATED APPROXIMATELY MIDWAY BETWEEN LOOP-4 & PRESSURIZER.
6. SPARGER AND SPRAY HEADER SUPPLIED WITH PRESSURIZER RELIEF TANK. VENT HOLE PROVIDED IN SPARGER LINE.
7. 600 LB. ASA FLANGES, FLANGE END ON 12" PIPE MACHINED TO FIT CONNECTION LOCATED AS CLOSE AS POSSIBLE TO PRESSURIZER RELIEF TANK.
8. DELETED
9. DRAIN LINES SLOPED DOWN FROM LOOP SEALS TO CONNECT WITH PRT HEADER LINE.
10. ⅜" I.D. FLOW RESTRICTOR WITH PIPING FOR CLASS-1 & CLASS-2 TRANSITION AT PRESSURIZER LIQUID SPACE LEVEL INSTRUMENT AND SAMPLE NOZZLES & OTHER LOCATIONS AS SHOWN SIMILAR TO ARRANGEMENT SHOWN ON MECHANICAL SYMBOLS & NOTES DWG. No. 2323-M1-0200 NOTE #15.
11. TYGON HOSE TO BE INSTALLED AT THESE CONNECTIONS TO VENT PRESSURIZER DURING DRAINING AND FILLING.
12. LOCATED CLOSE TO PRESSURIZER SPRAY VALVES IN ITS WATER SEAL.
13. DIGITAL PRESSURIZER INDICATOR LOCATED ON THIS LINE DOWNSTREAM OF SAMPLE HEAT EXCHANGER.
14. VALVE PROVIDED WITH POSITION LOCKING DEVICE.
15. INDICATES HERMETICALLY SEALED VALVE.
16. SEE REFERENCE "G" ON MECHANICAL SYMBOLS AND NOTES DWG. No. 2323-M1-0200.
17. ALL QUICK CONNECT COUPLINGS SHALL BE SWAGELOCK SS-QC4-S-400 & SS-QC4-B-400 FURNISHED WITH SAMPLE VESSEL.
18. NITROGEN SUPPLY REQUIRED FOR COLD OVERPRESSURE PROTECTION.
19. LOCATE ABOVE THE PRESSURIZER RELIEF TANK-PIPING MUST BE SLOPED CONTINUOUSLY DOWNWARD FROM THE REGULATOR TO THE PRT TO AVOID WATER SEALS.
20. FLANGES USED IN THE INSTRUMENT CONNECTIONS FOR 1LT-470 ARE RATED AT 150 LBS.
21. FLANGE TO REMAIN OPEN DURING OPERATION. LOCATE IN OPEN AREA EXTERNAL TO PRESSURIZER AND SPRAY VALVE COMPARTMENTS.
22. TEMPERATURE ELEMENT IS STRAP-ON RTD.

REVISED PER THE FOLLOWING DOCUMENTS

CMC #	INCORP. ON DWG. REV.	DCA #	INCORP. ON DWG. REV.
3911RO	6	3932RO	6
31266R1	7	4497RO	6
31267R1	7	6901RO	6
63728RO	7	8030RO	6
64637R2	8	8971RO	6
67963RO	7	9816RO	7
		11162RO	7
		13196RO	8

REFERENCE NOTE:

THIS FLOW DIAGRAM HAS BEEN REDRAWN FROM W DWG. 1193E96 SH. 2 OF 2 REV. 6 WITH EXCEPTIONS AS FOLLOWS:
a. VALVE AND LINE NUMBERS HAVE BEEN ADDED.
b. CONTROL LOOPS HAVE BEEN DELETED EXCEPT FOR THE PRIMARY AND THE FINAL ELEMENTS. THE DETAILS OF THE CONTROL LOOPS WILL BE SHOWN ON INSTRUMENTATION AND CONTROL DIAGRAM.

AMENDMENT 36
DECEMBER 10, 1982

COMANCHE PEAK S.E.S.
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UNITS 1 and 2

FLOW DIAGRAM
REACTOR COOLANT SYSTEM

FIGURE 5.1-1 sheet 2 of 3

center of the Containment. Butterfly valve disc leakage is the same in either direction due to the symmetrical design of the valve.

2. In valve arrangements 2 and 28 of Figure 6.2.4-1, the gate valve is tested towards the center of the Containment. The leakage of these gate valves is the same in either direction due to symmetrical design of the valves.

3. In valve arrangement 22 of Figure 6.2.4-1, the diaphragm valve inside containment will be tested toward the center of the
Containment. Diaphragm valve leakage is the same in either 12
direction due to the symmetrical design of the valve.

Containment isolation valve leakage rates are evaluated by methods discussed in Section 6.2.6.3.

Environmental qualification tests performed on the Containment Isolation System components are discussed in Section 3.11.

6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

Following a DBA, hydrogen gas may be generated inside the Containment by reactions such as zirconium metal with water, corrosion of materials of construction and radiolysis of aqueous solution in the core and sump. The following section is presented to describe the design of the Combustible Gas Control System. To ensure that the hydrogen concentration is maintained at a safe level, redundant, electrical hydrogen recombiners and a backup Hydrogen Purge System are provided in accordance with NRC Regulatory Guides 1.7 [5], 1.22 [6], 1.26 [7] and 1.29 [8], General Design Criteria 41, 42, 43, and 50 [1] [2] [3] [4], and Branch Technical Positions 6-2 [11] and 9.2 [10].

6.2.5.1 Design Bases

6.2.5.1.1 Generation, Accumulation, and Mixing of Combustible Gases

1. A combustible mixture can be formed when hydrogen gas
concentration in the Containment atmosphere is greater than four
36 volume percent (v/o). Following a LOCA, hydrogen gas is
generated and builds up inside the Containment as described in
Appendix 6.2.5A.

2. The Containment Spray System (CSS), which is used during the long term recirculation phase, ensures adequate mixing of hydrogen in the Containment atmosphere. In addition to the driving force of falling drops, the spray enhances natural air circulation by causing a temperature gradient. As a result, no recirculation fans are needed to mix the Containment atmosphere.

3. A large volume of hydrogen is generated by radiolysis in the core and the sump and is released in the compartment where the LOCA occurred and in the Containment sump where the coolant is collected.

   All subcompartments are provided with vents at the top and drains at the bottom. The vents provide for the release, caused by buoyancy, of any hydrogen generated within or beneath the subcompartment. The drains prevent the accumulation of water within a subcompartment, thus preventing substantial generation of hydrogen by radiolysis within that subcompartment.

   Arrangement of the subcompartments with bottom and top openings creates a stack effect. In addition to the driving forces generated by diffusion rate, the natural ventilation going through the subcompartments provides mixing and avoids hydrogen stratification. Therefore, the flow caused by the stack effect yields a hydrogen concentration within a subcompartment that does not substantially differ from the bulk containment conditions the flow caused by stack effect.

6.2.5.1.2 Electric Hydrogen Recombiners

The following design bases apply to the electric hydrogen recombiners:

1. The recombiners are designed to sustain all normal loads as well as accident loads including safe shutdown earthquake (SSE) and pressure-temperature transients from a design basis LOCA.

2. The recombiners are designed for a lifetime of 40 years, consistent with that of the plant.

3. All materials used in the recombiners are selected to be compatible with the environmental conditions inside the Containment Building during normal operation and during accident conditions.

4. The recombiners are located so that there is adequate area around the units for maintenance.

5. The recombiners are protected from damage by missiles or jet impingement from broken pipes.

6. The recombiners either are located away from high velocity air streams such as could emanate from fan cooler exhaust ports or are protected from direct impingement of such high velocity air streams by suitable barriers.

7. Process capacity is such that the Containment hydrogen concentration does not exceed 4 v/o based on the release model indicated in Regulatory Guide 1.7.

8. Two redundant, electric hydrogen recombiners are provided to meet the single failure criterion.

9. Inspection and testing of the electric hydrogen recombiners are made periodically. For further details, see Chapter 16, Technical Specifications.

10. Each Containment Building is provided with separate and independent permanently installed hydrogen recombiners.

11. The hydrogen recombiners are located in the Containment Building, which is inaccessible to plant personnel during an accident. Therefore, personnel protection from radiation in the vicinity of the operating units is not necessary.

12. The recombiners are mounted on a substantial foundation with no normal, ambient vibration.

6.2.5.1.3 Hydrogen Purge System

The following design bases apply to the Hydrogen Purge System:

The Hydrogen Purge System functions as a backup system for the electric hydrogen recombiners. This system provides the hydrogen removal capacity required by NRC Regulatory Guide 1.7, which defines a concentration limit for hydrogen accumulation following a LOCA.

As required by GDC 41, the system is designed to maintain the hydrogen concentration in the Containment below the lower flammability limit following an accident.

The Hydrogen Purge System has a process capacity of 700 scfm, which is sufficient to ensure that Containment hydrogen concentration will not exceed 3 v/o based on the NRC model as indicated in NRC Regulatory Guide 1.7. For hydrogen generation refer to Subsection 6.2.5.3.1.

Purging the Containment atmosphere through high-efficiency particulate air (HEPA) and iodine adsorbers limits the potential release of radioactive materials such that offsite radiation doses are within 10 CFR Part 100 limits.

The system components are designed for SSE loads and maximum temperature and pressure transients from a DBA. (For maximum temperature and pressure, see Subsection 6.2.1.)

The system is capable of operating with a Containment pressure range of 0 to 5 psig and temperature range of 50 to 160^{o}F.

Protection is provided to preclude damage by missiles. All materials are selected to be compatible with accident and normal operating environments.

6.2.5.1.4 Containment Hydrogen Monitoring System

The Containment Hydrogen Monitoring System measures the concentration of combustible gases in four well-ventilated areas of the Containment Building in order to obtain a typical value for hydrogen gas concentration.

The plant has two hydrogen monitoring systems. Each monitoring system consists of four (4) sensor modules and one (1) microprocessor analyzer. Of the four (4) sensor modules in each system, two (2) are located in each containment. The microprocessor analyzer is thus shared by Units 1 and 2. The system can be operational within 30 minutes after an accident and samples four independent sources in the Containment. The hydrogen gas analyzers alarm at 3 volume percent (wet) hydrogen. 36

The sensor modules and microprocessors are qualified to function under Seismic Category I requirements and post accident conditions as described in CPSES FSAR Appendix 3A, Table 5-1.

6.2.5.2 System Design

The primary means of reducing hydrogen concentration in the Containment following a LOCA are by the use of the electric hydrogen recombiners. A Hydrogen Purge System acts as a backup to the hydrogen recombiners. A Containment Hydrogen Monitoring System is provided to sample the containment atmosphere in various locations to determine the hydrogen concentration.

6.2.5.2.1 Electric Hydrogen Recombiners

The applicable codes, standards, and Regulatory Guides used in the design of the electric hydrogen recombiners are listed in Table 6.2.5-1. Redundant recombiners as shown on Figure 6.2.5-1 are located inside the Containment Building. The recombiner units are located in the Containment in such a way that they process a flow of Containment air containing hydrogen at a concentration which is generally typical of the average concentration throughout the Containment.

To meet the requirements for redundancy and independence, two electric hydrogen recombiners are provided for each Containment Building. Each recombiner is provided with a separate power panel and control panel, and is powered from a separate safeguards bus. There is no interdependency between this system and the other engineered safety features systems.

Containment atmosphere is circulated through the recombiner by natural circulation where hydrogen is removed by heating to a temperature sufficient to cause recombination with the containment oxygen.

The recombiner consists of a thermally-insulated, vertical metal duct with electric resistance metal-sheathed heaters provided to heat a continuous flow of containment air (containing hydrogen) to a temperature which is sufficient to cause a reaction between hydrogen and oxygen. The recombiner is provided with an outer enclosure to keep out containment spray water. The recombiner consists of an inlet preheater section, a heater-recombination section, and a discharge mixing chamber that lowers the exit temperature of the air.

The unit is manufactured of corrosion-resistant, high-temperature material except for the base which is carbon steel. The electric hydrogen recombiner uses commercial-type electric reistance heaters sheathed with Incoloy-800, which is an excellent corrosion-resistant material for this service. These recombiner heaters operate at significantly lower power densities than in commercial practice.

Air is drawn into the recombiner by natural convection and passes first through the preheater section. This section consists of a shroud placed around the central heater section to take advantage of heat conduction through the walls to preheat the incoming air. This process accomplishes the dual function of reducing heat losses from the recombiner and of preheating the air.

The warmed air passes through an orifice plate and then enters the electric heater section where it is heated to approximately 1150 to 1400^{0} F, thus causing recombination to occur. Tests have verified that the recombination is not a catalytic surface effect associated with the heaters but occurs as a result of the increased temperature of the process gases. Since the phenomenon is not a catalytic effect, saturation of the unit by fission products does not occur. The heater section consists of five assemblies of electric heaters stacked vertically with each assembly containing individual heating elements. Table 6.2.5-2 gives the recombiner design parameters. 13 Q032.85

Operation of the recombiner is done manually from a control panel located in an accessible area outside the Containment.

The recombiner, power supply panel, and control panel are shown schematically on Figure 6.2.5-2. The power panel for the recombiner contains an isolation transformer plus an SCR controller to regulate power into the recombiner. This equipment is not exposed to the post-LOCA environment. To control the recombination process, the correct power input to bring the recombiner above the threshold temperature for recombination is set on the controller. The correct power required for recombination depends upon containment atmosphere conditions and is determined when recombiner operation is required. For equipment tests and periodic checkouts, a thermocouple readout instrument is also provided in the control panel for monitoring temperatures in the recombiner.

Reference [12] provides a description of the testing of a full-scale prototype electric hydrogen recombiner.

6.2.5.2.2 Hydrogen Purge System

The Hydrogen Purge System shown in Section 9.4 on Figure 9.4-6 is common to both units and is designed to be completely independent of the Hydrogen Recombiner System. It is not considered credible that an accident could happen to both the recombiners and the Hydrogen Purge System. It is also not considered credible that a LOCA could incapacitate both plants simultaneously. Therefore, no safety problems related to sharing between the two units are anticipated. This system is not used during normal operation but is capable of operating intermittently or continuously after an accident. The Hydrogen Purge System consists of two 700 scfm blowers for supply, inlet and outlet ductwork, and piping, isolation valves, two atmospheric cleanup systems, and two exhaust fans. The blowers are capable of transporting 700 scfm of the fresh, filtered air to the Containment. Air is drawn from either Containment as required, passed through a filterplenum (particulate, iodine adsorbers, HEPA filters) and discharged through the plant discharge duct. A demister and heater are used to maintain the humidity entering the filters below 70 percent. Two trains are provided (one train is required to operate), each capable of exhausting the design airflow of 700 scfm.

The Hydrogen Purge System is manually operated and is normally isolated from the Containment by locked-closed valves. Mixing of the containment atmosphere is by natural convection.

Any radioactivity discharged is measured by the plant discharge duct monitoring system.

The expected efficiencies of the filters are in accordance with NRC Regulatory Guide 1.52.

A conservative value of 95 percent for both elemental and organic iodide is used in the calculations for the dose at the exclusion area boundary presented in Section 15.6.5.4.

6.2.5.2.3 Containment Hydrogen Monitoring System

The post-LOCA hydrogen concentration in each containment is monitored by four (4) sensors located on four (4) different elevations of the containment. Two (2) sensors from each containment are coupled to one of the two hydrogen analyzer microprocessors located in the control room. Each microprocessor is supplied from a safety-related uninterrupted power supply train. Thus, two independent analysis trains, each monitoring two points inside the containment, are provided for measurement. 36

The analyzers continuously monitor the hydrogen content of the containment atmosphere within 30 minutes after a LOCA and for the duration of the accident.

The analyzer system meets with the following requirements:

Sensitivity	0.1 percent hydrogen by volume
Accuracy	$\pm$2.0 percent of full scale
Range	0-10 percent hydrogen by volume
Calibration	Fully automatic sequencing for feeding known gaseous mixtures to the sensor modules and adjustment

The sensor modules are of the in-containment measurement type using an electrochemical sensor for specific measurement of hydrogen partial pressure.

Each sensor module consists of the following major components mounted on an integral rack: hydrogen sensor, calibration mechanism, calibration gas bottles, solenoid valves (calibration gas isolation),
36 RTD temperature transducer and an electronics interface terminal box. One absolute pressure transducer is provided with each pair of sensor modules. This transducer is mounted on one of the sensor modules.

The analyzer microprocessor modules accept, process and condition the sensor output signal. The microprocessor has a digital display of the following:

- Hydrogen volume percent (wet)
- Hydrogen volume percent (dry)
- Hydrogen partial pressure
- Temperature
- Pressure

The control room operators are able to select any display for instantaneous or continuous readout. The microprocessors also have two buffered 0-10 volt dc output signals for remote analog display of hydrogen volume percent (wet) on the Main Control Board.

The alarms from the microprocessor modules are from solid state relays and indicate the following conditions: high hydrogen concentration, power failure and system error. 36

The Containment Hydrogen Monitoring System is designated as IEEE Class 1E and qualified per requirements of IEEE 323-1974.

6.2.5.3 Design Evaluation

6.2.5.3.1 Hydrogen Generation

Calculations of hydrogen generation following a LOCA show that although the hydrogen production rate decreases with time following an accident, the hydrogen accumulation can exceed the lower flammability level of 4 volume percent. Therefore, control measures are implemented to prevent hydrogen accumulation to this level. 36

The potential sources of hydrogen, method of analysis, and typical assumptions are described in Appendix 6.2.5A. 36

The following sources were used as input parameters for the hydrogen accumulation calculations:

1. Zirconium Weight 36

   Weight of zirconium cladding: 43,204 lb 36

2. Corrosion Rates As a Funtion of Time

   a. For aluminum, hydrogen generated as a result of aluminum corrosion by spray is based on corrosion data obtained experimentally by ORNL (Reference 8, Section 6.2.5A). The long-term rate assumed for these calculations is 200 mils per year. The short-term corrosion rate is determined from 36 the post-LOCA temperature transient given in Table 6.2.5A-1 and a curve of corrosion rate versus temperature given in Appendix 6.2.5A on Figure 6.2.5A-1.

b. For zinc (paint): hydrogen generated by corrosion of paint
36 containing zinc is conservatively based on the corrosion of galvanized material.

c. For zinc (galvanized): Galvanized ductwork is assumed to have 2 oz of zinc per ft^2 of galvanizing. The curve of corrosion rate versus temperature for zinc was derived experimentally by Westinghouse (Reference 9, Section 6.2.5A). 36

3. Surface Area and Weight

For aluminum, zinc and zinc paint see Appendix 6.2.5A, Table 6.2.5A-2.

4. Hydrogen in the Primary Coolant at Start of an Accident

The total hydrogen within the primary system boundary, 1191 scf, is the sum of the hydrogen dissolved in the primary coolant water and that which is in the pressurizer gas space. This value is based on the 35 cm^3/kg (STP) in the primary coolant.

Assumptions for the pressurizer vapor space hydrogen are presented in Section 6.2.5A.1. 36

6.2.5.3.2 Hydrogen Mixing

As described in Subsection 6.2.5.1.1, all subcompartments are provided with vents to aid in hydrogen mixing and to avoid high concentration pockets of hydrogen. These vents cause a stack effect which maintains the subcompartments at virtually the same hydrogen concentration as the remainder of the Containment.

This stack effect is governed by the following formula:

$$Q = 7.2A\ ht$$

where

Q = air flow (ft^3/min)

A = area of bottom or top openings, whichever is smaller (ft^2)

h = height from bottom to top openings (ft)

t = temperature difference between the subcompartment and the bulk containment atmosphere (F)

7.2 = constant of proportionality, for conditions not favorable

At a minimum temperature gradient of $1^{o}F$, the number of air changes per hour is always in the range of 2 to 3.

With regard to mixing of post-LOCA hydrogen, the following aspects have been considered.

1. Mixing in the bulk Containment above operating floor at elevation of 905 ft 9 in.

   Experimental results from spray experiments conducted at Oak Ridge National Laboratories have substantiated the adequacy of the sprays to ensure mixing in the bulk Containment volume. These results apply to the Region A described in Section 6.5, Figure 6.5-2.

2. Subcompartments are enclosed between floor elevations 808 ft 0 in., 832 ft 6 in., 860 ft 0 in., and 905 ft 9 in.

These regions are described in Section 6.5 and shown on Figure 6.5-2, where they are referenced as Region B, C, and D. To avoid accumulation of hydrogen between floors, each of them is provided with openings to permit mixing with the bulk Containment.

Between the Region D and the other regions, the total opening is approximately 900 ft^2. Although the hottest subcompartment is the steam generator subcompartment, a temperature difference between the upper Containment and the lower floors exists as a result of the cold water sprayed in the dome which induces natural convection. This natural convection ensures a general mixing of hydrogen in the Containment.

These three regions are partially sprayed by nozzles (see Section 6.5, Table 6.5-5) during the injection and recirculation phases. In addition, the driving force of falling drops enhances air circulation.

3. Subcompartments Where LOCA Occurs

These subcompartments are the steam generator subcompartments, the connecting pipe tunnel to reactor vessel cavity, and the pressurizer relief tank subcompartment.

A large part of the hydrogen generated in the Containment is released in the subcompartment where the break occurs, as a result of radiolysis in the core.

After 30 days, the hydrogen flow rate from core radiolysis is 0.65 scfm. The flow rates from sump radiolysis and corrosion are only 0.159 scfm, and 0.058 scfm, respectively.

a. Steam Generator Subcompartments

Each steam generator subcompartment is fully open at the top and provided with two main openings in the bottom (for communication with another steam generator subcompartment and personnel access). This arrangement has the following effects:

1) Coverage of the steam generator subcompartment with spray, ensuring a mixing within the subcompartment

2) Release of hydrogen through the top as a result of its low atomic weight

3) Mixing of the contents of the steam generator subcompartment with the contents of the bulk Containment through natural convection effects. A sensible energy is introduced during the long-term with the core recirculation flow in the subcompartment and provides the driving forces for mixing

b. Reactor Cavity

Reactor coolant pipe tunnels connecting the steam generator subcompartments with the reactor vessel cavity have postulated pipe break locations as discussed in Section 6.2.1. If a break occurs in the cavity seal, a very limited space, the reactor coolant pipe weld inspection plugs blow out to limit the peak differential pressure. Part of the fluid spilled out of the break and part of the water volume sprayed in the refueling cavity fill the reactor cavity. Hydrogen generated in the cavity by radiolysis is released through the gap around the reactor vessel and through the relief openings located in each floor (elevations 808 ft 0 in., 832 ft 6 in., and 849 ft 0 in.).

c. Pressurizer Relief Tank Subcompartment

This subcompartment has a bottom drain opening of 100 ft^2. Hydrogen released within the subcompartment and hydrogen entering the subcompartment is vented through the top vents.

4. Subcompartments Where There Is No LOCA

a. Pressurizer Subcompartment

Hydrogen entering this subcompartment is vented through a 36-ft^2 top vent. This opening is in the vertical wall underneath the slab. Hydrogen is released above elevation 905 ft 9 in. in Region A, which is the bulk Containment volume (see Section 6.5, Figure 6.5-2).

b. Cubicles at Elevation 808 ft 0 in.

These cubicles are for excess letdown heat exchanger, reactor coolant drain tank and pumps, and reactor coolant drain tank heat exchanger. The Containment Building arrangement is such that the sump water will enter these cubicles and will generate hydrogen which could accumulate in the upper parts.

Consequently, all cubicles located at elevation 808 ft 0 in. are provided with a vent at their highest points to avoid local high level hydrogen concentration.

6.2.5.3.3 Electric Hydrogen Recombiners

Diagrams of the hydrogen production rate following the LOCA (refer to
36 Figure 6.2.5A-5 in Appendix 6.2.5A) show that although hydrogen production rate decreases with time after the loss-of-coolant, accident total hydrogen accumulation can exceed the lower flammability limit of 4 v/o and positive measures are necessary to limit hydrogen accumulation to acceptable levels. The electric recombiner provides the means to prevent unsafe levels of hydrogen concentration from being reached in the Containment following a LOCA.

For the purpose of showing that the electric recombiner is capable of maintaining safe hydrogen concentrations, an analysis was performed using the NRC Regulatory Guide 1.7 model. The result for the Containment volume is shown on Figure 6.2.5A-9. The NRC Regulatory Guide 1.7 model is based upon assuming a fission product activity release specified in Reference [13] and the values for postaccident hydrogen generation specified in this guide. The results using the Westinghouse model are also shown on this figure. Refer to Appendix 6.2.5A for further information. | 36

Each electric recombiner is capable of continually processing a minimum of 100 scfm of containment atmosphere. The hydrogen contained in the processed atmosphere is converted to steam which then exits to the containment atmosphere, thus reducing the overall containment hydrogen concentration. The hydrogen concentration in the Containment calculated for the previously described models is based on a recombiner capability of processing 100 scfm of containment atmosphere. This calculation shows that the maximum hydrogen concentration will be much less than the lower flammability limit of 4 v/o if the recombiner is started one day following the accident. Therefore, one of these units meets the design criterion of maintaining a safe hydrogen concentration with considerable margin, and the second unit provides a redundant system of equal capability on a redundant power supply.

The peak hydrogen concentration occurs when the amount of hydrogen being generated is equal to the amount of hydrogen being reprocessed. Since the production rate of hydrogen decreases with increasing time following the accident, once this peak has been reached the recombiner processess hydrogen at a faster rate than it is being produced, resulting in an overall reduction of the hydrogen concentration inside the Containment. Thus, once the peak has been reached, the electric recombiner provides a continually increasing margin between the containment hydrogen concentration and the lower flammability limit of 4 v/o.

The unit is designed to sustain all normal loads as well as accident loads such as seismic loads and temperature and pressure transients from a LOCA.

For further information on hydrogen production and accumulation, see Appendix 6.2.5A.

6.2.5.3.4 Hydrogen Purge System

The Hydrogen Purge System, which is used as a backup to the electric hydrogen recombiners, is capable of maintaining by purging a safe hydrogen concentration (below 3 v/o) in the Containment.

The system is capable of continually or intermittently processing a minimum of 700 scfm.

If the system is started approximately 14 days following the accident, the maximum hydrogen concentration is approximately 3 v/o, with the system operating three hr per day and a hydrogen accumulation based on the values given in Subsection 6.2.5.3.1.

If a supply blower or exhaust fan fails, redundant fans will be able to supply or exhaust air by changing the valve and damper arrangement. Air supply and exhaust lines are arranged so as not to be rendered inoperative by accumulation of water in the line from containment spray, condensation, or flooding.

All equipment is leaktight, and the filter housings are designed to facilitate replacement without undue exposure of personnel to radioactive sources. An analysis of the predicted doses at the exclusion area boundary that would result from Containment purging in the event of a design basis LOCA is provided in Section 15.6.5.4.

The Hydrogen Purge Air Filtration System, being an ESF, conforms to each regulatory position established in NRC Regulatory Guide 1.52 as shown in Section 6.5 and Table 6.5-1.

The supply and exhaust lines are routed through different Containment penetrations. Each fan is connected to an emergency standby diesel generator bus. (Section 8.3)

The equipment in the Containment and outside the Containment up to and including the Containment isolation valves is ANS Safety Class 2, and the equipment beyond the isolation valves is ANS Safety Class 3 [9].

6.2.5.3.5 Containment Hydrogen Monitoring System

The Containment Hydrogen Monitoring System is capable of determining the hydrogen concentration at four elevations in the Containment. Four sensor modules and two microprocessors analyzers are provided to ensure that sufficient redundancy is available. 36

6.2.5.4 Tests and Inspections

Test programs for preoperational testing and periodic testing are implemented. The inservice inspection as part of surveillance tests is conducted periodically throughout the life of the plant to verify that hydrogen recombiners are ready to perform their safety function.

1. Electric Hydrogen Recombiners

   The electric hydrogen recombiners have undergone extensive testing in the Westinghouse development program. These tests encompassed the initial analytical studies, laboratory proof-of-principal tests, and full-scale prototype testing. The full-scale prototype tests included the effects of:

a. Varying hydrogen concentrations

b. Alkaline spray atmosphere

c. Steam

d. Convection currents

e. Seismic activity

A detailed discussion of these tests is given in Reference [12].

Postoperational tests and inspections are performed in accordance with Technical Specification requirements. Inspections are performed to ensure the capability of the recombiner to perform its function. Testing is performed to verify operation of the control system and to verify functional performance of the heaters to the required temperature level.

2. Hydrogen Purge System

Component qualification tests demonstrate the characteristics of materials incorporated into components (e.g., efficiency of charcoal filter).

Component acceptance tests demonstrate the capability of the components incorporated. Fans are tested by the manufacturers to determine that their characteristic curves are within design limits.

A post-installation test is performed to demonstrate system compliance with design requirements. During this test, fans are tested in accordance with the standards of the Air Moving and Conditioning Association (AMCA), and filters are tested in accordance with the standards of the ANSI N101.1 [14].

After installation the Hydrogen Purge System can be tested. The system is normally idle, but periodic tests demonstrate its ability to function.

3. Containment Hydrogen Monitoring System

The Containment Hydrogen Monitoring System is calibrated when installed and periodically checked in accordance with manufacturer's instructions.

The analyzer will be checked with known calibration gases containing 2 and 6 percent hydrogen in high purity nitrogen to ensure minimal deviation from the calibration curve. The calibration is controlled by the microprocessor analyzers after manual initiation. 36

The Containment Hydrogen Monitoring System will be field tested in accordance with Regulatory Guide 1.68, "Preoperational and initial Startup Test Programs for Water-Cooled Power Reactors."

6.2.5.5 Instrumentation Requirements

The recombiners do not require any instrumentation inside the Containment for proper operation after a LOCA. The recombiners are started manually after a LOCA. The hydrogen monitoring system is used in determining containment hydrogen concentration that indicate when the recombiners or the venting system should be actuated. This measurement can be taken from any of four sensor locations within the 36

Containment. Control measures can be initiated when the hydrogen concentration reaches 3 volume percent (wet). A 3 volume percent (wet) hydrogen concentration initiates an alarm in the Control Room thereby alerting the operator. Instrumentation is provided to both monitor the hydrogen concentration in the Containment and to monitor the Hydrogen
36 Purge System operation. Two hydrogen indicators are provided, one mounted on the Main Control Board and the second one on the microprocessor analyzer.

The hydrogen purge supply and exhaust fans are manually started from the Control Room. A humidity control heater located in the filter is interlocked with the fan to come on when the fan is started and to shut off when the fan is stopped. A thermistor is provided on the discharge side of the iodine adsorber to provide a high temperature signal to the Fire Protection Systems panel. Differential pressure indicating switches are provided to monitor the differential pressure across the fans and filter banks. A low alarm is annunciated from the fan switch and a high alarm from the filter bank.

6.2.5.6 Materials

The materials of construction for the electric recombiner are selected for their compatibility with the post-LOCA environment.

The major structural components are manufactured from 300-Series stainless steel except for the base which is carbon steel. Incoloy-800 is used for the heater sheaths and Inconel-600 for other parts such as the heat duct, which operates at high temperature.

There are no radiolytic or pyrolytic decomposition products from these materials. The carbon steel base of the recombiner unit is coated with a paint that satisfies the requirements of ANSI N101.2 (1972) [15]. Materials of construction for Containment Hydrogen Purge System components are listed in Section 6.1B, Table 6.1B-1.

REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 41, Containment Atmosphere Cleanup.

2. 10 CFR Part 50, Appendix A, General Design Criterion 42, Inspection of Containment Atmosphere Cleanup Systems.

3. 10 CFR Part 50, Appendix A, General Design Criterion 43, Testing of Containment Atmosphere Cleanup Systems.

4. 10 CFR Part 50, Appendix A, General Design Criterion 50, Containment Design Basis.

5. NRC Regulatory Guide 1.7, Revision 1, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident, and the Supplement to NRC Regulatory Guide 1.7, March 1977, U.S. Nuclear Regulatory Commission. 36

6. NRC Regulatory Guide 1.22, Periodic Testing of Protection System Actuation Functions, February 1972, U.S. Nuclear Regulatory Commission.

7. NRC Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants, Rev. 3, February 1976, U.S. Nuclear Regulatory Commission.

8. NRC Regulatory Guide 1.29, Seismic Design Classification, Rev. 2, February 1976, U.S. Nuclear Regulatory Commission.

9. ANSI N18.2, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, 1973.

10. Branch Technical Position APCSB 9-2, Residual Decay Energy for Light Water Reactors for Long-Term Cooling.

11. Branch Technical Position CSB 6-2, Control of Combustible Gas Concentrations In Containment Following a Loss of Coolant Accident, Nov. 24, 1975.

12. J. F. Wilson, Electric Hydrogen Recombiner for PWR Containments, WCAP-7709 (Proprietary) and WCAP-7820, Supplements 1, 2, 3, 4 and 5 (Nonproprietary).

13. TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites, March 23, 1962.

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6.2.5A HYDROGEN PRODUCTION AND ACCUMULATION

Hydrogen accumulation in the containment atmosphere following the design basis accident can be the result of production from several sources. The potential sources of hydrogen are hydrogen dissolved in the reactor coolant, the zirconium-water reaction, corrosion of construction materials, and radiolytic decomposition of the emergency core cooling and sump solutions. 36

6.2.5A.1 Method of Analysis

The quantity of zirconium which reacts with the core cooling solution depends on the performance of the Emergency Core Cooling System (ECCS).

The criteria for evaluation of the ECCS requires that the zircaloy-water reaction be limited to 1 percent by weight of the total quantity of zirconium in the core. ECCS calculations have shown the zircaloy-water reaction to be less than 0.3 percent, much less than required by the criteria.

The use of aluminum inside the containment is limited, and is not used in safety-related components which are in contact with the recirculating core cooling fluid. Aluminum is much more reactive with the containment spray alkaline borate solution than other plant materials such as galvanized steel, copper and copper nickel alloys.

It should be noted that the zirconium-water reaction, and aluminum and zinc corrosion with containment spray are chemical reactions and thus essentially independent of the radiation field inside the containment following a loss of coolant accident. Radiolytic decomposition of water is dependent on the radiation field intensity. The radiation field inside the containment is calculated for the maximum credible accident in which the fission product activities given in TID-14844[1] are used.

The hydrogen generation calculations are performed using the NRC model
36 discussed in Regulatory Guide 1.7[2]. Plant specific parameters, used
in the hydrogen generation calculations, and the assumptions of
Regulatory Guide 1.7 are summarized in Table 6.2.5A-2.

6.2.5A.2 Assumptions

The following discussion outlines the assumptions used in the calculations.

1. Zirconium-water reaction

   The zirconium-water reaction is described by the chemical equation:

   $$Zr + 2H_2O \rightarrow Zr\,O_2 + 2H_2 + \text{Heat}$$

   The hydrogen generation due to this reaction will be completed during the first day following the loss of coolant accident. The NRC model assumes a 1.5 percent zirconium-water reaction. The hydrogen generated is assumed to be released immediately to the containment atmosphere.

36 2. Hydrogen from the Reactor Coolant System

   The quantity of hydrogen contained in the Reactor Coolant System during steady state operation is 1191 SCF. This includes hydrogen from the pressurizer gas space. The pressurizer gas space hydrogen is based on:

   a. A reactor coolant hydrogen concentration of 35 CC(STP)/kg of coolant.

b. Normal pressurizer heaters turned on 50 percent of the time and all of the heat input goes to the boiling water.

c. Minimum bypass spray rate of 1.0 GPM.

d. Normal liquid level in the pressurizer (60%).

e. Pressurizer relief valves being closed.

The reactor coolant chemistry specifications specify a maximum coolant hydrogen concentration of 35 cm^3/kg (STP) of coolant during steady state plant operation above 1 MW power level. The hydrogen from the reactor coolant system is available for release to the containment immediately following a LOCA. 36

3. Corrosion of plant materials

Oxidation of metals in aqueous solution results in the generation of hydrogen gas as one of the corrosion products. Extensive corrosion testing has been conducted to determine the behavior of the various metals used in the containment in the emergency core cooling solution at design basis accident conditions. Metals tested include Zircaloy Inconel aluminum alloys, cupronickel alloys carbon steel, galvanized carbon steel, and copper.

Tests conducted at Oak Ridge National Laboratories (ORNL)[3 and 4] have also verified the compatibility of the various materials (exclusive of aluminum) with alkaline borate solution. As applied to the quantitative definition of hydrogen production rates the results of the corrosion tests have shown that only aluminum and zinc will corrode at a rate that will significantly add to the hydrogen accumulation in the containment atmosphere.

The corrosion of aluminum may be described by the overall reaction:

$$2\ Al + 3\ H_2O \rightarrow Al_2O_3 + 3\ H_2$$

Therefore, three moles of hydrogen are produced for every two moles of aluminum that is oxidized. (Approximately 20 standard cubic feet of hydrogen for each pound of aluminum corroded.)

The corrosion of zinc may be described by the overall reaction:

$$Zn + 2H_2O \rightleftarrows Zn(OH)_2 + H_2$$

Therefore one mole of hydrogen is produced for each mole of zinc oxidized. This corresponds to 5.5 SCF hydrogen produced for each pound of zinc corroded.

The time-temperature cycle (Table 6.2.5A-1) considered in the calculation of aluminum and zinc corrosion is based on a conservative step-wise representation of the postulated post accident containment transient. The corrosion rates at the various steps
36 were determined from the aluminum [8] and zinc [9] corrosion rate design curves shown in Figures 6.2.5A-1 and 6.2.5A-2. Aluminum corrosion data points include the effects of temperature, alloy, and spray solution conditions. Based on these corrosion rates and the aluminum and zinc inventory given in Tables 6.2.5A-3 and
36 6.2.5A-4, the contribution of aluminum and zinc corrosion to hydrogen accumulation in the containment following the design basis accident has been calculated. For conservative estimation, no credit was taken for protective shielding effects of insulation or enclosures from the spray, and complete and continuous immersion was assumed.

Calculations based on Regulatory Guide 1.7 are performed by allowing an increased aluminum corrosion rate during the final step of the post accident containment temperature transient (Table 6.2.5A-1) corresponding to 200 mils/year (15.7 $mg/dm^2/hr$). The aluminum corrosion rates earlier in the accident sequence are the higher rates determined from Figure 6.2.5A-1.

4. Radiolysis of Core and Sump Water

Water radiolysis is a complex process involving reactions of numerous intermediates. However, the overall radiolytic process may be described by the reaction:

$$H_2O \rightleftarrows H_2 + 1/2\ O_2$$

Of interest here is the quantitative definition of the rates and extent of radiolytic hydrogen production following the design basis accident.

An extensive program has been conducted by Westinghouse to investigate the radiolytic decomposition of the core cooling solution following the design basis accident. In the course of this investigation, it became apparent that two separate radiolytic environments exist in the containment at design basis accident conditions. In one case, radiolysis of the core cooling solution occurs as a result of the decay energy of fission products in the fuel. In the other case, the decay of dissolved fission products, which have escaped from the core, results in the radiolysis of the sump solution. The results of these investigations are discussed in Reference [5].

6.2.5A.3 Core Solution Radiolysis

As the emergency core cooling solution flows through the core, it is subjected to gamma radiation by decay of fission products in the fuel. This energy deposition results in solution radiolysis and the production of molecular hydrogen and oxygen. The initial production rate of these species will depend on the rate of energy absorption and the specific radiolytic yields.

The energy absorption rate in solution can be assessed from knowledge of the fission products contained in the core, and a detailed analysis of the dissipation of the decay energy between core materials and the solution. The results of Westinghouse studies show essentially all of the beta energy will be absorbed within the fuel and cladding and that this represents approximately 50 percent of the total betagamma decay energy. This study shows further that of the gamma energy, a maximum of 7.4 percent will be absorbed by the solution in core. Thus, an overall absorption factor of 3.7 percent of the total core decay energy ($\beta + \gamma$) is used to compute solution radiation dose rates and the
36 time-integrated dose. Table 6.2.5A-5 presents the total decay energy ($\beta + \gamma$) of a reactor core, which assumes a full power operating time of 650 days prior to the accident. For the maximum credible accident case, the contained decay energy in the core accounts for the assumed TID-14844 release of 50 percent halogens and 1 percent other fission products. To be conservative, the noble gases have been assumed to remain in the core, whereas in reality, the noble gases are assumed by the TID-14844 model to escape to the containment vapor space where little or no water radiolysis would result from decay of these nuclides.

The radiolysis yield of hydrogen in solution has been studied extensively by Westinghouse and ORNL. The results of static capsule tests conducted by Westinghouse indicate that hydrogen yields much lower than the maximum of 0.44 molecules per 100 ev would be the case incore. With

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little gas space to which the hydrogen formed in solution can escape, the rapid back reactions of modecular radiolytic products in solution to reform water is sufficient to result in very low net hydrogen yields.

However, it is recognized that there are differences between the static capsule tests and the dynamic condition in core, where the core cooling fluid is continuously flowing. Such flow is reasoned to disturb the steady state conditions which are observed in static capsule tests, and while the occurrence of back reactions would still be significant, the overall net yield of hydrogen would be somewhat higher in the flowing system.

The study of radiolysis in dynamic systems was initiated by Westinghouse, which formed the basis for experimental work performed at ORNL. Both studies clearly illustrate the reduced yields in hydrogen from core radiolysis, i.e., reduced from the maximum yield of 0.44 molecules per 100 ev. These results have been published[5 and 6].

The calculations of hydrogen yield from core radiolysis are bounded by 36
the very conservative value of 0.44 molecules per 100 ev. That this value is conservative and a maximum for this type of aqueous solution and gamma radiation is confirmed by many published works. The Westinghouse results from the dynamic studies show 0.44 to be a maximum at very high solution flow rates through the gamma radiation field. The referenced ORNL[6] work also confirms this value as a maximum at high flow rates. A. O. Allen[7] presents a very comprehensive review of work performed to confirm the primary hydrogen yield to be a maximum of 0.44 - 0.45 molecules per 100 ev.

Calculations based on Regulatory Guide 1.7 assume a hydrogen yield 36
value of 0.5 molecules per 100 ev, 10 percent of the gamma energy produced from fission products in the fuel rods is absorbed by the solution in the region of the core, and the noblp gases escape to the containment vapor space.

6.2.5A.4 Sump Solution Radiolysis

Another potential source of hydrogen assumed for the post accident period arises from water contained in the reactor containment sump being subjected to radiolytic decomposition by fission products. In this consideration, an assessment must be made as to the decay energy deposited in the solution and the radiolytic hydrogen yield, much in the same manner as given above for core radiolysis.

The energy deposited in solution is computed using the following basis:

1. For the maximum credible accident, a TID-14844 release model[1] is assumed where 50 percent of the total core halogens and 1 percent of all other fission products, excluding noble gases, are released from the core to the sump solution.

2. The quantity of fission product release is equal to that from a reactor operating at full power for 650 days prior to the accident.

3. The total decay energy from the released fission products, both beta and gamma, is assumed to be fully absorbed in the solution.

Within the assessment of energy release by fission products in water, account is made of the decay of halogens, and a separate accounting for the slower decay of the 1 percent other fission products. To arrive at the energy deposition rate and time-integrated energy deposited, the contribution from each individual fission product class was computed. The overall contributions from each of the two classes of fission
36 products is shown in Table 6.2.5A-6.

The yield of hydrogen from sump solution radiolysis is most nearly represented by the static capsule tests performed by Westinghouse and ORNL with the alkaline sodium borate solution. The differences between these tests and the actual conditions for the sump solution, however, are important and render the capsule tests conservative in their predictions of radiolytic hydrogen yields.

In this assessemnt, the sump solution will have considerable depth, which inhibits the ready diffusion of hydrogen from solution, as compared to the case with shallow-depth capsule tests. This retention of hydrogen in solution will have a significant effect in reducing the hydrogen yields to the containment atmosphere. The build-up of hydrogen concentration in solution will enhance the back reaction to formation of water and lower the net hydrogen yield, in the same manner as a reduction in gas to liquid volume ratio will reduce the yield. This is illustrated by the data presented in Figure 6.2.5A.3 for capsule tests with various gas to liquid volume ratios. The data show a significant reduction in the apparent or net hydrogen yield from the published primary maximum yield of 0.44 molecules per 100 ev. Even at the very highest ratios, where capsule solution depths are very low, the yield is less than 0.30, with the highest scatter data point at 0.39 molecules per 100 ev. 36

Calculations based on Regulatory Guide 1.7 do not take credit for a reduced hydrogen yield in the case of sump radiolysis and a hydrogen yield value of 0.5 molecules per 100 ev has been used.

6.2.5A.5 Results

Figure 6.2.5A-5 shows hydrogen production rate as a function of time following a loss of coolant accident. 36

Figure 6.2.5A-7 shows quantity of hydrogen accumulated in the containment as a function of time, with no recombiners operating.

36 Figure 6.2.5A-9 shows volume percent of hydrogen as a function of time, for the maximum credible accident, with a 100 SCFM recombiner started on the second day, and with a recombiner started when the hydrogen concentration reaches 3.5 volume percent.

6.2.5A.6 Conclusion

A single hydrogen recombiner, put in operation on the second day following a LOCA or when the bulk containment hydrogen concentration reaches 3.5 volume percent, will maintain the bulk hydrogen concentration well below the combustible limit of 4 volume percent.

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6.2.6 CONTAINMENT LEAKAGE TESTING

The Containment Building, Containment penetrations, and Containment isolation barriers are designed to permit periodic leakage rate testing as required by GDC 52, 53, and 54 of Title 10, Code of Federal Regulations, Part 50, Appendix A [2] [3] [4], in accordance with the requirements of 10 CFR Part 50, Appendix J [1]. Containment leakage tests are performed periodically to verify that Containment leakage is maintained below the acceptable limits stated in the Technical Specifications.

The leakage testing program consists of the following types of leakage tests:

1. Type A Tests

   Type A tests are those tests, as defined in 10 CFR Part 50, Appendix J, which are performed after the Containment Building has been completed, prior to operation, and at periodic intervals thereafter, to determine the overall Containment integrated leakage rate.

2. Type B Tests

   Type B tests are those tests, as defined by 10 CFR Part 50 Appendix J, which are performed periodically to determine leakage rates for individual mechanical and electrical penetrations, air locks, and hatches.

3. Type C Tests

   Type C tests are those tests, as defined by 10 CFR Part 50, which are performed periodically to measure Containment isolation valve leakage rates.

TABLE 6.2.4-6
(Sheet 6 of 9)
CLASSIFICATION OF SYSTEM PATHS PENETRATING CONTAINMENT WALL

Item	Penetration Number	System	Normal Operating Function	Classification	Post-Accident Function	
64	MIII-24	-	Spare	-	-	
65	MIII-25	-	Spare	-	-	
66	MIII-26	-	Spare	-	-	
67	MIII-27	SF	To Refueling Water Purification Pump	non-essential	none	14
68	MIII-28	-	Spare	-	-	
69	MIII-29	-	Spare	-	-	
70	MIII-30	-	Spare	-	-	
71	MIII-31	SF	Refueling Cavity Skimmer Pump	non-essential	none	
72	MIII-32	-	Spare	-	-	
73	MIV-1(a)	MS	Sample From Steam Generator #1	non-essential	none	
74	MIV-1(b)	PS	R.C. Sample From Hot Legs	non-essential	RC PASS Sample From Hot Legs	36
75	MIV-1(c)	-	Spare	-	-	
76	MIV-2(a)	MS	Sample From Steam Generator #2	non-essential	none	
77	MIV-2(b)	PS	Pressurizer Liquid Space Sample	non-essential	none	
78	MIV-2(c)	PS	Pressurizer Steam Space Sample	non-essential	none	14
79	MIV-3(a)	MS	Sample From Steam Generator #3	non-essential	none	
80	MIV-3(b)	PS	Sample From Accumulators	non-essential	none	
81	MIV-3(c)	WP	RC PASS Sample Return to Reactor Coolant Drain Tank	non-essential	same as normal	36

TABLE 6.2.4-6
(Sheet 7 of 9)
CLASSIFICATION OF SYSTEM PATHS PENETRATING CONTAINMENT WALL

Item	Penetration Number	System	Normal Operating Function	Classification	Post-Accident Function	
82	MIV-4(a)	MS	Sample From Steam Generator #4	non-essential	none	14
83	MIV-4(b)	SI	Accumulator Test & Fill	non-essential	none	
84	MIV-4(c)	PS	None	non-essential	CA PASS Sample Return	36
85	MIV-5(a)	-	Spare	-	-	
86	MIV-5(b)	-	Spare	-	-	14
87	MIV-5(c)	-	Spare	-	-	
88	MIV-6(a)	-	Spare	-	-	36
89	MIV-6(b)	-	Spare	-	-	14
90	MIV-6(c)	-	Spare	-	-	
91	MIV-7(a)	-	Spare	-	-	36
92	MIV-7(b)	ESFAS	Containment Pressure Sensing PT-934	essential	Containment pressure measurement	14
93	MIV-7(c)	-	Spare	-	-	
94	MIV-8(a)	RM	Radiation Monitoring Sample	non-essential	Radiation Monitoring Sampling	
95	MIV-8(b)	ESFAS	Containment Pressure Sensing PT-935	essential	Containment pressure measurement	
96	MIV-8(c)	-	Spare	-	-	
97	MIV-9(a)	PS	None	non-essential	CA PASS Sample Intake	36
98	MIV-9(b)	ESFAS	Containment Pressure Sensing PT-936	essential	Containment Pressure measurement	14
99	MIV-9(c)	-	Spare	-	-	
100	MIV-10(a)	PS	None	non-essential	CA PASS Sample Intake	36

TABLE 6.2.5A-1

POST ACCIDENT CONTAINMENT TEMPERATURE TRANSIENT USED IN THE CALCULATION OF ALUMINUM CORROSION

Time Interval (Seconds)	Temperature
0 - 150	265
150 - 1000	242
1000 - 4000	224
4000 - 10,000	214
10,000 - 30,000	190
30,000 - 86,400	172
86,400 - 1.7×10^5	153*

*For temperature of 153°F and less, the long term aluminum corrosion rate of 200 mils/year specified in Regulatory Guide 1.7 is used. 36

TABLE 6.2.5A-2 (Sh. 1 of 3)

PARAMETERS USED TO DETERMINE HYDROGEN GENERATION

Thermal Power Rating	3565 MWt
Containment Free Volume	2.9×10^6 ft^3
Containment Temperature at Accident	120°F
Weight Zirconium Cladding	43,304 lb
Hydrogen Generated Zirconium-Water Reaction	
Based on 1.5 percent value	5140 SCF
Hydrogen in Primary Coolant	1191 SCF
Hydrogen Recombiner Capacity	100 SCFM

HYDROGEN PRODUCTION CALCULATIONAL ASSUMPTIONS
OF REGULATORY GUIDE 1.7

CORE COOLING SOLUTION RADIOLYSIS

Sources

- Percent of total halogens retained in the core	50
- Percent of total noble gases retained in the core	0
- Percent of other fission products retained in the core	99

36

TABLE 6.2.5A-2 (Sh. 2 of 3)

HYDROGEN PRODUCTION CALCULATIONAL ASSUMPTIONS
OF REGULATORY GUIDE 1.7

Energy Distribution	
- Percent of total decay energy - gamma	50
- Percent of total decay energy - beta	50
Energy Absorption by Core Cooling Solution	
- Percent of gamma energy absorbed by solution	10
- Percent of beta energy absorbed by solution	0
Hydrogen Production	
- Molecules H_2 produced per 100 ev energy absorbed by solution	0.50
SUMP SOLUTION RADIOLYSIS	
Sources	
- Percent of total halogens released to sump solution	50
- Percent of noble gases released to sump solution	0
- Percent of other fission products released to sump solution	1

36

TABLE 6.2.5A-2 (Sh. 3 of 3)

HYDROGEN PRODUCTION CALCULATIONAL ASSUMPTIONS
OF REGULATORY GUIDE 1.7

Energy Absorption by Sump Solution	
- Percent of total energy (beta and gamma) which is absorbed by the sump solution	100
Hydrogen Production	
- Molecules of hydrogen produced per 100 ev of energy absorbed by the sump solution	0.5
Long Term Aluminum Corrosion Rate	
- Mills per year	200
- Milligrams per square decimeter per hour	16

36

TABLE 6.2.5A-3

INVENTORY OF ALUMINUM IN CONTAINMENT

No.	Item	Material	Source	Wt. (lbs.)	S.A. (ft^2)
1.	Flux Mapping Drive System	Al	-	205	88
2.	Nuclear Instrumentation System	Al	-	280	95
3.	Rod Position Indicators	Al	-	177	93
4.	Misc. Valves	Al	-	230	86
5.	Control Rod Drive Mechanism Connectors	Al	-	193	42
6.	Contingency	Al	-	250	85
7.	Refueling Machine	Al	-	28	5
8.	Miscellaneous Mechanical Equipment	Al	Parts	240	38
9.	Miscellaneous Electrical Equipment	Al	Parts	7	14
10.	I and C Valves and Accessories	Al	Parts	55	20

TABLE 6.2.5A-4

INVENTORY OF ZINC IN CONTAINMENT

No.	Item	Material	Source	Wt. (lbs.)	S.A. (ft^2)
1.	Miscellaneous Mechanical Equipment	Zn	Paint	5135	117,693
2.	Miscellaneous Mechanical Equipment	Zn	Galvan.	98	1,254
3.	Miscellaneous Mechanical Equipment	Zn	Parts	13	1
4.	HVAC Equipment and Ducts	Zn	Paint	539	12,272
5.	HVAC Equipment and Ducts	Zn	Galvan.	5113	107,477
6.	Miscellaneous Electrical Equipment	Zn	Galvan.	4126	56,010
7.	Electrical Cable Trays	Zn	Galvan.	2300	26,000
8.	I and C Valves and Accessories	Zn	Paint	98	2,250
9.	I and C Valves and Accessories	Zn	Galvan.	870	8,000
10.	I and C Valves and Accessories	Zn	Parts	86	1,242
11.	Structural Liner	Zn	Paint	4830	111,330
12.	Structural Miscellaneous Steel	Zn	Paint	2875	66,284
13.	Structural Miscellaneous Steel	Zn	Galvan.	2980	23,838
	TOTAL ZINC			29,063	533,651

TABLE 6.2.5A-5 | 36

CORE FISSION PRODUCT ENERGY AFTER 650 FULL POWER DAYS

Time After Reactor Trip Days	Core Fission Product Energy* Energy Release Rate Watts/MWt x 10^{-3}	Integrated Energy Release Watt Days/MWt x 10^{-4}
1	3.887	0.574
5	2.595	1.777
10	2.211	2.967
20	1.760	4.934
30	1.475	6.541
40	1.291	7.919
50	1.163	9.143
60	1.068	10.259
70	0.992	11.289
80	0.926	12.249
90	0.867	13.139
100	0.814	13.979

*Assumes release of 50 percent core halogens + 1 percent other fission products, includes 100 percent noble gases. Values are for total (β and γ) energy.

TABLE 6.2.5A-6 36

FISSION PRODUCT DECAY DEPOSITION IN SUMP SOLUTION

	50 PERCENT HALOGENS		1 PERCENT OTHER FISSION PRODUCTS		TOTAL	
Time After Reactor Trip Days	Energy Release Rate Watt/MWt	Integrated Energy Release Watt-Day/MWt x 10^{-2}	Energy Release Rate Watts/MWt x 10^{-1}	Integrated Energy Release Watt-Day/MWt x 10^{-2}	Energy Release Rate Watts/MWt x 10^{-1}	Integrated Energy Release Watt-Day/MWt x 10^{-3}
1	145	4.27	3.78	0.536	18.28	0.481
3	49.4	5.88	2.90	1.18	7.85	0.707
5	31.0	6.65	2.59	1.73	5.69	0.838
10	18.2	7.82	2.22	2.92	4.03	1.07
20	7.63	9.03	1.77	4.89	2.53	1.39
30	3.22	9.54	1.49	6.51	1.81	1.61
40	1.36	9.76	1.30	7.90	1.44	1.77
60	0.241	9.89	1.08	10.3	1.10	2.02
80	0.043	9.91	0.935	12.3	0.940	2.22
100	0.008	9.92	0.822	14.0	0.823	2.39

FIGURE 6.2.5-3

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FIGURE 6.2.5A-4

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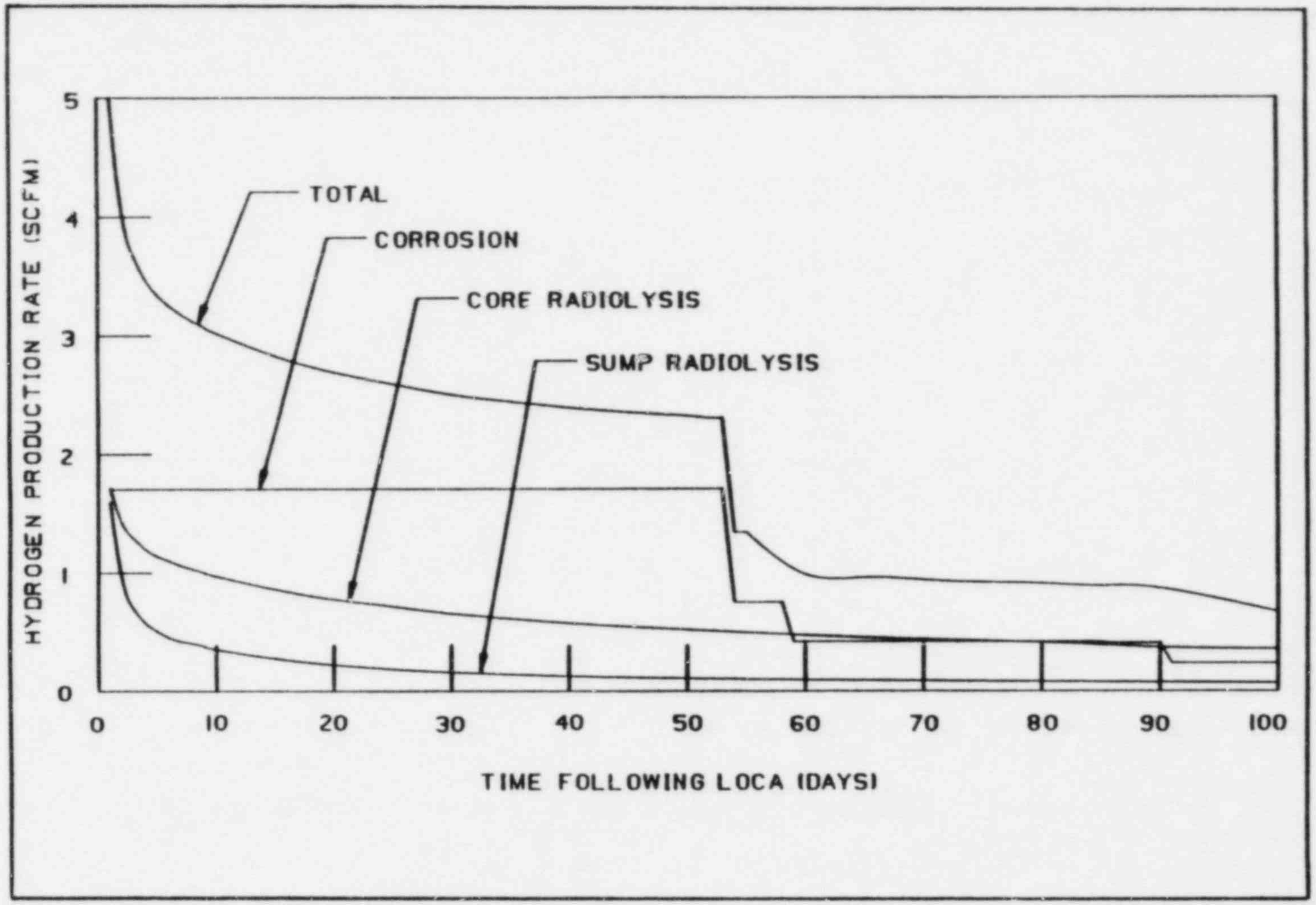

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COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

HYDROGEN PRODUCTION RATE
NRC MODEL

FIGURE 6.2.5A-5

FIGURE 6.2.5A-6

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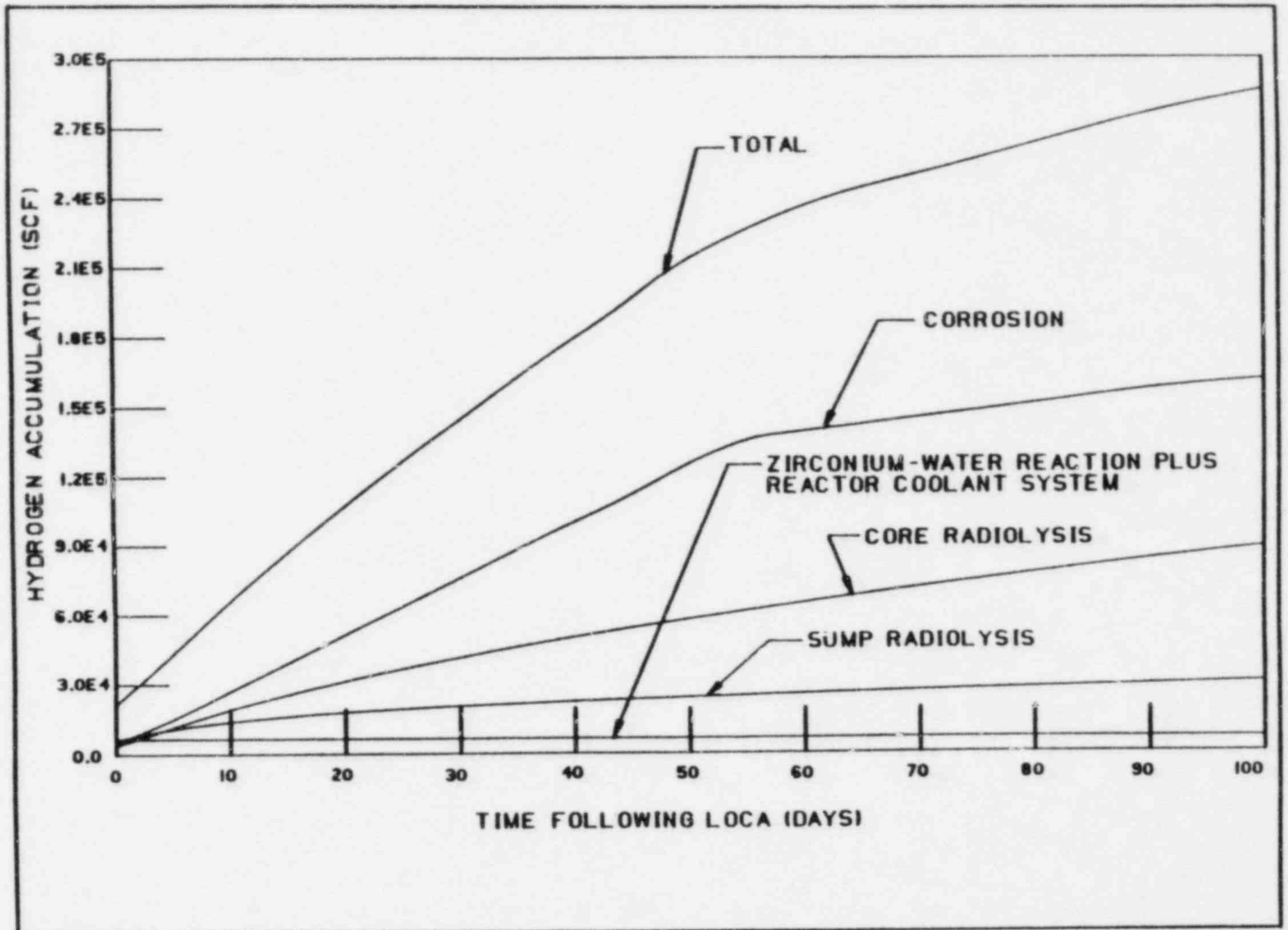

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COMANCHE PEAK S.E.S. FINAL SAFETY ANALYSIS REPORT UNITS 1 and 2
HYDROGEN ACCUMULATION FROM ALL SOURCES, NRC MODEL-NO RECOMBINER
FIGURE 6.2.5A-7

FIGURE 6.2.5A-8

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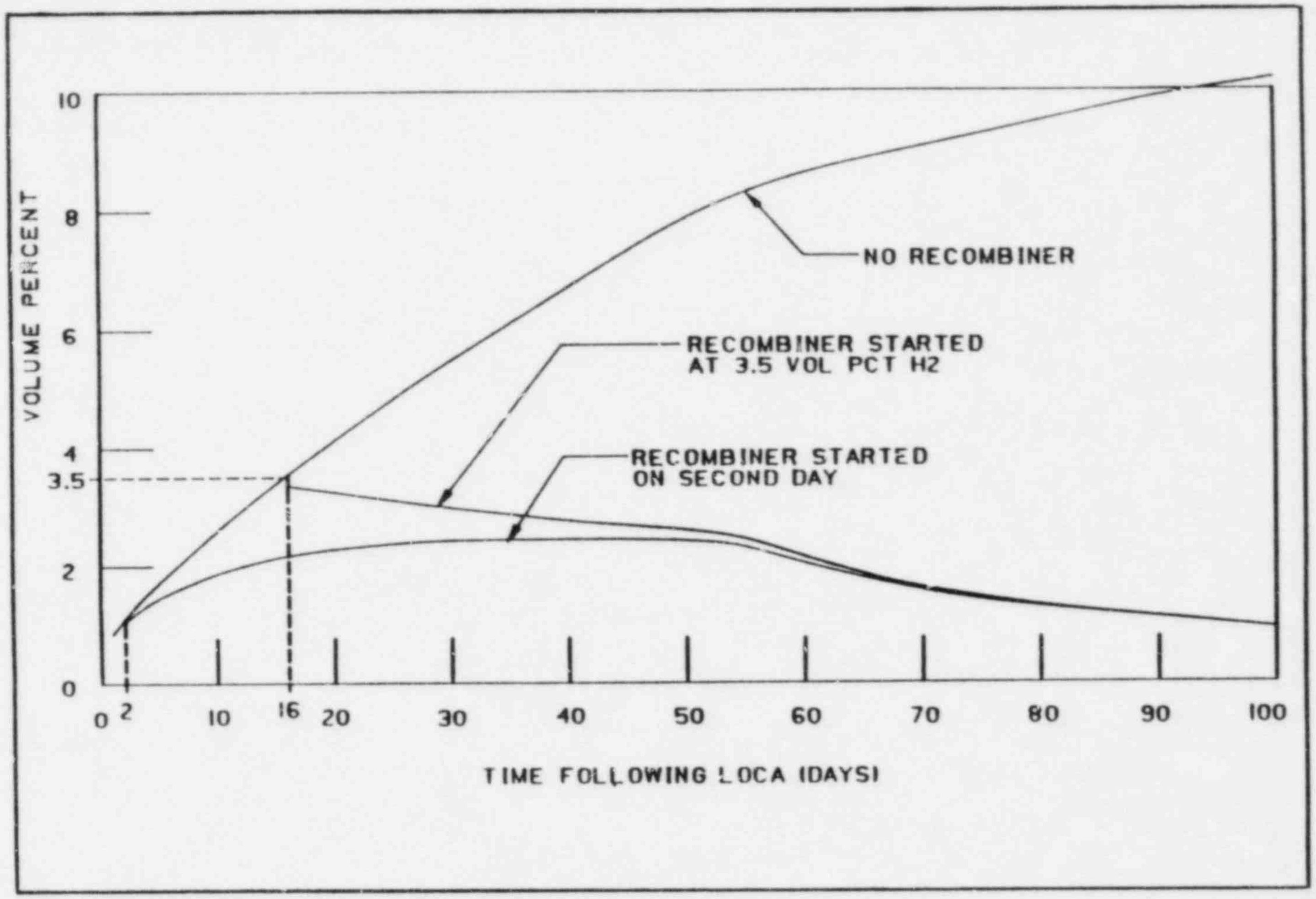

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UNITS 1 and 2

VOLUME PERCENT OF HYDROGEN IN CONTAINMENT
NRC MODEL

FIGURE 6.2.5A-9

alarms, Control Room to outside environs differential pressure indicator, and a Control Room area radiation monitor are provided in the Control Room to enable the operator to evaluate the system performance.

n. Sequencing

The sequencing of the Control Room Air-Conditioning System upon loss of offsite power concurrent with a LOCA is shown in Tables 8.3-1A and 8.3-1B.

5. Auxiliary Feedwater System

The Auxiliary Feedwater System is outlined in Subsection 10.4.9. The system flow diagram is shown in Figure 10.4-11.

a. Initiating circuits 11 Q032.17

The motor-driven pumps are started by any of the following signals:

1) Safety Injection sequence
2) Blackout sequence
3) Trip of both main feedwater pumps (pumps 1-A and 1-B)
4) Steam generator low-low level (two of four from any steam generator)
5) Manual start (from Control Room or Hot Shutdown Panel)

The turbine-driven pump is started by any of the following signals:

1) Steam generator low-low level (two of four from any two steam generators)

2) Blackout (loss of voltage) signal

3) Manual start (from Control Room or Hot Shutdown Panel)

The same signal that starts any of the three auxiliary feedwater pumps closes the blowdown and sample line isolation valves for all the steam generators.

b. Logic

11 See instrumentation and control diagrams listed under "Auxiliary Feedwater" in Tables 1.7-1 and 1.7-2.

Q032.17

c. Bypass

Bypass is indicated on Safety System Inoperable Indicator (SSII) described in Section 7.1.2.6 and generally illustrated in Figure 7.1-4.

d. Interlocks

1) Low pump suction pressure will stop either the motor-driven or the turbine-driven pumps.

2) The operator lockout (OL) or automatic lockout (AL) feature of the sequencer can, on the appearance of a safety injection or blackout signal, annul either manual or autostart of the AFW System motor-driven pumps in order to properly sequence them on the diesel. For local override control at the Hot
36 Shutdown Panel refer to Section 7.4.1.3.1.

3) Starting the Auxiliary Feedwater System also closes the steam generator blowdown isolation and sample
11 line isolation valves.

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c. Bypass

Bypass is indicated on Safety System Inoperable Indicator described in Section 7.1.2.6 and typically illustrated in Figure 7.1-4. 11 Q032.17

d. Interlocks

The Component Cooling Water pumps cannot be started either manually or automatically when the Safety Injection Sequencing or Blackout Sequencing is in progress. For local override control at the Hot Shutdown Panel, refer to Section 7.4.1.3.1. 36

e. Redundancy

Separate switches and actuation circuitry are provided for redundant components which are physically and electrically separated from one another.

f. Diversity

The Component Cooling System is started by such diverse signals as: 11 Q032.17

1) Safety Injection Sequence
2) Blackout Sequence
3) Service water pump autostart on low pressure
4) Low pressure in the discharge header of redundant Component Cooling Water pump.
5) Manual command at either the Control Room or Hot Shutdown Panel.

g. Actuated devices

13 Actuated devices are listed in Table 7.3-4, (for all
Q032.84 equipment actuated by "s" signal) and Table 7.3-5 (for all equipment actuated by "p" signal).

h. Supporting systems

The following support systems are required by the Component Cooling Water System:

1) Station Service Water System (see Section 9.2.1)

2) Class 1E electric power (See Section 8.3)

11

3) Engineered Safety Features Ventilation System (See Section 9.4.5 and Figure 9.4-2, Sheet 1).

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i. Design basis

See Section 9.2.2.2.2 for the design basis of the Component Cooling Water System.

j. Electric schematic drawings

See Table 1.7-1 for Unit 1 and Table 1.7-2 for Unit 2 schematics (electrical) associated with "Component Cooling Water."

k. Portion of system not required for safety

13 Except for those instruments required to actuate the
Q032.84 Component Cooling Water System components listed in Table 7.3-5 or the Safety Related Display instrumentation

described in Section 7.5 and Table 7.5-1, the instruments and monitoring equipment are not required for safety. Local indicators, the plant annunciator and the plant computer are also not required for safety. 13 Q032.84

1. Control of system operation 11

The CCW System is designed as an on/off system during normal operation. Pump recirculation valves are provided to maintain required minimum pump flow under abnormal conditions. Q032.17

The CCW pumps are automatically started by low pressure in the pump discharge header.

The CCW surge tank level is used to detect system leakage. The control system for each half of each partitioned tank consists of the following:

1) Level Indication - Local and Control Room
2) Level Recording
3) Hi-Hi/Lo Alarms
4) Automatic Make-up from reactor make-up water on Lo-Lo level
5) Automatic Make-up "Initiated" alarm
6) Automatic Make-up termination on Hi-Level
7) Manual Make-up form demineralized water. 11 Q032.17

m. Monitoring of system operation

The CCW System is provided with monitoring instrumentation as described in Table 7.5-2. In addition, each component is supplied with local temperature indication. Sufficient local flow indication is provided to allow flow balancing of the system. Each flow indicator has a low flow alarm contact to alert the operator of a system malfunction. Flow indication is provided in the Control Room for Containment Spray and RHR heat exchangers.

Each power operated valve is supplied with a control switch and position indicating lights in the Control Room.

Radiation monitors are provided with readout in the Control Room for each of the three loops in each subsystem.

n. Sequencing

The sequencing of CCW pumps upon loss of offsite power concurrent with a LOCA is shown in Tables 8.3-1A and 8.3-1B.

8. Station Service Water

The Station Service Water (SSW) System is discussed in Section 9.2.1. The System Flow diagram is shown in Figure 9.2-1.

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a. Initiating circuits

The Station Service Water System is started by any of the following signals:

1) Safety Injection Sequence
2) Blackout Sequence
3) Component Cooling Water pump start signal
4) Low pressure signal of Train B supply header starts Train A service water pump and vice versa.
5) Manual command either at Control Board or at Hot Shutdown Panel.

b. Logic

See instrumentation and control diagrams listed under "Station Service Water System" in Tables 1.7-1 and 1.7-2.

c. Bypass

Station Service Water System Bypass is indicated on the Safety System Inoperable Indicator (SSII) described in Section 7.1.2.6 and typically illustrated in Figure 7.1-4. 11 Q032.17

d. Interlocks

The Station Service Water pumps cannot be started either manually or automatically when the Safety Injection Sequencing or Blackout Sequencing is in progress. For local override control at the Hot Shutdown Panel, refer to Section 7.4.1.3.1. 36

e. Redundancy

Separate switches and actuation circuitry are provided for redundant components which are physically and electrically separated from one another.

f. Diversity 11 Q032.17

The Service Water System may be controlled from the following locations:

a) Each Service Water pump may be controlled by:

1) A control switch on the Main Control Board
2) A local control switch on the Hot Shutdown Panel
3) Blackout Sequence
4) Safety Injection Sequence
5) Component Cooling Water pump start signal

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In addition to above, low pressure signal of Train B supply header starts Train A Service Water pump and vice versa.

g. Actuated devices

For devices actuated directly by the "S" signal see Table 7.3-4.

For devices actuated directly by the Station Service Water System, refer to Station Service Water instrumentation and control drawings listed in Tables 1.7-1 and 1.7-2.

h. Support Systems

11

The Service Water System is supported by:

Q037.17

1) Class 1E electric power (See Section 8.3)

2) Service Water Intake Structure Ventilation Exhaust System (See Section 9.4B and Item 6 below)

i. Design basis

For information concerning basis of design see Section 9.2.1.1.

j. Electrical schematic drawings

See Table 1.7-1 for Unit 1 and 1.7-2 for Unit 2 schematics (electrical) associated "Station Service Water."

k. Portion of system not required for safety

Except for those instruments required for actuating Service Water System components listed in Table 7.3-4, or the

b. Logic

See instrumentation and control diagrams listed under "Ventilation Safety Chilled Water" in Tables 1.7-1 and 1.7-2. 11

c. Bypass Q032.17

Bypass is indicated on Safety System Inoperable Indicator described in Section 7.1.2.6 and typically illustrated in Figure 7.1-4.

d. Interlocks

Chilled water recirculation pumps cannot be started manually while Safety Injection Sequence or Blackout Sequence is in progress. For local override control at the Hot Shutdown Panel, refer to Section 7.4.1.3.1. 36

e. Redundancy

Separate switches and actuation circuitry are provided for redundant components which are physically and electrically separated from one another. 11

f. Diversity Q032.17

The Safety Chilled Water Recirculation pumps are started by such diverse signals as:

1) Safety Injection Sequence
2) Blackout Sequence
3) Manual command at the Control Room.

g. Actuated devices

1) Safety Chilled Water recirculation pumps
2) Water Chillers

h. Supporting systems

The following supported systems are required by the Safety Chilled Water System:

1) Class 1E electric power (See Section 8.3)
2) Component Cooling Water (see Section 9.2.2).

i. Design basis

See Section 9.4F.1 for the design basis of the Safety Chilled Water System.

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Q032.17

j. Electric schematic drawings

See Table 1.7-1 for Unit 1 and Table 1.7-2 for Unit 2 schematic (electrical) associated with "Ventilation Chilled Water and Safety Chilled Water."

k. Portion of system not required for safety

The following portions of the Safety Chilled Water System are not required for safety:

1) Remote and local indicator
2) Plant annunciators

The manual reset feature associated with containment spray, therefore, does not perform a bypass function. It is merely the first of several manual operations required to take control from the automatic system or interrupt its completion should such an action be considered necessary.

In event that the operator anticipates system actuation and erroneously concludes that it is undesirable or unnecessary and imposes a standing reset condition in one train (by operating and holding the corresponding reset switch at the time the initiate signal is transmitted) the other train will automatically carry the protective action to completion. In the event that the reset condition is imposed simultaneously in both trains at the time the initiate signals are generated, the automatic sequential completion of system action is interrupted and control has been taken by the operator. Manual takeover will be maintained, even though the reset switches are released, if the original initiate signal exists. Should the initiate signal then clear and return again, automatic system actuation will repeat.

Note also that any time delays imposed on the system action are to be applied after the initiating signals are latched. Delay of actuate signals for fluid systems line-up, load sequencing, etc., do not provide the operator time to interrupt automatic completion, with manual reset alone, as would be the case if time delay was imposed prior to sealing of the initial actuate signal.

The manual block features associated with pressurizer and steam line safety injection signals provide the operator with the means to block initiation of safety injection during plant startup. These block features meet the requirements of Section 4.12 of IEEE Standard 279-1971 in that automatic removal of the block occurs when plant conditions require the protection system to be functional.

7.3.2.2.7 Manual Initiation of Protective Actions (Regulatory Guide 1.62)

There are four individual main steam stop valve momentary control switches (one per loop) mounted on the control board. Each switch when actuated will isolate one of the main steam lines. In addition, there will be two system level switches. Each switch will actuate all four main steam line isolation and bypass valves of the system level. Manual initiation of switchover to recirculation is in compliance with Section 4.17 of IEEE Standard 279-1971 with the following comment.

Manual initiation of either one of two redundant safety injection actuation main control board mounted switches provides for actuation of the components required for reactor protection and mitigation of adverse consequences of the postulated accident, including delayed actuation of sequenced started emergency electrical loads as well as components providing switchover from the safety injection mode to the cold leg recirculation mode following a loss of primary coolant accident. Therefore, once safety injection is initiated, those components of the Emergency Core Cooling System (see Section 6.3) which are automatically realigned as part of the semiautomatic switchover go
36 to completion on low-low refueling water storage tank (RWST) water level without any manual action. Manual operation of other components or manual verification of proper position as part of emergency procedures is not precluded nor otherwise in conflict with the above described compliance to Section 4.17 of IEEE Standard 279-1971 of the semiautomatic switchover circuits.

No exception to the requirements of IEEE Standard 279-1971 has been taken in the manual initiation circuit of safety injection. Although Section 4.17 of IEEE Standard 279-1971 requires that a single failure within common portions of the protective system shall not defeat the protective action by manual or automatic means, the standard does not specifically preclude the sharing of initiated circuitry logic between automatic and manual functions. It is true that the manual safety

SAFETY INJECTION ACTUATED EQUIPMENT LIST (NSSS)

Unit Equipment(*) Identification	Description	ESFAS Train	SIS Signal Function	Drawing Number	
X-TBX-RHAPRH-01	Residual Heat Removal Pump No. 1	A	Start	2323-E1-0031/49,50	
X-TBX-RHAPRH-02	Residual Heat Removal Pump No. 2	B	Start	2323-E1-0031/51,52	
X-TBX-SIAPSI-01	Safety Injection Pump No. 1	A	Start	2323-E1-0031/45,46	
X-TBX-SIAPSI-02	Safety Injection Pump No. 2	B	Start	2323-E1-0031/47,48	
X-TBX-CSAPCH-01	Charging Pump No. 1	A	Start	2323-E1-0031/53,54	
X-TBX-CSAPCH-02	Charging Pump No. 2	B	Start	2323-E1-0031/55,56	
1-LCV-112B	Volume Control Tank Isolation Valve	A	Close	2323-E1-0061/14	
1-LCV-112C	Volume Control Tank Isolation Valve	B	Close	2323-E1-0061/15	
1-8220	Volume Control Tank Isolation Valve	A	Close	2323-E1-0061/84	36
1-8221	Volume Control Tank Isolation Valve	B	Close	2323-E1-0061/91	
1-LCV-112D	SIS Refueling Water to Charging Pump Header, Isolation Valve	A	Open	2323-E1-0061/16	
1-LCV-112E	SIS Refueling Water to Charging Pump Header, Isolation Valve	B	Open	2323-E1-0061/17	
1-8105	Charging Pump Discharge to RCS Isolation Valve	B	Close	2323-E1-0061/02	
1-8106	Charging Pump Discharge to RCS Isolation Valve	A	Close	2323-E1-0061/03	
1-8110	Charging Pump Miniflow Isolation Valve	A	Close	2323-E1-0061/04	
1-8111	Charging Pump Miniflow Isolation Valve	B	Close	2323-E1-0061/05	
1-8800A	RWST Isolation Valves to SFPCS Pump	A	Close	2323-E1-0062/01	
1-8800B	RWST Isolation Valves to SFPCS Pump	B	Close	2323-E1-0062/02	
1-8801A	High Head Safety Injection Isolation Valve	A	Open	2323-E1-0062/05	16
1-8801B	High Head Safety Injection Isolation Valve	B	Open	2323-E1-0062/06	
1-8808A	SIS Accumulator #1 Isolation Valve	A	Open	2323-E1-0062/16	

SAFETY INJECTION ACTUATED EQUIPMENT LIST (NSSS)

Unit Equipment (*) Identification	Description	ESFAS Train	SIS Signal Function	Drawing Number
1-8808B	SIS Accumulator #2 Isolation Valve	B	Open	2323-E1-0062/17
1-8808C	SIS Accumulator #3 Isolation Valve	A	Open	2323-E1-0062/18
1-8808D	SIS Accumulator #4 Isolation Valve	B	Open	2323-E1-0062/19
1-8811A	Sump Control Valves to RHR Pump-01	A	Open	2323-E1-0062/22
1-8811B	Sump Control Valves to RHR Pump-02	B	Open	2323-E1-0062/23
X-TBX-CSAPPD-01	CVCS Positive Displacement Charging Pump	A	Stop	2323-E1-0033-Sh. 13, Sh. 14
1-8202A	Suction Stabilizer Vent Isolation Valve	A	Close	2323-E1-0061/81
1-8202B	Suction Stabilizer Vent Isolation Valve	B	Close	2323-E1-0061/82
1-8210A	Nitrogen Supply to Suction Stabilizer Tank	A	Close	2323-E1-0061/11
1-8210B	Nitrogen Supply to Suction Stabilizer Tank	B	Close	2323-E1-0061/79
1-TCX-RCPCPR-01	**Pressurizer Heater Backup Group A	AA	Trip	2323-E1-033-Sh. 35
1-TCX-RCPCPR-02	**Pressurizer Heater Backup Group B	BB	Trip	2323-E1-033-Sh. 37
1-TCX-RCPCPR-03	**Pressurizer Heater Backup Group C	AA	Trip	2323-E1-033-Sh. 33
1-TCX-RCPCPR-04	**Pressurizer Heater Backup Group D	BB	Trip	2323-E1-033-Sh. 39

SAFETY INJECTION ACTUATED EQUIPMENT LIST (BOP)

Unit Equipment(*) Identification	Description	ESFAS Train	SIS Signal Function	Drawing Number	
1-A	Feedwater Pump Turbine	A/B	Trip	2323-M1-2203-02	36
1-B	Feedwater Pump Turbine	A/B	Trip	2323-M1-2203-02	
1-HV-2134	Stm. Gen. #1 F.W. Isolation Valve	A/B	Close	2323-M1-2203-04	
1-HV-2135	Stm. Gen. #2 F.W. Isolation Valve	A/B	Close	2323-M1-2203-04	
1-HV-2136	Stm. Gen. #3 F.W. Isolation Valve	A/B	Close	2323-M1-2203-04	
1-HV-2137	Stm. Gen. #4 F.W. Isolation Valve	A/B	Close	2323-M1-2203-04	
1-CP1-AFAPMD-01	Motor Driven Aux. F.W. Pump	A	Start	2323-M1-2206-01	
1-CP1-AFAPMD-02	Motor Driven Aux. F.W. Pump	B	Start	2323-M1-2206-01	
1-HV-2484	Condensate Makeup and Reject	A	Close	2323-M1-2206-01	
1-HV-2397	SG #1 Blowdown Isolation Valve	A/B	Close	2323-M1-2206-01	
1-HV-2398	SG #2 Blowdown Isolation Valve	A/B	Close	2323-M1-2206-01	
1-HV-2399	SG #3 Blowdown Isolation Valve	A/B	Close	2323-M1-2206-01	
1-HV-2400	SG #4 Blowdown Isolation Valve	A/B	Close	2323-M1-2206-01	
1-HV-2401A	SG #1 Drum Sample Isolation Valve	A	Close	2323-M1-2206-01	
1-HV-2401B	SG #1 BLDN Sample Isolation Valve	A	Close	2323-M1-2206-01	
1-HV-2402A	SG #2 Drum Sample Isolation Valve	A	Close	2323-M1-2206-01	
1-HV-2402B	SG #2 BLDN Sample Isolation Valve	A	Close	2323-M1-2206-01	
1-HV-2403A	SG #3 Drum Sample Isolation Valve	A	Close	2323-M1-2206-01	
1-HV-2403B	SG #3 BLDN Sample Isolation Valve	A	Close	2323-M1-2206-01	
1-HV-2404A	SG #4 Drum Sample Isolation Valve	A	Close	2323-M1-2206-01	
1-HV-2404B	SG #4 BLDN Sample Isolation Valve	A	Close	2323-M1-2206-01	
1-HV-2485	Condensate Make-up and Reject	B	Close	2323-M1-2232-03	

SAFETY INJECTION ACTUATED EQUIPMENT LIST (BOP)

Unit Equipment (*) Identification	Description	ESFAS Train	SIS Signal Function	Drawing Number
1-FR-4772	Containment Spray Pumps 01/03 Disch. Flow Rec.	A	Chart Motor Start	2323-M1-2232-03
1-FR-4773	Containment Spray Pumps 02/04 Disch. Flow Rec.	A	Chart Motor Start	2323-M1-2232-03
X-HV-5825C	Control Rm. A/C Unit and Emergency Filter Inlet Damper	B	Open	2323-M1-2304-01A
X-HV-5828C	Control Rm. A/C Unit and Emergency Filter Inlet Damper	A	Open	2323-M1-2304-
1-HV-2405	SG #1 Sample Isolation Valve	B	Close	2323-M1-2202-06
1-HV-2406	SG #2 Sample Isolation Valve	B	Close	2323-M1-2202-06
1-HV-2407	SG #3 Sample Isolation Valve	B	Close	2323-M1-2202-06
1-HV-2408	SG #4 Sample Isolation Valve	B	Close	2323-M1-2202-06
1-CP1-CTAPCS-01	Containment Spray Pump	A	Start	2323-M1-2232-03
1-CP1-CTAPCS-03	Containment Spray Pump	A	Start	2323-M1-2232-03
1-CP1-CTAPCS-02	Containment Spray Pump	B	Start	2323-M1-2232-03
1-CP1-CTAPCS-04	Containment Spray Pump	B	Start	2323-M1-2232-03
1-HV-5640A	Motor Driven Aux. FWP Room Supply Damper	B	Close	2323-M1-2302-05
1-HV-5640B	Motor Driven Aux. FWP Room Exhaust Valve	B	Close	2323-M1-2302-05
1-HV-5641A	Motor Driven Aux. FWP Room Supply Damper	A	Close	2323-M1-2302-05
1-HV-5641B	Motor Driven Aux. FWP Room Exhaust Valve	A	Close	2323-M1-2302-05
1-HV-5658A	RHR Pump Room Supply Damper	A	Close	2323-M1-2302-05

SAFETY INJECTION ACTUATED EQUIPMENT LIST (BOP)

Unit Equipment(*) Identification	Description	ESFAS Train	SIS Signal Function	Drawing Number
1-HV-5658B	RHR Pump Room Exhaust Valve	A	Close	2323-M1-2302-05
1-HV-5659A	Containment Spray Pump Room Supply Damper	A	Close	2323-M1-2302-05
1-HV-5659B	Containment Spray Pump Room Exhaust Valve	A	Close	2323-M1-2302-05
1-HV-5660A	Safety Injection Pump Room Supply Damper	A	Close	2323-M1-2302-05
1-HV-5660B	Safety Injection Pump Room Exhaust Valve	A	Close	2323-M1-2302-05
1-HV-5663A	Safety Injection Pump Room Supply Damper	B	Close	2323-M1-2302-05
1-HV-5663B	Safety Injection Pump Room Exhaust Valve	B	Close	2323-M1-2302-05
1-HV-5664A	Containment Spray Room Supply Damper	B	Close	2323-M1-2302-05
1-HV-5664B	Containment Spray Room Exhaust Valve	B	Close	2323-M1-2302-05
1-HV-5665A	RHR Pump Room Supply Damper	B	Close	2323-M1-2302-05
1-HV-5665B	RHR Pump Room Exhaust Valve	B	Close	2323-M1-2302-05
X-CPX-VAFNAV-37	Control Room Air Makeup Fan	A	Start	2323-M1-2304-01
X-CPX-VAFNAV-38	Control Room Air Makeup Fan	B	Start	2323-M1-2304-01
X-CPX-VAFNCB-05	Control Room Pressurization Fan	A	Start	2323-M1-2304-02
X-CPX-VAFNCB-06	Control Room Pressurization Fan	B	Start	2323-M1-2304-02
X-CPX-VAFNCB-23	Emergency Filtration Fan	A	Start	2323-M1-2304-04
X-CPX-VAFNCB-24	Emergency Filtration Fan	B	Start	2323-M1-2304-04
1-CP1-VAFNID-01	Control Room Main Exhaust Fan	A	Start	2323-M1-2304-06

SAFETY INJECTION ACTUATED EQUIPMENT LIST (BOP)

Unit Equipment(*) Identification	Description	ESFAS Train	SIS Signal Function	Drawing Number
X-CP1-VAFNID-02	Control Room Main Exhaust Fan	B	Start	2323-M1-2304-06
X-CP1-VAFNID-03	Control Room Complex Kitchen and Toilet Exhaust Fan	A	Start	2323-M1-2304-07
X-CP1-VAFNID-04	Control Room Complex Kitchen and Toilet Exhaust Fan	B	Start	2323-M1-2304-07
X-HV-5859	Control Room Kitchen/Toilet Storage Air Recirc Damper	A	Close	2323-M1-2304-08
X-HV-5883	Control Room Vent Control Damper	A	Close	2323-M1-2304-13
X-HV-5747A	Charging Pump Room Supply Damper	B	Close	2323-M1-2303-05A
X-HV-5747B	Charging Pump Room Exhaust Valve	B	Close	2323-M1-2303-05A
X-HV-5758A	Comp. Cooling Water Pump Room Supply Damper	A	Close	2323-M1-2303-05A
X-HV-5758B	Comp. Cooling Water Pump Room Exhaust Valve	A	Close	2323-M1-2303-05A
X-HV-5762A	Charging Pump Room Supply Damper	A	Close	2323-M1-2303-05A
X-HV-5762B	Charging Pump Room Exhaust Valve	A	Close	2323-M1-2303-05A
X-HV-5764A	Comp. Cooling Water Pump Room Supply Damper	A	Close	2323-M1-2303-05A
X-HV-5764B	Comp. Cooling Water Pump Room Exhaust Valve	A	Close	2323-M1-2303-05A
X-HV-5765A	Comp. Cooling Water Pump Room Supply Damper	B	Close	2323-M1-2303-05A
X-HV-5765B	Comp. Cooling Water Pump Room Exhaust Valve	B	Close	2323-M1-2303-05A

SAFETY INJECTION ACTUATED EQUIPMENT LIST (BOP)

Unit Equipment(*) Identification	Description	ESFAS Train	SIS Signal Function	Drawing Number	
X-HV-5766A	Comp. Cooling Water Pump Room Supply Damper	B	Close	2323-M1-2303-05A	
X-HV-5766B	Comp. Cooling Water Pump Room Exhaust Valve	B	Close	2323-M1-2303-05A	
X-HV-5774A	Charging Pump Room Supply Damper	A	Close	2323-M1-2303-05A	
X-HV-5774B	Charging Pump Room Exhaust Valve	A	Close	2323-M1-2303-05A	
X-HV-5779A	Charging Pump Room Supply Damper	B	Close	2323-M1-2303-05A	
X-HV-5779B	Charging Pump Room Exhaust Valve	B	Close	2323-M1-2303-05A	
X-HV-5785A	Spent Fuel Pool HX and Pump Room Supply Damper	A	Close	2323-M1-2303-05A	
X-HV-5785B	Spent Fuel Pool HX and Pump Room Exhaust Valve	A	Close	2323-M1-2303-05A	
X-HV-5792A	Spent Fuel Pool HX and Pump Room Supply Damper	B	Close	2323-M1-2303-05A	13
X-HV-5792B	Spent Fuel Pool HX and Pump Room Exhaust Valve	B	Close	2323-M1-2303-05A	13
CPX-VAFNCB-07	Primary Plant Vent Exhaust Fan	A	Start	2323-M1-2309-01	
CPX-VAFNCB-08	Primary Plant Vent Exhaust Fan	B	Start	2323-M1-2309-01	
CPX-SWAPTS-01	Service Water Screen Wash Pump	A	Start	2323-M1-2233-08	
CPX-SWAPTS-02	Service Water Screen Wash Pump	B	Start	2323-M1-2233-08	
X-WL-4290-6	"Traveling Screen -01 Running" Monitor Light	A	Status	2323-M1-2233-07	
1-HV-5365	Demineralized Water, Containment Isolation Valve	B	Close	2323-M1-2242-01	13

SAFETY INJECTION ACTUATED EQUIPMENT LIST (BOP)

Unit Equipment(*) Identification	Description	ESFAS Train	SIS Signal Function	Drawing Number
1-HV-5366	Demineralized Water, Containment Isolation Valve	A	Close	2323-M1-2202-01
1-HV-4710	CCW System, Containment Isolation Valve	B	Close	2323-M1-2231-05
1-HV-4711	CCW System, Containment Isolation Valve	B	Close	2323-M1-2231-05
1-HV-4725	CCW System, Containment Isolation Valve	A	Close	2323-M1-2231-07
1-HV-4726	CCW System, Containment Isolation Valve	B	Close	2323-M1-2231-07
X-WL-4291-7	"Traveling Screen -02 Running" Monitor Light	B	Status	2323-M1-2233-07
X-LV-4288	Service Water Screen Wash Pump-01 Disch. Vlv.	A	Open	2323-M1-2233-07
X-LV-4289	Service Water Screen Wash Pump-02 Disch. Vlv.	B	Open	2323-M1-2233-07
X-HV-5826	Outside Air to Emergency Filtration Units, Damper	A	Open	2323-M1-2304-01
X-HV-5829	Outside Air to Emergency Filtration Units, Damper	B	Open	2323-M1-2304-01
X-HV-5837-A	Control Room Intake Damper	A	Close	2323-M1-2304-03
X-HV-5838-A	Control Room Intake Damper	B	Close	2323-M1-2304-03
1-CP1-CCAPDP-03	Containment CCW Drain Tank Pump	A/B	Stop	2323-M1-2231-07
1-CP1-CCAPDP-04	Containment CCW Drain Tank Pump	A/B	Stop	2323-M1-2231-07
1-CP1-VAFNID-07	Battery Room Exhaust Fan	A	Start	2323-M1-2305-04
1-CP1-VAFNID-08	Battery Room Exhaust Fan	A	Start	2323-M1-2305-04
1-CP1-VAFNID-09	Battery Room Exhaust Fan	B	Start	2323-M1-2305-04

SAFETY INJECTION ACTUATED EQUIPMENT LIST (BOP)

U n i t Equipment(*) Identification	Description	ESFAS Train	SIS Signal Function	Drawing Number
1-CP1-VAFNID-10	Battery Room Exhaust Fan	B	Start	2323-M1-2305-04
1-CP1-AFAPCT-01	Condensate Transfer Pump	A/B	Stop	2323-M1-2206-09 36
1-FV-4536	Component Cooling Water Loop 1 Recirc. Control Vlv.	A	Close	2323-M1-2229-04
1-FV-4537	Component Cooling Water Loop 2 Recirc. Control Vlv.	B	Close	2323-M1-2229-04
1-HV-4572	RHR Heat Exchanger Discharge Control Valve	A	Half-Open	2323-M1-2229-06
1-HV-4573	RHR Heat Exchanger Discharge Control Valve	B	Half-Open	2323-M1-2229-06
1-HV-4631A	CCW Non-Safeguard Loop to Primary Sampling Sys. Valve	A	Close	2323-M1-2230-01
1-HV-4631B	Primary Sample Sys. to CCW Return HDR Valve	A	Close	2323-M1-2230-01
1-HV-4663A	CCW To I.A. Compressor Control Valve	A	Close	2323-M1-2230-02
1-HV-4663B	I.A. Compressor CCW Disch. Control Valve	A	Close	2323-M1-2230-02
1-FV-4650A	Vent Chillers CCW Supply Control Valve	B	Close	2323-M1-2230-02
1-FV-4650B	Vent Chillers CCW Disch. Control Valve	B	Close	2323-M1-2230-02
1-CP1-SWAPSW-01	Station Service Water Pump-01	A	Start	2323-M1-2233-01
1-CP1-SWAPSW-02	Station Service Water Pump-02	B	Start	2323-M1-2233-01
1-PV-4252	SSW Pump-01 Recirc. Line Control Valve	A	Close	2323-M1-2233-01
1-PV-4253	SSW Pump-02 Recirc. Line Control Valve	B	Close	2323-M1-2233-01
1-CP1-CHAPCP-05	Chilled Water Recirc. Pump-05	A	Start	2323-M1-2311-01
1-CP1-CHAPCP-06	Chilled Water Recirc. Pump-06	B	Start	2323-M1-2311-01
1-CP1-CCAPCC-01	Component Cooling Water Pump-01	A	Start	2323-M1-2229-03

SAFETY INJECTION ACTUATED EQUIPMENT LIST (BOP)

Unit Equipment(*) Identification	Description	ESFAS Train	SIS Signal Function	Drawing Number
1-CP1-CCAPCC-02	Component Cooling Water Pump-02	B	Start	2323-81-2229-03
1-CP1-MEDGEE-01	Diesel Generator -01	A	Start	FIG. 8.3-3
1-CP1-MEDGEE-02	Diesel Generator -02	B	Start	FIG. 8.3-3

(*) This table lists only that I&C Equipment actuated directly by an "S" signal for any subsidiary equipment, in turn started by this I&C equipment refer to applicable drawings.

** All other Non-class 1E equipment powered by Class-1E busses, is stripped by an "S" signal.

Shutdown Panel for Train B) that transfers control of equipment 32
from the Control Room to the Hot Shutdown Panel. Placing this Q032.25
local transfer switch in the local position will electrically
isolate all Control Room and Cable Spreading Room equipment from Q032.23
the rest of the circuitry, will provide audible and visible Q032.66
indication in the Control Room and will turn off status lights on
the main control board. Interlocks which are generated in the
Control Room or Cable Spreading Room, such as the lockout signals 36
from the sequencer, are also isolated when control is tranferred
to the Hot Shutdown Panel.

11. Each control circuit consists of cables that 1) connect the
transfer switches to control switches in the Control Room, 2)
connect the transfer switches to the HSP-mounted control
switches, 3) connect the transfer switches to the pertinent motor
control center. Each of these cables are inherently separated
from their redundant counterpart. In addition, the cables from 11
the HSP to the Control Room and the motor control centers are Q040.64
external to the HSP for almost all of their run. Loss of a
control circuit does not mean loss of function since control
circuits are available at the HSP for each of the redundant
systems.

12. To prevent a single event (e.g., short-circuit) from affecting both the Control Room controls and the HSP controls, control-circuit fuses are either located in equipment accessible in such an event (e.g., 6.9 KV and 480V switchgear) or separate fuses, located in separate fire zones, are provided for both the Control Room control circuit and the HSP control circuit.

The controls and monitoring indicators provided on the Hot Shutdown Panel and required for hot standby are listed in Table 7.4-1. Instrumentation and controls provided on the Hot Shutdown Panel for operating convenience or cold shutdown are listed in Table 7.4-2.

Switches provided on the Shutdown Transfer Panel are listed in Table 7.4-3. Instrumentation and controls required for cold shutdown are
32 marked with the letter (c) in Tables 7.4-1, 7.4-2 and 7.4-3. Those marked with the letter (f) are added for Alternate Shutdown for a fire in the Control Room or Cable Spreading Room.

7.4.1.3.2 Hot Standby From Outside The Control Room

Should the Control Room become uninhabitable, the reactor will be manually tripped, the neutron level and control rod position will be verified before evacuation takes place. Also, the reactor can be tripped locally at the reactor trip switchgear which is in close proximity to the Hot Shutdown Panel.

13 Sufficient controls are provided outside the Control Room on a Seismic Category I Hot Shutdown Panel (See Tables 7.4-1 and 7.4-2) and other locations to:

- Achieve prompt hot standby of the reactor

- Maintain the unit in a safe condition during hot standby.

7.4.1.3.3 Cold Shutdown From Outside the Control Room

Cold shutdown can be achieved from outside the Control Room through the use of suitable procedures and by virtue of local control of the systems listed in Section 7.4.1.2. The design bases for the achievement and maintenance of cold shutdown are as listed in Section 7.4.1.3.1. Instrumentation and controls on the Hot Shutdown Panel, marked with the letter (c) on Tables 7.4-1 and 7.4-2, are utilized to attain cold shutdown outside the Control Room. In addition, certain local manipulations of controls and initiating devices, as described below, are required.

For cold shutdown from outside the Control Room due to a Control Room or Cable Spreading Room fire, see Section 7.4.1.3.4. The basic procedure to established cold shutdown from the hot standby condition, for reasons other than fire in the Control Room or Cable Spreading Room and assuming the Control Room is uninhabitable, is as follows. 32 Q212.84

1. Transfer control from the Control Room to the Hot Shutdown Panel, as required, using the transfer switches at the Shutdown Transfer Panel for Train A and the transfer switches at the Hot Shutdown Panel for Train B.

2. Borate to the cold shutdown boron concentration using boric acid transfer pumps and charging pumps.

   Location: Hot Shutdown Panel.

   Available Indications: Reactor Coolant System (RCS) boron concentration determined by sampling.

3. Cooldown the RCS by use of the steam generator power-operated atmospheric relief valves. 9 Q212.84

   Location: Hot Shutdown Panel and local patch panel or manual handwheel operation.

   Available Indications: RCS temperature, pressurizer pressure and level, steam generator level, auxiliary feedwater flow and condensate storage tank level.

4. Depressurize the RCS by throttling the pressurizer spray valve while maintaining pressurizer level.

   Location: Hot Shutdown Panel and local patch panel

Available Indications: pressurizer pressure and level, charging and letdown flow.

5. At 1900 psig in the RCS, block the low pressurizer pressure safety injection (SI) and the low steamline pressure SI signals.

9 Location: Cable spreading room

Q212.84 Available Indications: Pressurizer pressure indication on the Hot Shutdown Panel.

6. At 1000 psig in the RCS, close the accumlator discharge isolation valves and rack out the safety injection pump and containment spray pump breakers.

Location: Switchgear and motor control centers.

Available Indications: Pressurizer pressure indication on the Hot Shutdown Panel.

32 Q212.84 7. At 350°F and 425 psig in the RCS, align the RCS for cooldown with the Residual Heat Removal System.

Location: Switchgear and Hot Shutdown Panel.

Available Indications: RCS pressure indication and RCS temperature indication on the Hot Shutdown Panel.

7.4.1.3.4 Alternate Shutdown System

32 The Hot Shutdown Panel had been designed to enable control to a hot standby condition if, for unspecified but non-catastrophic reasons, the Control Room (CR) had to be evacuated. The circuits required for

Instrumentation Summary Data

Variable	Type/ Category	R.G. 1.97 Type/Cat.	Instrument Quantity	Tag Numbers	Range	Qualification
RCS Pressure (WR)	A1, B1, B2 C1, D2	B1, C1	2 per unit	PT-403, PT-405	0-3000 psig	EQ, SQ
T(HOT) RCS (WR)	A1, B2, C2	B1	1 per loop	TE-413A TE-423A TE-433A TE-443A	0-700^{o}F	EQ, SQ
T(COLD) RCS (WR)	A1, B2	B1, B3	1 per loop	TE-413B TE-423B TE-433B TE-443B	0-700^{o}F	EQ, SQ
Steam Gen. Water Level (WR)	A1, B1, D2, B2	D1	1 per Steam Gen.	LT-501 LT-502 LT-503 LT-504	0-100%	Note 1 SQ(1)
Steam Gen. Water Level (NR)	A1, B1, D2	None	4 per Steam Gen.	LT-517 to 519 LT-527 to 529 LT-537 to 539 LT-547 to 549 LT-551 to 554	0-100%	EQ, SQ
Pressurizer Level	A1, B1, D2	D1	3 per unit	LT-459 LT-460 LT-461	0-100%	EQ, SQ

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Instrumentation Summary Data

Variable	Type/ Category	R.G. 1.97 Type/Cat.	Instrument Quantity	Tag Numbers	Range	Qualification
Containment Pressure (NR)	A1, B1, B2, C2, D2	B1, C1	4 per unit	PT-934 to 937	0-60 psig	EQ, SQ
Steamline Pressure	A1, B1, D2	D2	3 per loop	PT-514 to 516 PT-524 to 526 PT-534 to 536 PT-544 to 546	0-1300 psig	EQ, SQ
RWST Level	A1, D2	D2	4 per unit	LT-930 to 933	0-100%	EQ, SQ
Containment Water Level (WR)	A1, B1, B2, C2, D2	(NR) B2, C2 (WR) B1, C1	2 per unit	LT-4779 (WR) LT-4781 (WR)	808' - 817' 6"	EQ, SQ Note 2
CST Water Level	A1, D2	D1	2 per unit	LT-2478 LT-2479	0-45'	EQ, SQ
Aux. Feedwater Flow	A1, B1, D2	D2	2 per Steam Gen.	PT-2463 A&B PT-2464 A&B PT-2465 A&B PT-2466 A&B	0-550 gpm	EQ, SQ
Containment Rad. Level (High Range)	A1, B1, B2, C2, E2	E1, C3	2 per unit	RE-6290 A&B	10^{0} - 10^{7}R/hr	EQ, SQ
Main Steamline Radiation	A2, B2, C2, E2	E2	1 per steamline	RE-2325 RE-2326 RE-2327 RE-2328	10^{-1} - 10^{3} Ci/cc	EQ(R) Note 3
Steam Gen. Blowdown Rad.	A2, B2, C2	None	1 per unit	RE-4200	10^{-5} - 10^{-1} Ci/cc	EQ(R) Note 3

TABLE 7.5-7
(Sheet 3 of 13)

Instrumentation Summary Data

Variable	Type/ Category	R.G. 1.97 Type/Cat.	Instrument Quantity	Tag Numbers	Range	Qualification	
Condenser Off-gas Radiation	A2, B2, C2	E2, C3	1 per unit	RE-2959	10^{-5} - 10^{-1} Ci/cc	EQ(R)	
Core exit Temperature	A1, B1, C1	C1, B3	25 per train	TE-001 to 050	0-2300^{0}F	EQ, SQ	
RCS Subcooling	A2, B2	B2	2 per unit	TY-CCM-A&B	-300 to 300^{0}F	EQ, SQ	
Neutron Flux	B1	B1	2 IR & 2 SR	NM-35B & 36B NI-31B & 32B	10^{-11} - 10^{-3} amps 1 - 10^{6} cps	Note 4	
Control Rod Position	B3	B3	1 per control rod group	RB1 & RB2	0-100%	Note 5	36
Reactor Vessel Water Level	B2, C2	B1	2 per unit	LY-RVLS-A LY-RVLS-B	Later	Later	
Main Feedwater Flow	B2, D3	D3	1 per Steam Gen.	FT-511 FT-521 FT-531 FT-541	0-5x10^{6} lb/hr	None	36
Containment Hydrogen Concentration	B1	C1	4 per unit	AE-5506 A thru D	0-10%	EQ, SQ	
Reactor Coolant Fission Product Concentration	C3	C3	1 per unit	RCS Sampling System	N/A	N/A	

Instrumentation Summary Data

Variable	Type/ Category	R.G. 1.97 Type/Cat.	Instrument Quantity	Tag Numbers	Range	Qualification
Pressurizer Relief Tank Level	C3	D3	1 per unit	LT-470	0-100%	None
Pressurizer Relief Tank Temperature	C3	D3	1 per unit	TE-468	50-350^{0}F	None
Pressurizer Relief Tank Pressure	C3	D3	1 per unit	PT-469	0-100 psig	None
Containment Pressure (WR)	C1, D2	B1, C1	2 per unit	PT-938 PT-939	0-180 psig	EQ, SQ
Plant Vent Effluent Rad.	C2, E2	C2, E2	1 per vent stack	X-RE-5570 A&B	10^{-7} - 10^{5} Ci/cc	EQ(R)
Area Rad. Levels Adjacent Containment	C2	C2	12 per unit	RE-6259 A&B RE-6291 A&B RE-6292 thru 6299	10^{-1} - 10^{4}R/hr	EQ(R)
			1 per unit	RE-5637	10^{-4} - 10^{0} Ci/cc	EQ(R)
			2 (shared)	XRE-6273 XRE-6275	10^{-1} - 10^{4}mR/hr	EQ(R)
Containment Isolation Valve Status (Active)	C2	B1	1 per active valve	See FSAR Figure 6.2.4-2	N/A	EQ, SQ

Instrumentation Summary Data

Variable	Type/ Category	R.G. 1.97 Type/Cat.	Instrument Quantity	Tag Numbers	Range	Qualification
CCW to RCP Valve Status	D2	None	1 per valve	HV-4699 HV-4700 HV-4709 HV-4708 HV-4701 HV-4696	N/A	None 36
Pressurizer PORV Status	D2	D2	1 per valve	PCV-455A PCV-456	N/A	EQ, SQ
RCS Safety	D2	D2	1 per valve	8010A 8010B 8010C	N/A	EQ, SQ
RCP Seal Water Injection Flow	D2	None	1 per RCP	FT-142 FT-143 FT-144 FT-145	0-20 gpm	None
Pressurizer Heater Breaker Position	D3	None	1 per heater bank	PCPR1 PCPR2 PCPR3 PCPR4	N/A	None
CVCS Makeup Flow	D2	D2	1 per unit	FT-121	0-200 gpm	None
CVCS Letdown Flow	D2	D2	1 per unit	FT-132	0-200 gpm	None
Volume Control Tank Level	D2	D2	1 per unit	LT-112	0-100%	None

TABLE 7.5-7
(Sheet 6 of 13)

Instrumentation Summary Data

Variable	Type/ Category	R.G. 1.97 Type/Cat.	Instrument Quantity	Tag Numbers	Range	Qualification
CVCS Valve Status	D2	None	1 per valve	8100, 8112, 8160, 8152, 8146, 8147, LCV-459, LCV-460, 8145, 8149 A, B, C, 8153, 8154, 8143, HCV-123, TCV-129, PCV-131, TCV-381B	N/A	None
Steam Gen. PORV Status	D2	D2	1 per valve	PV-2325 thru PV-2328	N/A	None
MSIV & Bypass Valve Status	D2	None	1 per valve	HV-2333 A&B HV-2334 A&B HV-2335 A&B HV-2336 A&B	N/A	None
Steam Gen. Safety Valve Status	D2	D2	1 per valve	IMS-093 to 097 IMS-058 to 062 IMS-021 to 025 IMS-129 to 133	N/A	None
AFW Pump Turbine Main Steam Header Isolation Valve Status	D2	None	1 per valve	HV-2452-1 HV-2452-2	N/A	None
Main Feedwater Control and Bypass Valve Status	D3	None	1 per valve	FCV-510, 520, 530, 540 LV-2162, 2163, 2164, 2165	N/A	None

TABLE 7.5-7

Instrumentation Summary Data

Variable	Type/ Category	R.G. 1.97 Type/Cat.	Instrument Quantity	Tag Numbers	Range	Qualification	
Main Feedwater Isolation Valve Status	D3	None	1 per valve	HV-2134 to 2137 FV-2193 to 2196 HV-2185 to 2188	N/A	None	36
High Head Safety Injection Flow	D2	D2	1 per train	FT-918 FT-922	0-800 gpm	None	
Low Head Safety Injection Flow (RHR)	D2	D2	1 per train	FT-618 FT-619	0-6000 gpm	None	36
ECCS Valve Status	D2	None	1 per valve	LCV-112 B,C, 8105, 8106, 8479 A, B, 8808A, B, C, D, 8812A, B, 8811 A, B, 8809A, B, 8804A, B	N/A	None	
SI Accumulator Tank Level	D3	D2	2 per tank	LT-950 to 957	0-100%	None	
SI Accumulator Tank Pressure	D3	D2	2 per tank	PT-960 to 967	0-100%	None	
S/G Blowdown Isolation Valve Status	D2	None	1 per valve	HV-2397, 2398, 2399, & 2400	N/A	None	
Aux. Feedwater Valve Status	D2	None	1 per valve	PV-2453A, B, PV-2454A, B, HV-2459 to 2462 HV-2491A, B, HV-2492A, B, HV-2493A, B, HV-2494A, B	N/A	None	

TABLE 7.5-7
(Sheet 8 of 13)

Instrumentation Summary Data

Variable	Type/ Category	R.G. 1.97 Type/Cat.	Instrument Quantity	Tag Numbers	Range	Qualification	
Containment Spray Flow	D2	D2	1 per pump	FT-4772-1&2 FT-4773-1&2	0-4000 gpm	None	36
Containment Spray Valve Status	D2	None	1 per valve	HV-4777, FV-4773-1&2, HV-4759, HV-4776, FV-4771-1&2, HV-4758	N/A	None	
Containment Atmosphere Temperature	D2	D2	5 per unit	TE-5400 to 5404	0-300^{o}F	EQ, SQ	
CCW Header Pressure	D2	None	1 per header	PT-4520 PT-4521	0-200 psig	None	
CCW Header Temperature	D2	D2	1 per header	TE-4530 TE-4534	0-200^{o}F	None	
CCW Surge Tank Level	D2	None	1 per train	LT-4500 LT-4501	0-6'		
CCW Flow	D2	D2	1 per train	FT-4536A FT-4537A	0-20,000 gpm	None	
CCW Valve Status	D2	None	1 per valve	HV-4513, HV-4512, HV-4572, HV-4573, HV-4574, HV-4576, HV-4575, HV-4537, HV-4536, HV-4514, HV-4515	N/A	None	

Instrumentation Summary Data

Variable	Type/ Category	R.G. 1.97 Type/Cat.	Instrument Quantity	Tag Numbers	Range	Qualification
Service Water Header Flow	D2	None	1 per train	FT-4258 FT-4259	0-20,000 gpm	None
CR A/C Units	D2	None	1 per A/C unit	WL-5847B WL-5848B WL-5851 WL-5853	N/A	None
CR Vent Damper Position	D2	D2	1 per damper	X-HV-5826, -5829, -5837, -5838, -5839, -5840, HV-5847, -5848, -5851, -5853, PV-5855, -5856	N/A	None
Chilled Water Flow	D2	None	1 per train	FT-6708 FT-6709	0-300 gpm	None
AC & DC Bus Availability	D2	D2	1 per bus	1EB1, 1EB2, 1EB3, 1EB4 1PC1, 1PC2, 1PC3, 1PC4 1EC1, 1EC2, 1EC3, 1EC4 1ED1, 1ED2	N/A	None
RHR Heat Exchanger Discharge Temperature	D2	D2	1 per train		50-400^{0}F	None

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TABLE 7.5-7
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Instrumentation Summary Data

Variable	Type/ Category	R.G. 1.97 Type/Cat.	Instrument Quantity	Tag Numbers	Range	Qualification
RHR Valve Status	D2	None	1 per valve	HCV-606, -607, 8701-A&B, 8702-A&B, 8716-A&B, FCV-610, -611, FCV-618, -619	N/A	None
CR Radiation	E2	E2	2 plus mobile monitor	XRE-6281 XRE-6282	10^{-1} - 10^{4}mR/hr	
RHR Pump Room Radiation	E2	E2	1 per room	RE-6260A RE-6260B	10^{-1} - 10^{4}R/hr	EQ(R)
Sample Room Radiation	E2	E2	1	RE-6261	10^{-1} - 10^{4}mR/hr	EQ(R)
Plant Vent Stack Sample Area Rad.	E2	E2	1	RE-6259	10^{-1} - 10^{4}mR/hr	EQ(R)
Hot Lab Area Radiation	E2	E2	1	X-RE-6283	10^{-1} - 10^{4}mR/hr	EQ(R)
Liquid Waste Effluent Rad.	E2	None	1	X-RE-5253	10^{-5} - 10^{-1} Ci/cc	EQ(R)
Turbine Building Drains Radiation	E2	None	1	RE-5100	10^{-5} - 10^{-1} Ci/cc	EQ(R)
Evirons Radiation	E2	E3	As required	N/A Portable	As required	None
Wind Speed	E2	E3	2	X-SR-4128 X-SR-4129	0-100 mph	None

TABLE 7.5-7
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Instrumentation Summary Data

Variable	Type/ Category	R.G. 1.97 Type/Cat.	Instrument Quantity	Tag Numbers	Range	Qualification
Wind Direction	E2	E3	2	X-ZR-4126 X-ZR-4127	0-540^{o}	None
Atmospheric Stability	E2	E3	1	X-TR-4130	N/A	None

Instrumentation Summary Data

Gen.	- generator	ECCS	- Emergency Core Cooling System
WR	- wide range	SI	- Safety Injection
NR	- narrow range	CR	- Control Room
RCS	- Reactor Coolant System	A/C	- air conditioning
Aux.	- auxiliary	Vent	- ventilation
RWST	- Reactor Water Storage Tank	RHR	- Residual Heat Removal
CST	- Condensate Storage Tank	EQ	- Environmentaly Qualified
Rad.	- radiation	SQ	- Seismically Qualified
IR	- intermediate range	SQ(1)	- Sensor only seismically qualified
SR	- source range	AFW	- Auxiliary feedwater
CCW	- Component Cooling Water	EQ(R)	- Environmentally Qualified for Radiation Extremes
RCP	- Reactor Coolant Pump		
CVCS	- Chemical and Volume Control System		
N/A	- not applicable		
MSIV	- main steam isolation valve		

TABLE 7.5-7
(Sheet 13 of 13)

Instrumentation Summary Data

NOTES

1. Steam generator water level (WR) is utilized in conjunction with AFW flow for determining when to terminate SI for secondary breaks outside containment. Steam generator water level (WR) is only used for secondary breaks outside containment (the hostile environment that results from secondary breaks inside containment induces unacceptable errors). SI termination for secondary breaks inside containment is based on AFW flow.

2. The Containment Water Level (WR) covers the entire range of expected water level in the Containment for post accident conditions. Therefore, Containment Water Level (NR) is not considered as required for accident monitoring.

3. Main Steamline Radiation and Steam Generator Blowdown Radiation are only required for (and qualified for) steam generator tube rupture detection.

4. The Source Range and Intermediate Range Neutron Flux detectors are not qualified for the accident environment inside containment. Qualified detectors are not available. Adequate determination of reactivity control can be made using backup variables.

5. Control Rod Position signals do not exist for input to the ERF Computer. Control Rod Position Indication is available on the Main Control Boards and Plant Process Computer. 36

7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

The general design objectives of the plant control systems are:

1. To establish and maintain power equilibrium between primary and secondary system during steady state unit operation.

2. To constrain operational transients so as to preclude unit trip and re-establish steady state unit operation.

3. To provide the reactor operator with monitoring instrumentation that indicates all required input and output control parameters of the systems and provides the operator the capability of assuming manual control of the system.

7.7.1 DESCRIPTION

The plant control systems described in this section perform the following functions:

1. Reactor Control System

   a. Enables the nuclear plant to accept a step load increase or decrease of 10 percent and a ramp increase or decrease of 5 percent per minute within the load range of 15 percent to 100 percent without reactor trip, steam dump, or pressurizer relief actuation, subject to possible xenon limitations.

   b. Maintains reactor coolant average temperature within prescribed limits by creating the bank demand signals for moving groups of full length rod cluster control assemblies during normal operation and operational transients. The reactor coolant average temperature control also supplies a signal to pressurizer water level control, and steam dump control.

2. Rod Control System

   a. Provides for reactor power modulation by manual or automatic control of full length control rod banks in a pre-selected sequence and for manual operation of individual banks.

   b. Systems for monitoring and indicating

      1) Provide alarms to alert the operator if the required core reactivity shutdown margin is not available due to excessive control rod insertion.

      2) Display control rod position.

      3) Provide alarms to alert the operator in the event of control rod deviation exceeding a preset limit.

3. Plant Control System interlocks

   a. Prevent further withdrawal of the control banks when signal limits are approached that predict the approach of a departure from nucleate boiling ratio (DNBR) limit or kilowatt per feet (kW/ft) limit.

   b. Inhibit automatic turbine load change as required by the Nuclear Steam Supply System (NSSS).

4. Pressurizer pressure control

36 a. Maintains or restores the pressurizer pressure to the design pressure ± 30 pounds per square inch (psi) (which is well within reactor trip and relief and safety valve actuation setpoint

Two diesel generators are provided for each unit. Each diesel generator is connected exclusively to its associated 6.9-kV Class 1E bus, which ensures independence in the onsite standby power sources.

Each Class 1E DC bus can be energized either by a battery or by one of two battery chargers (one spare) or combination of battery and battery charger. There are no automatic or manual ties between Class 1E redundant DC load groups. Arrangement of the AC and DC systems is described in Subsections 8.3.1 and 8.3.2, respectively.

Because there are no bus ties between redundant load groups, interlocks are not required.

4. Compliance With NRC Regulatory Guide 1.9 [3]

The rating of the diesel generators is based on the maximum continuous load demand. This rating exceeds the sum of the conservatively rated loads. Motor loads are based on nameplate rating, pump runout conditions, or flow pressure conditions. 6600-V motor efficiency is based on design data. Low-voltage motor efficiency is assumed to be 80 percent.

During preoperational testing, the maximum continuous load demand is verified by tests.

Each diesel generator set is capable of starting and accelerating to rated speed all Class 1E loads in the required sequence.

Sequencing of large loads at 5-sec intervals ensures that large motors have reached rated speed and that voltage and frequency have stabilized before the succeeding loads are applied. The voltage may dip below 75 percent of nominal voltage when the

diesel generator breaker closes and energizes the two 2000/2666 kVA, 6.9 kV/480-V unit substation transformers supplied from each diesel generator. This dip is due to magnetizing inrush current which exists for two to three cycles. The diesel generators are designed to recover to 80 percent of nominal voltage within 10 cycles for this transient. The effect on the first load group would, therefore, be a maximum possible delay of 12 to 13 cycle after closure of the diesel generator breaker. However, the objective of first load group and subsequent load groups is not affected. During recovery from transients caused by step load increases or resulting from the disconnection of the largest
36 single load, the speed of the diesel generator set should not exceed the nominal speed plus 75 percent of the difference between nominal speed and the overspeed trip setpoint or 115 percent of nominal, whichever is lower. The voltage is restored to within 10 percent of nominal; and the frequency is restored to within two percent of nominal in less than 40 percent of each
7 load sequence time internal. The diesel generator supplier has
Q040.60 successfully performed these tests in his facility on one CPSES diesel generator set.

The prototype qualification test program of

8 a) Start and load capability at full load, and

7 Q040.60 b) 300 valid start and load tests

on the diesel generator are discussed in Section 8.3.1.1.11.

5. Compliance With Regulatory Guide 1.32 [7]

The offsite power system includes the preferred design stated in NRC Regulatory Guide 1.32: namely, two immediate access circuits from the transmission network are available to the emergency

(Class 1E) bus systems. A third supply is available upon the removal of the isolated phase bus links in less than eight hours. In this case, power is backfed from the 345 kV Switchyard through the main and unit auxiliary transformers to the Class 1E buses.

Each battery charger is sized to handle the combined steady-state loads while recharging the battery from the design minimum charge state to the fully charged state under all modes of plant operation.

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
INJECTION PHASE(7)

Component**	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After SIAS (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(13) FLA	(13) P.F.	(12) Eff.	kW	Basis for hp Required*	Time To Stop	
Centrifugal charging pump	600	yes	10	2	1	660 (8)	6600	50.6	.914	-	528.7	(b)	(4)	36
Motor-operated valves	230	yes	10	-	as required	230	460	-	-	-	(14)	estimated	(6)	
Diesel-generator oil day tank area fan	1.5	yes	10	2	1	1.5	460	-	-	.8	1.4	(a)	(4)	
Diesel-generator room fans	40	yes	10	8	4	160	460	-	-	.8	149.2	(a)	(4)	
Diesel-generator fuel oil transfer pump	3	yes	10	4	2	6	460	-	-	.8	5.6	(a)	(5)	36
Floor drain sump pump	5	yes (10)	as required after 10 sec.	4	2	10	460	-	-	.8	9.3	(a)	(6)	
BOP Inverter (Class 1E)	21 kW	yes	10	4	2	-	460	-	-	-	42	(a)	(4)	
NSSS Inverter (Class 1E)	18.7 kW	yes	10	4	2	-	460	-	-	-	37.4	(a)	(4)	
Control room emergency pressurization fan	5	yes	10	2 (2)	1	5	460	-	-	0.8	4.7	(a)	(4)	
Control room emergency filter fan	30	yes	10	2 (2)	1	30	460	-	-	0.8	28	(a)	(4)	36

TABLE 8.3-1A
(Sheet 2 of 8)

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
INJECTION PHASE(7)

Component**	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After SIAS (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(13) FLA	(13) P.F.	(12) Eff.	kW	Basis for hp Required*	Time To Stop	
Centrifugal charging pump area fan coil unit	3	yes (10)	10	2	1	3	460	-	-	0.8	2.8	(a)	(4)	
Battery room exhaust fans	1.5	yes	10	4	2	3	460	-	-	0.8	2.8	(a)	(4)	
Emergency ac lighting	204.5 kW	yes	10	-	as required	-	120	-	-	-	204.5	estimated	(4)	
Battery chargers (Class 1E)	44.8 kW	yes	10	8	2	-	480	-	-	-	44.8	estimated	(4)	36
SUPS bypass transformer	7.5 kW	yes	10	2	1	-	480	-	-	-	7.5	estimated	(4)	
Auxiliary building isolation transformer	67.5 kW	yes	10	2 (2)	1	-	480	-	-	-	67.5	(d)	(4)	
Control room A/C unit transformer	7.1 kW	yes (10)	10	4 (2)	2	-	480	-	-	-	14.2	estimated	(4)	
UPS and DIST.RM. A/C unit control transformer	0.75 kW	yes	10	2	1	-	480	-	-	-	0.75	estimated	(4)	
Alternate shutdown isolation transformer	7.5 kW	yes	10	1	1	-	480	-	-	-	7.5	(d)	(4)	
Safeguard bldg. transformer T1EC3-2	45 kW	yes	10	2	1	-	480	-	-	-	15 kW	estimated	(5)	36
Technical support center isolation transformer	45 kW	yes	10	1	1	-	480	-	-	-	45	(d)	(4)	

TABLE 8.3-1A
(Sheet 3 of 8)

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
INJECTION PHASE(7)

Component**	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After SIAS (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(13) FLA	(13) P.F.	(12) Eff.	kW	Basis for hp Required*	Time To Stop	
Safety injection pump	450	yes	15	2	1	495 (8)	6600	38.5	.914	-	402.3	(b)	(4)	36
Chilled water recirculation pump NSR Class 3	25	yes	15	2	1	25	460	-	-	.8	23.3	(a)	(4)	
SIS pump area fan coil unit	3	yes (10)	15	2	1	3	460	-	-	.8	2.8	(a)	(4)	
RHR pump	450	yes	20	2	1	420	6600	31.7	.93	-	337	(c)	(4)	36
RHR pump area fan coil unit	3	yes (10)	20	2	1	3	460	-	-	.8	2.8	(a)	(4)	
Containment spray pumps	700	yes	25	4	2	1600 (8)	6600	116.6	.945	-	1259.6	(b)	(4)	36
Containment spray pump area fan coil units	3	yes (10)	25	4	2	6	460	-	-	.8	5.6	(a)	(4)	
CCWP	1000	yes	30	2	1	1000	6600	74.6	.918	-	782.9	(a)	(4)	
CCWP area fan coil unit	3	yes (10)	30	2	1	3	460	-	-	.8	2.8	(a)	(4)	36
Service water pump	900	yes	35	2	1	900	6600	72.4	.873	-	722.5	(a)	(4)	
Auxiliary feedwater pump	700	yes	40	2	1	700	6600	54.5	.907	-	565.1	(a)	(4)	
Auxiliary feedwater pump area fan coil unit	3	yes (10)	40	2	1	3	460	-	-	.8	2.8	(a)	(4)	

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
INJECTION PHASE(7)

Component**	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After SIAS (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(13) FLA	(13) P.F.	(12) Eff.	kW	Basis for hp Required*	Time To Stop
Primary plant vent. exhaust fan	60	yes	40	16 (2)	1	60	460	-	-	.8	56	(a)	(4)
Primary Plant ESF Filtration Unit Electric Heater	100 kW	yes	40	2 (2)	1	-	480	-	-	-	100	(a)	(5)
HVAC Centrifugal water chillers (NSR Class 3)	154	yes	75	2	1	154	460	181.6	0.904	-	130.8	(a)	(5)
UPS and DIST.RM. A/C unit compressor	50	yes	90	2	1	50	460	-	-	0.8	46.6	(a)	(4)
UPS and DIST.RM. A/C unit fan	10	yes	90	2	1	10	460	-	-	0.8	9.3	(a)	(4)
UPS and DIST.RM. A/C unit air-compressor	1	yes (10)	as required after 90 sec.	2	1	1	460	-	-	0.8	0.93	(a)	(5)
UPS and DIST.RM. booster return fan	20	yes	90	2	1	20	460	-	-	0.8	18.7	(a)	(5)
Control room air conditioning unit fan	50	yes	90	4 (2)	2	100	460	-	-	.8	93.3	(a)	(4)
Control room electric heater	30 kW	yes (10)	as required after 90 sec.	4 (2)	2 as required	-	480	-	-	-	(15)	(a)	(4)

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LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
INJECTION PHASE(7)

Component**	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After SIAS (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(13) FLA	(13) P.F.	(12) Eff.	kW	Basis for hp Required*	Time To Stop	
Control room emergency pressurization heater	20 kW	yes (10)	as required after 10 sec.	2 (2)	1	-	480	-	-	-	20	(a)	(5)	
Control room A/C unit compressor	175	yes (10)	as required after 90 sec.	4 (2)	2	350	460	394	0.895	-	281	(a)	(4)	36
Control room makeup supply fan	2	no (3)	as required	2 (2)	1	2	460	-	-	.8	1.9	(a)	(4)	
Control room exhaust fan	5	no (3)	as required	2 (2)	1	5	460	-	-	.8	4.7	(a)	(4)	
Control room kitchen and toilet exhaust fan	1.5	no (3)	as required	2 (2)	1	1.5	460	-	-	.8	1.4	(a)	(4)	
Electrical area fan coil unit	5	yes	90	4	2	10	460	-	-	.8	9.3	(a)	(4)	
Containment H2 analysis panel	1	no (3)	as required	2 (2)	1	1	460	-	-	.8	.93	(a)	(4)	36
Service water intake structure fans	3	no (3)	as required	8 (2)	4	12	460	-	-	.8	11.2	(a)	(4)	
Spent fuel pool pump	250	no (3)	as required after 99 sec.	2 (2)	1	250	460	287	0.881	-	201.5	(a)	(4)	

TABLE 8.3-1A
(Sheet 6 of 8)

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
INJECTION PHASE(7)

Component**	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After SIAS (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(13) FLA	(13) P.F.	(12) Eff.	kW	Basis for hp Required*	Time To Stop	
Reactor makeup water pump	40	no (3)	as required after 90 sec.	1 (11)	1	40	460	-	-	0.8	37.3	(a)	(5)	36
Spent fuel pool area fan coil unit	3	no (3)	as required	2 (2)	1	3	460	-	-	.8	2.8	(a)	(4)	
Total kW											6356			36

TABLE 8.3-1A
(Sheet 7 of 8)

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
INJECTION PHASE(7)

Total Load on Diesel Generator (kW) Train A	6356	36
Total Load on Diesel Generator (kW) Train B	6356	

*Basis for hp Required

**Where "Pump (Fan)" components are listed, it is understood that the pump (fan) motor is the actual electrical load component.

(a) Nameplate Rating
(b) Pump Runout Condition
(c) Estimated Flow-Pressure Condition or Bhp
(d) Maximum (Current Limiting) Output

NOTES:

(1) Maximum time to close breaker including 10 sec for diesel to come up to speed and voltage; the delay times for automatic start are 10 sec less when offsite power is available.

(2) Equipment is shared between two units and number shown is total for two units.

(3) Manual start when required

(4) Manually stopped

(5) Stops automatically with assigned diesel or pump, or temperature, or pressure, and so forth

(6) Motor stops automatically when valve action is completed, or receives signal to stop (e.g. sump pump stops on low water level).

(7) In steam Line Break condition automatic loading sequence is same as in Loss-of-Coolant Accident condition.

(8) Motor Service Factor of 1.15 meets this hp requirement

(9) Not Used 36

(10) Starts automatically with assigned load or upon temperature, pressure, level switch signal, etc.

(11) One additional component is supplied and is shared by same train of both units;

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
INJECTION PHASE(7)

(12) The following assumptions were made for 480V motors in calculating total load on diesel generator:

a) Less than 100 hp
Power factor 0.8 and efficiency 80%

b) 100 hp and above
per manufacturer's data or, if not available,
Power factor 0.9 and efficiency 85%

$$kW = \frac{HP \times .746}{Efficiency}$$

(13) Manufacturers data is used for all medium voltage motors in calculating total load on diesel generator. $kW = \sqrt{3}\ VICos\ \phi$

(14) Valves have short operating times and are not included in the total continuous load.

(15) Not included in total continuous load since heater and full compressor demand do not occur simultaneously. Compressor load is included in total continuous load as worst case.

Abbreviations: component cooling water pump (CCWP); safety injection system (SIS); residual heat removal (RHR); heating, ventilating, and air conditioning (HVAC); balance of plant (BOP); Nuclear Safety Related (NSR), air conditioning (A/C) full load ampere (FLA), power factor (PF), Efficiency (Eff.)

TABLE 8.3-1B
(Sheet 1 of 8)

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
RECIRCULATION PHASE(7)

Component**	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After SIAS (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(13) FLA	(13) P.F.	(12) Eff.	kW	Basis for hp Required*	Time To Stop
Centrifugal charging pump	600	yes	10	2	1	660 (8)	6600	50.6	.914	-	528.7	(b)	(4) 36
Motor-operated valves	230	yes	10	-	as required	230	-	-	-	-	(14)	estimated	(6)
Diesel-generator oil day tank area fan	1.5	yes	10	2	1	1.5	460	-	-	.8	1.4	(a)	(4)
Diesel-generator room fans	40	yes	10	8	4	160	460	-	-	.8	149.2	(a)	(4)
Diesel-generator fuel oil transfer pump	3	yes	10	4	2	6	460	-	-	.8	5.6	(a)	(5)
Floor drain sump pump	5	(10)	as required after 10 sec.	4	2	10	460	-	-	.8	9.3	(a)	(6) 36
BOP Inverter (Class 1E)	21 kW	yes	10	4	2	-	460	-	-	-	42	(a)	(4)
NSSS Inverter (Class 1E)	18.7 kW	yes	10	4	2	-	460	-	-	-	37.4	(a)	(4)
Control room emergency pressurization fan	5	yes	10	2 (2)	1	5	460	-	-	0.8	4.7	(a)	(4)
Control room emergency filter fan	30	yes	10	2 (2)	1	30	460	-	-	0.8	28	(a)	(4) 36

TABLE 8.3-1B
(Sheet 2 of 8)

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
RECIRCULATION PHASE(7)

Component**	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After SIAS (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(13) FLA	(13) P.F.	(12) Eff.	kW	Basis for hp Required*	Time To Stop	
Centrifugal charging pump area fan coil unit	3	yes (10)	10	2	1	3	460	-	-	0.8	2.8	(a)	(4)	
Battery room exhaust fans	1.5	yes	10	4	2	3	460	-	-	0.8	2.8	(a)	(4)	
Emergency ac lighting	204.5 kW	yes	as 10	-	as required	-	120	-	-	-	204.5	estimated	(4)	
Battery chargers (Class 1E)	44.8 kW	yes	10	8	2	-	480	-	-	-	44.8	estimated	(4)	36
SUPS bypass transformer	7.5 kW	yes	10	2	1	-	480	-	-	-	7.5	estimated	(4)	
Auxiliary building equipment isolation transformer	67.5 kW	yes	10	2 (2)	1	-	480	-	-	-	67.5	(d)	(4)	
Control room A/C unit transformer	7.1 kW	yes	10	4 (2)	2	-	480	-	-	-	14.2	estimated	(4)	
UPS and DIST.RM A/C unit control transformer	0.75 kW	yes	10	2	1	-	480	-	-	-	0.75	estimated	(4)	36
Alternate shutdown isolation transformer	7.5 kW	yes	10	1	1	-	480	-	-	-	7.5 kW	(d)	(4)	
Safeguard Bldg. transformer T1EC3-2	45 kW	yes	10	2	1	-	480	-	-	-	15 kW	estimated	(5)	

TABLE 8.3-1B
(Sheet 3 of 8)

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
RECIRCULATION PHASE(7)

Component**	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After SIAS (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(13) FLA	(13) P.F.	(12) Eff.	kW	Basis for hp Required*	Time To Stop	
Technical support center isolation transformer	45 kW	yes	10	2	1	-	480	-	-	-	45	(d)	(4)	36
Safety injection pump	450	yes	15	2	1	495 (8)	6600	38.5	.914	-	402.3	(b)	(4)	
Chilled water recirculation pump NSR Class 3	25	yes	15	2	1	25	460	-	-	.8	23.3	(a)	(4)	
SIS pump area fan coil unit	3	yes (10)	15	2	1	3	460	-	-	.8	2.8	(a)	(4)	
RHR pump	450	yes	20	2	1	420	6600	31.7	.93	-	337	(c)	(4)	36
RHR pump area fan coil unit	3	yes (10)	20	2	1	3	460	-	-	.8	2.8	(a)	(4)	
Containment spray pumps	700	yes	25	4	2	1600 (8)	6600	116.6	.945	-	1259.6	(b)	(4)	36
Containment spray pump area fan coil units	3	yes (10)	25	4	2	6	460	-	-	.8	5.6	(a)	(4)	
CCWP	1000	yes	30	2	1	1000	6600	74.6	.918	-	782.9	(a)	(4)	
CCWP area fan coil unit	3	yes (10)	30	2	1	3	460	-	-	.8	2.8	(a)	(4)	36
Service water pump	900	yes	35	2	1	900	6600	72.4	.873	-	722.5	(a)	(4)	
Auxiliary feedwater pump	700	yes	40	2	1	700	6600	54.5	.907	-	565.1	(a)	(4)	

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
RECIRCULATION PHASE(7)

Component**	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After SIAS (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(13) FLA	(13) P.F.	(12) Eff.	kW	Basis for hp Required*	Time To Stop
Auxiliary feedwater pump area fan coil unit	3	yes (10)	40	2	1	3	460	-	-	.8	2.8	(a)	(4)
Primary plant vent exhaust fan	60	yes	40	16 (2)	1	60	460	-	-	.8	56	(a)	(4)
Primary Plant ESF Filtration Unit Electric Heater	100 kW	yes	40	2 (2)	1	100 kW	-	-	-	-	100	(a)	(5)
HVAC Centrifugal water chillers (NSR Class 3)	154	yes	75	2	1	154	460	181.6	0.904	-	130.8	(a)	(5)
UPS and DIST.RM. A/C unit compressor	50	yes	90	2	1	50	460	-	-	0.8	46.6	(a)	(4)
UPS and DIST.RM. A/C unit fan	10	yes	90	2	1	10	460	-	-	0.8	9.3	(a)	(4)
UPS and DIST.RM. A/C unit air-compressor	1	yes (10)	as required after 90 sec.	2	1	1	460	-	-	0.8	0.93	(a)	(5)
UPS and DIST.RM. booster return fan	20	yes	90	2	1	20	460	-	-	0.8	18.7	(a)	(5)
Control room air conditioning unit fan	50	yes	90	4 (2)	2	100	460	-	-	.8	93.3	(a)	(4)

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TABLE 8.3-18
(Sheet 5 of 8)

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
RECIRCULATION PHASE[7]

Component**	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After SIAS (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(13) FLA	(13) P.F.	(12) Eff.	kW	Basis for hp Required*	Time To Stop
Control room electric heater	30 kW	yes (10)	as required after 90 sec.	4 (2)	2	-	480	-	-	-	(15)	(a)	(4)
Control room emergency pressurization heater	20 kW	yes (10)	as required after 10 sec.	2 (2)	1	-	480	-	-	-	20	(a)	(5)
Control room A/C unit compressor	175	yes (10)	as required after 90 sec.	4 (2)	2	350	460	394	.895	-	281	(a)	(4)
Control room makeup supply fan	2	no (3)	as required	2 (2)	1	2	460	-	-	.8	1.9	(a)	(4)
Control room exhaust fan	5	no (3)	as required	2 (2)	1	5	460	-	-	.8	4.7	(a)	(4)
Control room kitchen and toilet exhaust fan	1.5	no (3)	as required	2 (2)	1	1.5	460	-	-	.8	1.4	(a)	(4)
Electrical area fan coil unit	5	yes	90	4	2	10	460	-	-	.8	9.3	(a)	(4)
Containment H2 analyze panel	1	no (3)	as required	2 (2)	1	1	460	-	-	.8	.93	(a)	(4)
Reactor makeup water pump	40	no (3)	as required after 90 sec.	1 (11)	1	40	460	-	-	0.8	37.3	(a)	(5)

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
RECIRCULATION PHASE[7]

Component**	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After SIAS (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(13) FLA	(13) P.F.	(12) Eff.	kW	Basis for hp Required*	Time To Stop
Service water intake structure fans	3	no (3)	5 min.	8 (2)	4	12	460	-	-	.8	11.2	(a)	(4)
Spent fuel pool pump	250	no (3)	as required after 1[illegible] sec.	2 (2)	1	250	460	287	0.881	-	201.5	(a)	(4)
Spent fuel pool area fan coil unit	3	no (3)	as required	2 (2)	1	3	460	-	-	.8	2.8	(a)	(4)
Boric acid transfer pump	15.5 kW	yes (10)	as required	2	1	-	460	-	-	-	15.5	(a)	(4)
Electric hydrogen recombiner	75 kW	no	12 hrs.	2	1	-	480	-	-	-	(16)	(a)	(4)
Hydrogen purge electric heater	30 kW	no	as required after 5 days	2 (2)	1	-	480	-	-	-	(16)	(a)	(5)
Hydrogen purge supply fan	40	no	as required after 5 days	2 (2)	1	40	460	-	-	-	(16)	(a)	(4)
Hydrogen purge exhaust fan	10	no	as required after 5 days	2 (2)	1	-	460	-	-	-	(16)	(a)	(4)
Total kW											6371		

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
RECIRCULATION PHASE(7)

Total Load on Diesel Generator (kW) Train A	6371	36
Total Load on Diesel Generator (kW) Train B	6371	

*Basis for hp Required

**Where "Pump (Fan)" components are listed, it is understood that the pump (fan) motor is the actual electrical load component.

(a) Nameplate Rating
(b) Pump Runout Condition
(c) Estimated Flow-Pressure Condition or Bhp
(d) Maximum (Current Limiting) Output

NOTES:

(1) Maximum time to close breaker including 10 sec for diesel to come up to speed and voltage; the delay times for automatic start are 10 sec less when offsite power is available.

(2) Equipment is shared between two units and number shown is total for two units.

(3) Manual start when required

(4) Manually stopped

(5) Stops automatically with assigned diesel or pump, or temperature, or pressure, etc.

(6) Motor stops automatically when valve action is completed, or receives signal to stop (e.g., sump pump stops on low water level).

(7) In steam Line Break condition automatic loading sequence is same as in Loss-of-Coolant Accident condition.

(8) Motor Service Factor of 1.15 meets this hp requirement 36

(9) Not Used

(10) Starts automatically with assigned load or upon temperature, pressure, level switch signal, etc.

(11) One additional component is supplied and is shared by same train of both units;

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT
COINCIDENT WITH LOSS OF OFFSITE POWER
RECIRCULATION PHASE(7)

(12) The following assumptions were made for 480V motors in calculating total load on diesel generator:

a) Less than 100 hp
Power factor 0.8 and efficiency 80%

b) 100 hp and above
per manufacturer's data or, if not available,
Power factor 0.9 and efficiency 85%

$$kW = \frac{HP \; .746}{Efficiency}$$

(13) Manufacturers data is used for all medium voltage motors in calculating total load on diesel generator. $kW = \sqrt{3}\, VI\cos\phi$

(14) Valves have short operating times and are not included in the total continuous load.

(15) Not included in total continuous load since heater and full compressor demand do not occur simultaneously. Compressor load is included in total continuous load as worst case.

(16) Not included in total continuous load to offset load reductions expected at the required start time.

Abbreviations: component cooling water pump (CCWP); safety injection system (SIS); residual heat removal (RHR); heating, ventilating, and air conditioning (HVAC); balance of plant (BOP); Nuclear Safety Related (NSR), air conditioning (A/C) full load ampere (FLA), power factor (PF), Efficiency (Eff.)

TABLE 8.3-2
(Sheet 1 of 11)

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR BLACKOUT CONDITIONS

Component***	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After Blackout (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(11) FLA	(11) P.F.	(10) Eff.	kW	Basis for hp Required*	Time To Stop
Centrifugal charging pump	600	yes	10	2	1	660 (7)	6600	50.6	.914	-	528.7	(b)	(4) 36
Motor-operated valves	230	yes	10	-	as required	230	-	-	-	-	(14)	estimated	(6)
Diesel-generator oil day tank area fan	1.5	yes	10	2	1	1.5	460	-	-	.8	1.4	(a)	(5)
Diesel-generator room fans	40	yes	10	8	4	160	460	-	-	.8	149.2	(a)	(4) 36
Diesel-generator fuel oil booster pump	3	yes	10 (8)	2	1	3	460	-	-	.8	2.8	(a)	(4)
Diesel-generator fuel oil transfer pump	3	yes	10	4	2	6	460	-	-	.8	5.6	(a)	(5)
Diesel-generator auxiliary lube oil pump	60	yes	10 (8)	2	1	60	460	-	-	.8	56	(a)	(5)
Station service water traveling screens	1.5	yes	10	2	1	1.5	460	-	-	.8	1.4	(a)	(5)
Floor drain sump pump	5	yes (9)	as required after 10 sec.	4	2	10	460	-	-	.8	9.3	(a)	(6)
Chilled water recirculation pump NSR Class 3	25	yes	10	2	1	25	460	-	-	.8	23.3	(a)	(4) 36

TABLE 8.3-2
(Sheet 2 of 11)

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR BLACKOUT CONDITIONS

Component***	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After Blackout (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(11) FLA	(11) P.F.	(10) Eff.	kW	Basis for hp Required*	Time To Stop	
Control room emergency pressurization fan	5	yes	10	2 (2)	1	5	460	-	-	0.8	4.7	(a)	(5)	
Control room emergency filter fan	30	yes	10	2 (2)	1	30	460	-	-	0.8	28	(a)	(4)	36
Centrifugal charging pump area fan coil unit	3	yes	10	2	1	3	460	-	-	.8	2.8	(a)	(4)	
Battery room exhaust fans	1.5	yes	10	6	3	4.5	460	-	-	.8	4.2	(a)	(8)	
Control rod drive mechanism vent. fans	125	yes	10	2	1	125	460	143	0.88	-	100.3	(a)	(4)	36
Emergency ac lighting (includes security lighting)	244.5 kW	yes	10	-	-	-	120	-	-	-	244.5	estimated	(4)	
Battery chargers (Class 1E)	44.8 kW	yes	10	8	2	-	480	-	-	-	44.8	estimated	(4)	36
Battery chargers (BOP)	5.6 and 2.8 kW	yes	10	2 (16)	2	-	480	-	-	-	8.4	(a)	(4)	
BOP inverter (Non-Class 1E)	21 kW	yes	10	2	1	-	460	-	-	-	21	(a)	(4)	
NSSS computer inverter	18.7 kW	yes	10	1	-	-	460	-	-	-	18.7	(a)	(4)	36

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR BLACKOUT CONDITIONS

Component***	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After Blackout (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(11) FLA	(11) P.F.	(10) Eff.	kW	Basis for hp Required*	Time To Stop
Control room rad. monitoring system sample pump	1.5	yes	10	2 (2)	1	1.5	460	-	-	.8	1.4	(a)	(4)
Vent. stack rad. monitoring system sample pump	1.5	yes	10	2 (2)	1	1.5	460	-	-	.8	1.4	(a)	(4)
Containment air rad. monitoring system sample pump	1.5	yes	10	1 (15)	1	1.5	460	-	-	.8	(15)	(a)	(4)
SUPS bypass transformer	7.5 kW	yes	10	2	1	-	480	-	-	-	7.5	estimated	(4)
Auxiliary building isolation transformer	67.5 kW	yes	10	2 (2)	1	-	480	-	-	-	67.5	(d)	(4)
Control room A/C unit transformer	7.1 kW	yes (10)	10	4 (2)	2	-	480	-	-	-	14.2	estimated	(4)
Safeguard bldg. transformer T1EC3-2	45 kW	yes	10	2	1	-	480	-	-	-	15	estimated	(5)
Alternate shutdown isolation transformer	7.5 kW	yes	10	1	1	-	480	-	-	-	7.5	(d)	(4) 36
UPS and DIST.RM.. A/C unit control	0.75 kW	yes	10	2	1	-	480	-	-	-	0.75	estimated	(4)
Technical support center isolation transformer	45 kW	yes	10	2	1	-	480	-	-	-	45	(d)	(4)

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR BLACKOUT CONDITIONS

Component***	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After Blackout (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(11) FLA	(11) P.F.	(10) Eff.	kW	Basis for hp Required*	Time To Stop	
BOP inverter (Class IE)	21 kW	yes	10	4	2	-	460	-	-	-	42	(a)	(4)	36
NSSS inverter (Class IE)	18.7 kw	yes	10	4	2	-	460	-	-	-	37.4	(a)	(4)	
CCWP	1000	yes	30	2	1	1000	6600	74.6	.918	-	782.9	(a)	(4)	36
CCWP area fan coil unit	3	yes	30	2	1	3	460	-	-	.8	2.8	(a)	(4)	
Diesel-generator jacket water standby pump	75	yes	30 (8)	2	1	75	460	-	-	.8	70	(a)	(5)	
Service water pump	900	yes	35	2	1	900	6600	72.4	.873	-	722.5	(a)	(4)	36
Auxiliary feedwater pump	700	yes	40	2	1	700	6600	54.5	.907	-	565.1	(a)	(4)	
Auxiliary feedwater pump area fan coil unit	3	yes	40	2	1	3	460	-	-	.8	2.8	(a)	(4)	
Containment air recirculation fans	125	yes	50	4	2	250	460	286	0.85	-	193.7	(a)	(4)	36
Neutron detector well fans	40	yes	50	2	1	40	460	-	-	.8	37.3	(a)	(4)	
Chilled water recirculation pumps (NNS)	125	yes	60	4 (2)	2	250	460	-	-	.85	219.4	(a)	(4)	36

TABLE 8.3-2
(Sheet 5 of 11)

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR BLACKOUT CONDITIONS

Component***	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After Blackout (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(11) FLA	(11) P.F.	(10) Eff.	kW	Basis for hp Required*	Time To Stop
Auxiliary building ventilation equipment room exhaust fan	40	yes	60	2 (2)	1	40	460	-	-	.8	37.3	(a)	(4)
HVAC chiller purge unit (NNS)	0.33	yes	60	4 (2)	1	0.33	460	-	-	.8	.31	(a)	(5)
HVAC centrifugal water chillers (NSR Class 3)	154	yes	75	2	1	154	460	181.6	0.904	-	130.8	(a)	(5)
HVAC chiller lube oil pump (NNS)	0.25	yes	(23) 85	4 (2)	2	0.50	460	-	-	.8	.47	(a)	(5)
UPS and DIST.RM. A/C unit compressor	50	yes	90	2	1	50	460	-	-	0.8	46.6	(a)	(4) 36
UPS and DIST.RM. A/C unit fan	10	yes	90	2	1	10	460	-	-	0.8	9.3	(a)	(4)
UPS and DIST.RM. A/C unit air-compressor	1	yes (10)	as required after 90 sec.	2	1	1	460	-	-	0.8	0.93	(a)	(5)
UPS and DIST.RM. booster return fan	20	yes	90	2	1	20	460	-	-	0.8	18.7	(a)	(5)
Control room air conditioning unit fan	50	yes	90	4 (2)	2	100	460	-	-	.8	93.3	(a)	(4)
Control room air humidifier	20 kW	yes	90	4 (2)	2	-	480	-	-	-	40	(a)	(4)

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR BLACKOUT CONDITIONS

Component***	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After Blackout (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(11) FLA	(11) P.F.	(10) Eff.	kW	Basis for hp Required*	Time To Stop
HVAC water chiller (NNS)	670	yes	(23) (24) 100	4 (2)	1	670	2300	148	.91	-	536.5	(a)	(5) 36
Control room electric heater	30 kW	yes	as required after 90 sec.	4 (2)	2 as required	-	480	-	-	-	(19)	(a)	(4)
Room adjacent to reactor makeup water storage tank heater	10 kW	yes	as required after 10 sec.	2	1	-	480	-	-	-	10	(a)	(5)
Room adjacent to cond. water storage tank heater	10 kW	yes	as required after 10 sec.	2	1	-	480	-	-	-	10	(a)	(5) 36
Room adjacent to refueling water storage tank heater	7.5 kW	yes	as required after 10 sec.	2	1	-	480	-	-	-	7.5	(a)	(5)
Control room emergency pressurization heater	20 kW	yes (9)	as required after 10 sec.	2 (2)	1	-	480	-	-	-	20	(a)	(5)
Control room A/C unit compressor	175	yes (9)	as required after 90 sec.	4 (2)	2	350	460	394	0.895	-	281	(a)	(4) 36
Control room makeup supply fan	2	no (3)	as required	2 (2)	1	2	460	-	-	.8	1.9	(a)	(4)
Control room exhaust fan	5	no (3)	as required	2 (2)	1	5	460	-	-	.8	4.7	(a)	(4)

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR BLACKOUT CONDITIONS

Component***	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After Blackout (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(11) FLA	(11) P.F.	(10) Eff.	kW	Basis for hp Required*	Time To Stop
Control room kitchen and toilet exhaust fan	1.5	no (3)	as required	2 (2)	1	1.5	460	-	-	.8	1.4	(a)	(4)
Service water screen wash pump	15	no (3)	as required	2	1	15	460	-	-	.8	14	(a)	(5)
Diesel-generator startup air compressor	30	yes (9)	as required after 10 sec.	4	1	30	460	-	-	.8	28	(a)	(5)
Boric acid transfer pump	15.5 kW	yes (9)	required after 10 sec.	2	1	-	480	-	-	-	15.5	(a)	(4)
Containment H2 analysis panel	1	no (3)	as required	2 (2)	1	1	460	-	-	.8	.93	(a)	(4)
Electrical area fan coil unit	5	yes	90	4	2	10	460	-	-	.8	9.3	(a)	(4)
Fire pump	350	yes (9)	as required after 90 sec.	1 (2) (12)	1	350	460	372	0.934	-	276.8	(a)	(4)
Instrument air compressor	100	yes (9)	as required after 90 sec.	1 (14) (17)	1	100	460	-	-	.85	87.8	(a)	(4)
Reactor makeup water pump	40	no (3)	as required after 105 sec.	1 (13)	1	40	460	-	-	.8	37.3	(a)	(5)

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR BLACKOUT CONDITIONS

Component***	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After Blackout (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(11) FLA	(11) P.F.	(10) Eff.	kW	Basis for hp Required*	Time To Stop
Jockey fire pump	7.5	yes (9)	as required after 120 sec.	1 (2)	1	7.5	460	-	-	-	(18)	(a)	(4)
Primary plant ventilation exhaust fans	60	no (3)	as required after 120 sec.	16 (2)	1	60	460	-	-	.8	56	(a)	(4)
Instrument air dryer reactivation heater	11.5 kW	yes (9)	as required after 10 sec.	2	1	-	480	-	-	-	11.5	(a)	(5)
Positive displacement charging pump	200	no (3)	as required after 120 sec.	1	1	200	460	-	-	-	(20)	(a)	(4)
Positive displacement charging pump fan coil unit	1	no (3) (21)	as required after 120 sec.	1	1	1	460	-	-	-	(20)	(a)	(5)
Primary plant ESF filtration unit heater	100 kW	no(3)(22)	as required after 130 sec.	2 (2)	1	-	480	-	-	-	100	(a)	(5)
Service water intake structure fans	3	no (3)	5 min.	8 (2)	4	12	460	-	-	.8	11.2	(a)	(4)
Spent fuel pool pump	250	no (3)	30 min.	2 (2)	1	250	460	287	0.881	-	201.5	(a)	(4)
Spent fuel pool area fan coil unit	3	no (3)	30 min.	2 (2)	1	3	460	-	-	.8	2.8	(a)	(4)

36

36

TABLE 8.3-2
(Sheet 9 of 11)

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR BLACKOUT CONDITIONS

Component***	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After Blackout (sec) (1)	Number Installed Per Unit	Number Required	hp	Volts	(11) FLA	(11) P.F.	([illegible]) Eff.	kW	Basis for hp Required*	Time To Stop	
Pressurizer heaters	485 kW	no (3)	as required after 60 min.	4	1	-	480	-	-	-	485	(a)	(5)	
RHR pump	450	no (3)	as required after 4 hrs.	2	1	420	6600	31.7	.93	-	337	(c)	(4)	36
RHR pump area fan coil unit	3	no (3)	as required after 4 hrs.	2	1	3	460	-	-	.8	2.8	(a)	(4)	
Total kW											7019			36

AMENDMENT 36
DECEMBER 10, 1982

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR BLACKOUT CONDITIONS

Total Load on Diesel Generator (kW) Train A	7019**
Total Load on Diesel Generator (kW) Train B	6686

36

*Basis for hp Required

***Where "Pump (Fan)" components are listed, it is understood that the pump (fan) motor is the actual electrical load component.

(a) Nameplate Rating
(b) Pump Runout Condition
(c) Estimated Flow-Pressure Condition or Bhp
(d) Maximum (current limiting) output

NOTES:

(1) Maximum time to close breaker including 10 sec for diesel to come up to speed and voltage; the delay times for automatic start are 10 sec less when offsite power is available.

(2) Equipment is shared between two units and number shown is for two units.

(3) Manual start when required

(4) Manually stopped

(5) Stops automatically with assigned diesel or pump, or temperature, or pressure, and so forth.

(6) Motor stops automatically when valve action is completed, or receives signal to stop (e.g., sump pump stops on low water level).

(7) Motor Service Factor of 1.15 meets this HP requirement.

(8) Starts automatically with assigned load or upon failure of engine-driven pump

(9) Starts automatically upon temperature, pressure, level switch signal, etc.

**Unit 2, Train A Diesel-Generator load will not exceed 6750 kW since the 350 hp electric fire pump is connected to Unit 1, Train A only. 36

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR BLACKOUT CONDITIONS

(10) The following assumptions were made for 460V motors in calculating total load on diesel generator:

a) Less than 100 HP
Power factor 0.8 and efficiency 80%

b) 100 HP and above
Per manufacturer's data or, if not available.
Power factor 0.9 and efficiency 85%

$$kW = \frac{HP.746}{Efficiency}$$

(11) Manufacturers data is used for all medium voltage motors in calculating total load on diesel generator. $kW = \sqrt{3}\ VICos\ \phi$

(12) Train A, unit 1 only.

(13) One additional component is supplied which is shared by train "B" of both units;

(14) Valves have short operating times and are not included in the total continuous load.

(15) Train "B" only. Unit 2 channels are on Train "A" only.

(16) Train "A" load. Train "B" load = 31.5 kW

(17) Train "A" only, units 1 & 2.

(18) Train "B" only.

(19) Not included in total continuous load since heater and full compressor demand do not occur simultaneously. Compressor load is included in total continuous load as worst case.

(20) Operates in lieu of the centrifugal charging pump (CCP). The CCP is included in the total continuous load as the worst case.

(21) Operates in association with positive displacement charging pump.

(22) Heater starts 10 sec. after fan starts on temperature signal.

(23) Permissive signal actual start may be delayed upon 60 sec. maximum depending upon interlocks

(24) Start signal from 85 sec. step. Additional 15 sec. delay due to equipment interlock logic.

NOTE: Centrifugal Changing Pump (CCP); Component Cooling Water Pump (CCWP); Residual Heat Removal (RHR); Balance of Plant (BOP); Nuclear Safety Related (NSR); Non Nuclear Safety Related (NNS), Full load ampere (FLA), power factor (PF), Efficiency (Eff.)

The associated fuel handling structures in Containment are divided by a
refueling gate into two areas: the refueling cavity and Containment
fuel transfer area. These areas are flooded only during plant shutdown
for refueling. The associated fuel handling structures in the Fuel 36
Building are divided by refueling gates into three areas: the fuel
transfer canal, the shipping cask pit, and the spent fuel pools. These
areas are kept full of water whenever spent fuel is being moved or
stored in them. The Containment fuel transfer area and the fuel
transfer canal are connected by a fuel transfer tube which is fitted
with a blind flange on the Containment end and a gate valve on the Fuel
Building end. The blind flange is in place except during refueling to
ensure Containment integrity. Fuel is carried through the tube on an
underwater transfer car.

Fuel is moved between the reactor vessel and the Containment fuel
transfer area by the refueling machine. A rod cluster control changing 22
fixture is located in the refueling canal for transferring control
elements from one fuel assembly to another. The FTS is used to move
fuel assemblies between the Containment Building and the Fuel Building.
After a fuel assembly is placed in the fuel container, the lifting arm
pivots the fuel assembly to the horizontal position for passage through
the fuel transfer tube. After the transfer car transports the fuel
assembly through the transfer tube, the lifting arm at that end of the
tube pivots the assembly to a vertical position so that the assembly
can be lifted out of the fuel container.

In the Fuel Building, fuel assemblies are moved about by the fuel handling bridge crane. When lifting fuel assemblies, the hoist uses a long-handled tool to ensure that sufficient radiation shielding is maintained. Initially, a shorter tool is used to handle new fuel assemblies, but the new fuel elevator is used to lower the assembly to a depth at which the fuel handling machine, using the long-handled tool, can place the new fuel assemblies into or out of the fuel container.

Decay heat, generated by the spent fuel assemblies in the spent fuel pools, is removed by the Spent Fuel Pool Cooling and Cleanup System. After a sufficient decay period, the spent fuel assemblies are removed from the fuel racks and loaded into a spent fuel shipping cask for removal from the site.

9.1.4.2.2 Refueling Procedure

The refueling operation follows a detailed procedure which provides safe and efficient refueling. Prior to initiating refueling operation, the RCS is borated and cooled down to refueling shutdown conditions as specified in the Technical Specifications. Criticality protection for refueling operations, including a requirement for daily checks of boron concentration, is also specified in the Technical Specifications. The following significant points are ensured by the refueling procedure:

1. The refueling water and the reactor coolant contain approximately 2000 parts per million (ppm) boron. This concentration, together with the the negative reactivity of control rods, is sufficient to keep the core approximately 10 percent delta k/k subcritical during the refueling operation. It is also sufficient to maintain the core subcritical in the unlikely event, that all of the rod cluster control assemblies (RCCA) were removed from the core.

2. The water level in the refueling cavity is high enough to keep the radiation levels within acceptable limits when the fuel assemblies are being removed from the core.

The refueling operation is divided into four major phases: preparation, reactor disassembly, fuel handling, and reactor assembly. A general description of a typical refueling operation through the phases is given as follows:

and pressures for laboratory analysis. The samples are drawn from the points designated in Table 9.3-4. These samples are routed to a central location where the components of the Process Sampling System are located. The fluid properties at the designated sample points are critical to proper functioning of the plant and require frequent testing, as detailed in Chapter 16.

Local sampling is performed where batch processing is required, i.e., Liquid Waste Processing System (LWPS), Gaseous Waste Processing System (GWPS), Boron Recycle System (BRS), and for plant effluent streams. In addition, local samples can be taken from the boric acid tanks, boric acid batching tank, boron injection tank, Refueling Water Storage Tank, and the chemical additive tank. In general, the local sampling points require less frequent testing and are therefore not routed to the Process Sampling System. Local and effluent sampling is discussed in Section 11.5.4.

This sampling system has no emergency function. The Post Accident Sampling System (PASS), as described in FSAR Section II.B.3, samples critical parameters during postulated accident conditions. During a 36
loss-of-coolant accident (LOCA), sample lines are isolated on both sides of the Containment boundary. However, the Containment Isolation Valves associated with the PASS have the capability to override the isolation signal. See Section 6.2.4 for sample line isolation details.

9.3.2.2 System Description

9.3.2.2.1 System Operation

Samples from the points designated in Table 9.3-4 are conveyed to the sampling room in stainless steel tubing. The line sizes are determined to meet analyzer and grab sample flow requirements. The distance between the sample point and Process Sampling System has been kept to a minimum to reduce possible plateout and precipitation.

Samples enter the sample conditioning panel which houses the sample coolers, pressure-reducing and pressure-regulating valves, and local pressure and temperature indicators. The requirements for conditioning of a sample are based on the design temperature and pressure at the point of sampling. After conditioning, sample temperature and pressure are $105^{o}F \pm 10^{o}F$ and 5 to 10 psig, respectively.

All the sample lines are provided with grab sample valves or sample vessels with quick disconnect fittings, as shown on the Primary Sampling System piping and instrumentation diagram (P&ID), Figure 9.3-4.

Automatic online analyzers for cation conductivity, pH, and sodium ion concentration are provided for the steam generator blowdown samples only. An inline radiation monitor is provided for the steam generator blowdown samples to indicate steam generator tube leakage.

The samples are segregated so that primary and secondary fluids are not mixed. The routing of each sample line is shown on the Primary Sampling System P&ID, Figure 9.3-4.

9.3.2.2.2 Components Description

1. Samples Coolers

   The samples coolers are of tube-in-shell design with pressure relief valves on the shells. The samples flow through the tube side and component cooling water flows through the shells. The sample cooler tubes are of stainless steel construction and the shells are carbon steel.

2. Pressure-Reducing and Pressure-Regulating Equipment

   The pressure-reducing and pressure-regulating equipment consists of a variable capillary device or a velocity-controlled pressure-reducing valve in series with a pressure-regulating the Auxiliary Building above elevation 810 ft 6 in. is drained by means of a gravity drainage system that ties into the drainage system for the southern half of the Auxiliary Building and is eventually routed to floor drain tank No.1. The portions of the northern half of the Auxiliary Building at elevation 790 ft 6 in. are drained into sumps. Sump Nos. 6 and 8 take drainage from the waste evaporator condensate tank and recycle evaporator feed pump areas, respectively. The sump pumps are aligned to discharge into the waste holdup tanks. Sump No. 2 drains a portion of the floor at elevation 790 ft 6 in. and discharges to the laundry and hot shower tank. Sump Nos. 1, 7, and 12 collect the remainder of the drainage from the lowest part of the northern half of the Auxiliary Building. The sump pumps are aligned to discharge to floor drain tank No. 1.

After completion of Unit 2, the floor drains routed to these sumps will be routed to the floor drain tank for Unit 2.

Boron recycle tanks and waste holdup tanks are located within watertight compartments to prevent leakage into other areas if a tank break occurs. Floor drains from these areas have a locked-closed, manually operated gate valve located outside each compartment. | 36

9.3.3.2.4 Turbine Building Floor Drains

In the Turbine Building, the bulk of the drainage is collected in the two Turbine Building sumps. The sump pumps are normally aligned to discharge to the evaporation ponds. However, in the extremely unlikely event of radioactive contamination in the sump discharge, release to the evaporation pond would tend to concentrate the contamination. Thus, the discharge is diverted from the evaporation pond into the circulating water discharge canal, and a leak of radioactive material and subsequent contamination of sump water is detected and alarmed in the Control Room. The radiation level of sump water can be monitored from the Control Room, and discharge can be terminated manually if sump discharge radioactivity becomes excessive. For information on radioactive leakage, see Section 11.2.

Other means of drainage in the Turbine Building are as follows:

1. Drains in the hot and cold labs drain by gravity to the floor drain tank for Unit 1.

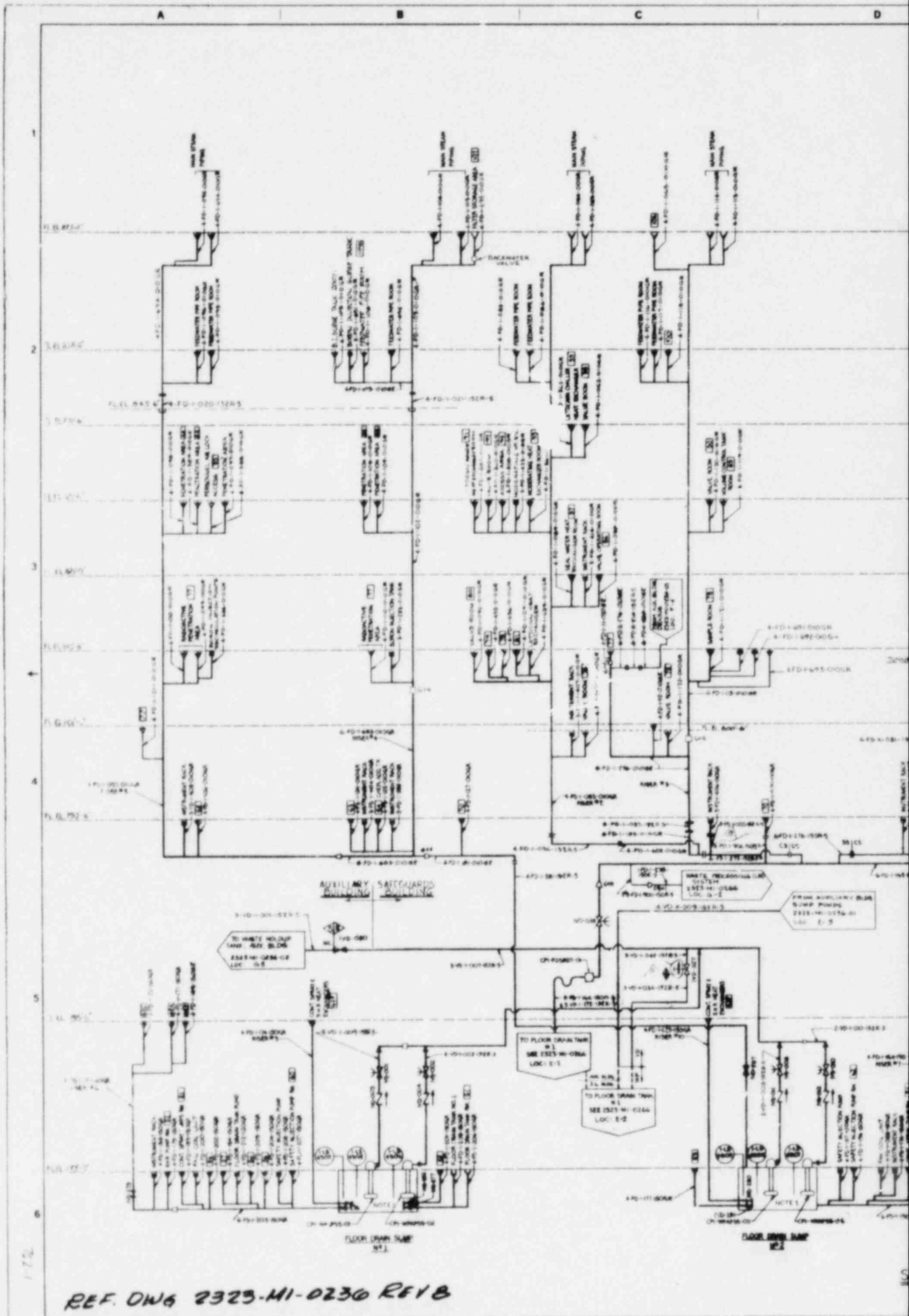
2. Drains in the personnel decontamination and laundry room drain by gravity to the laundry and hot shower drain tank.

3. Leakoffs for various components drain to the atmospheric drain tank. The tank is drained to the main condenser spray headers or main condenser gravity drains. Overflow from the tank is drained to Turbine Building sump No. 2.

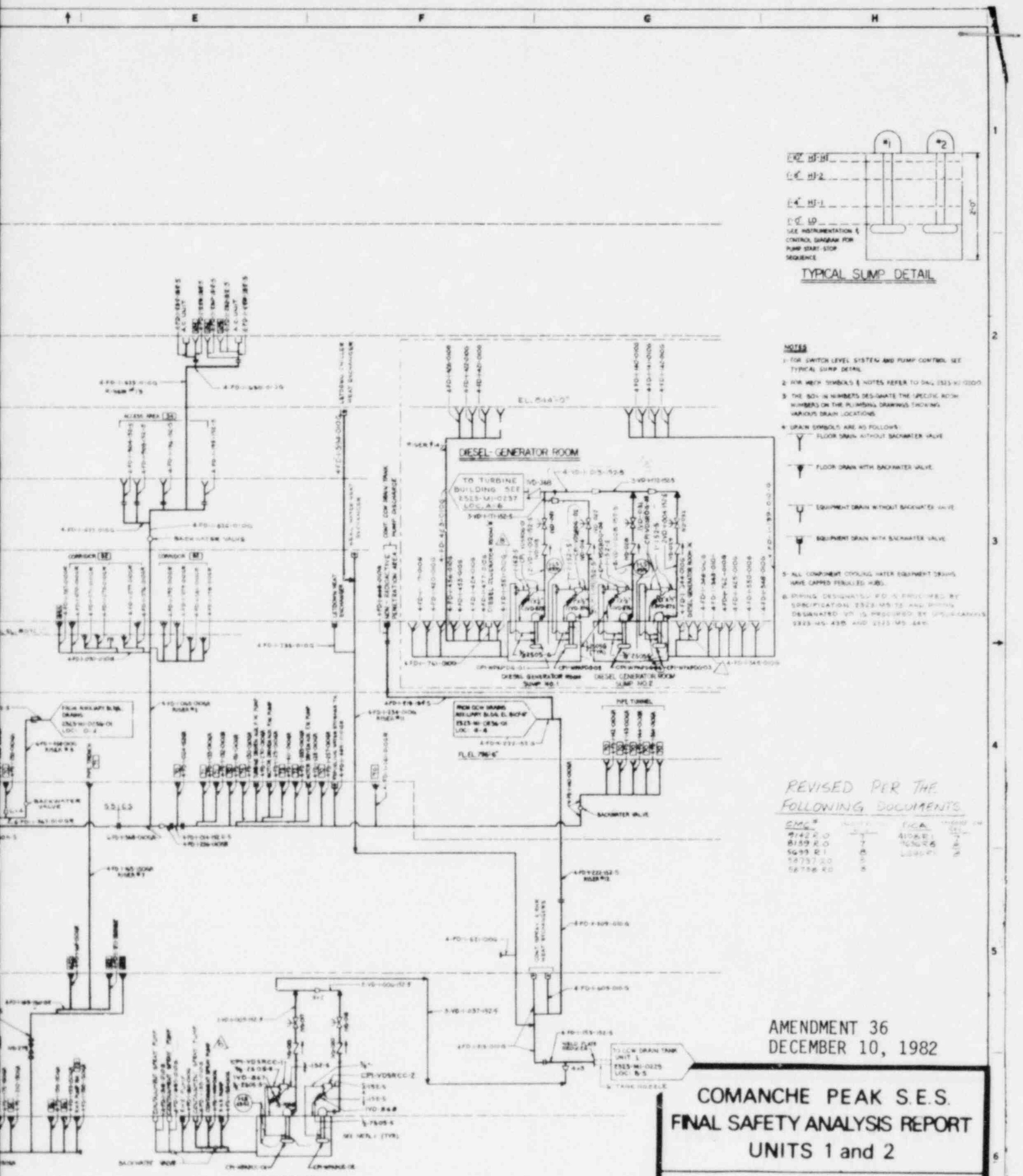
   All drainage piping and equipment in the Turbine Building are non-nuclear-safety-related and are not required to function after an accident or for safe reactor shutdown.

9.3.3.2.5 Fuel Building Floor Drains

All drainage from the Fuel Building is directed into one of two sumps. Drainage from the drum storage area is directed into the Fuel Building sump No. 2, which is located at elevation 810 ft 6 in. The sump pumps in sump No. 2 are aligned to discharge to the chemical drain tank, which is located on the north side of the auxiliary building. All other drainage from the Fuel Building is directed to sump No. 1, which is located at elevation 800 ft 2 in. The sump pumps in sump No. 1 are aligned to discharge to floor drain tank No. 1, which is located in the Safeguards Building at elevation 773 ft 0 in.



REF. DWG 2323-MI-0236 REV B



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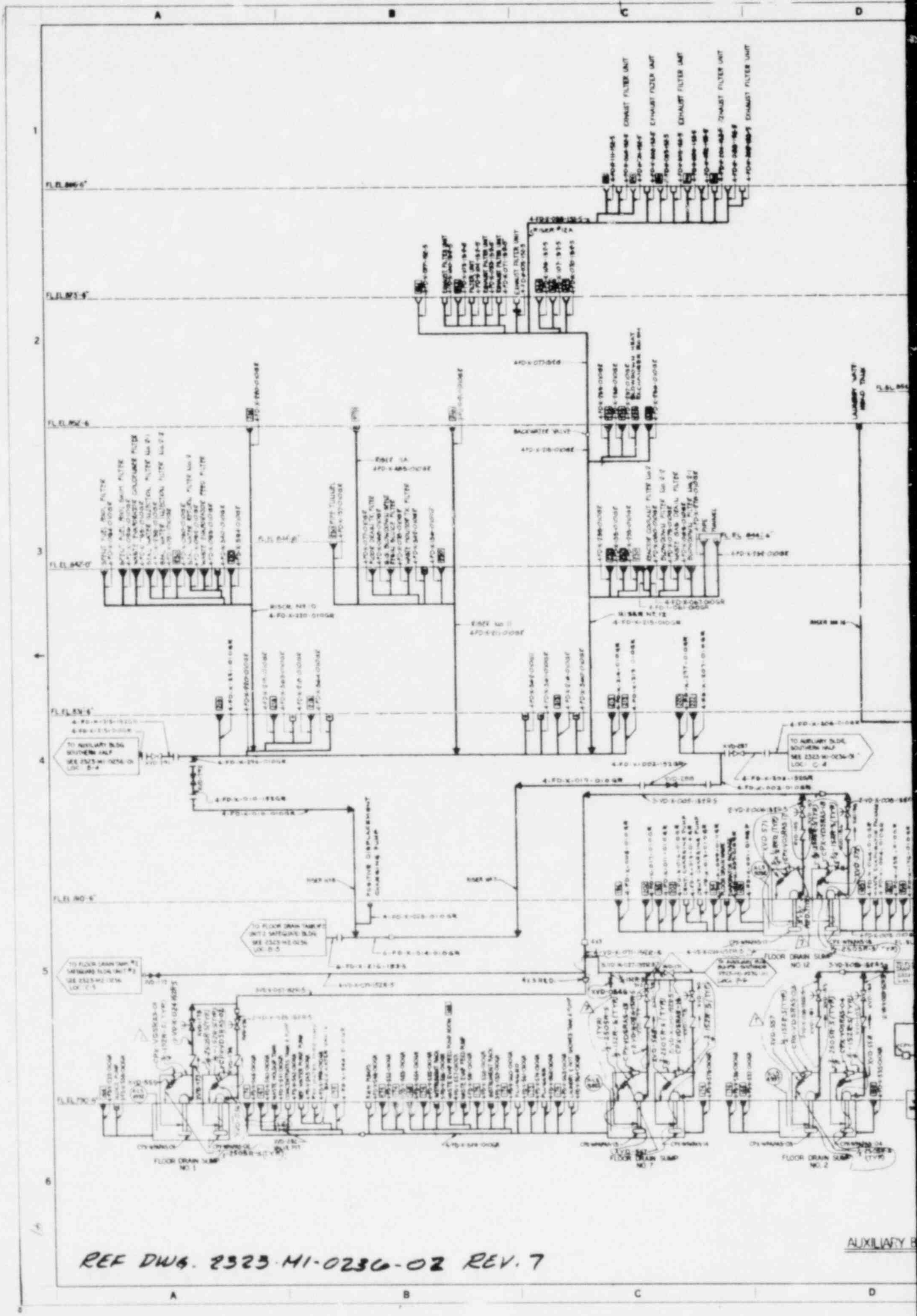
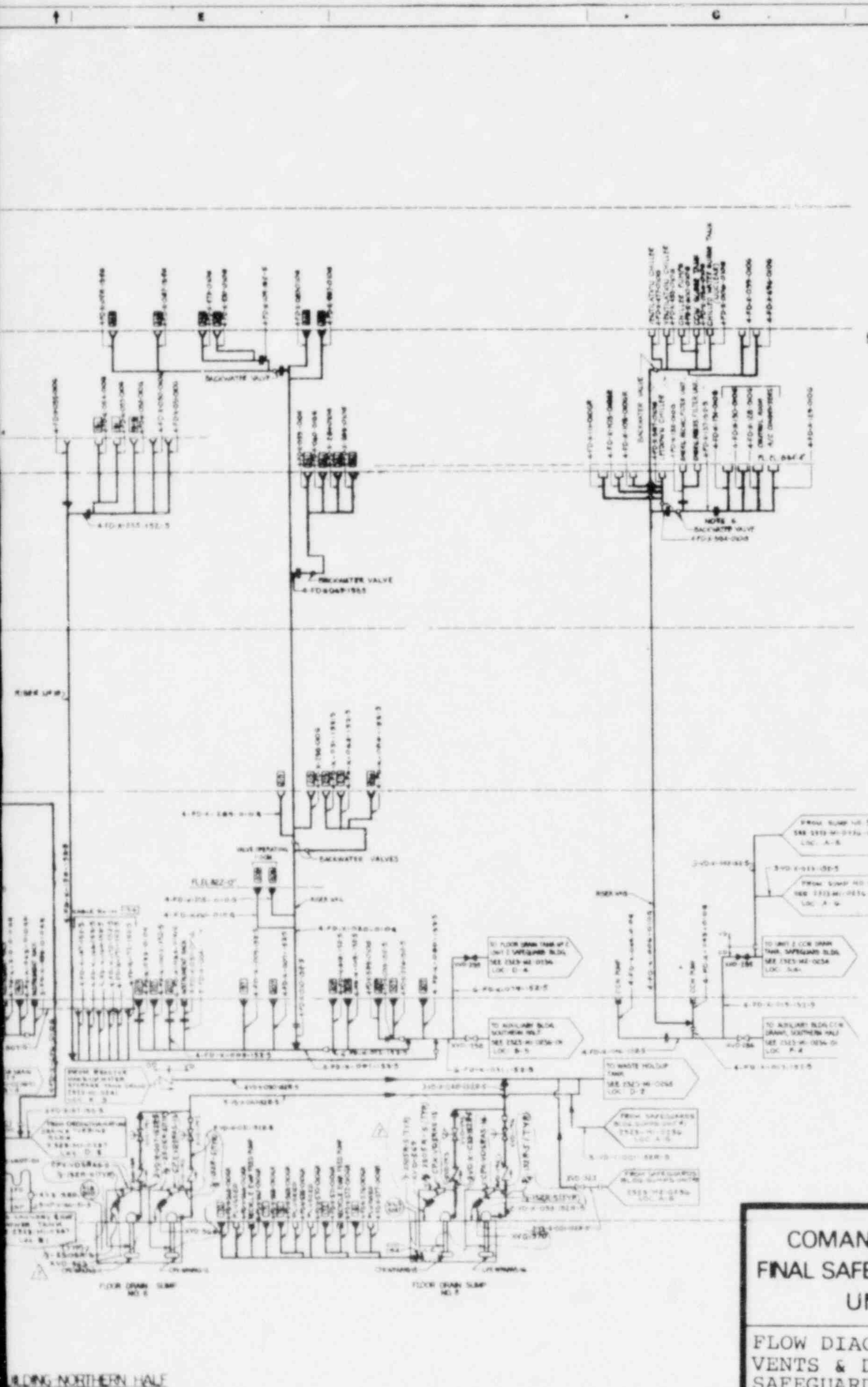
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

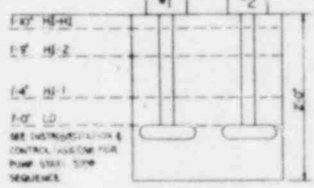
FLOW DIAGRAM
VENTS & DRAINS SYSTEM
SAFEGUARDS & AUX. BLDG

FIGURE 9.3-6



REF DWG. 2323-M1-0236-02 REV. 7





TYPICAL SUMP DETAIL

NOTES:

1. FOR SWITCH LEVEL SYSTEM & PUMP CONTROL SEE TYPICAL SUMP DETAIL.
2. THE BOXED IN NUMBERS DESIGNATE THE SPECIFIC ... NUMBERS ON THE PLUMBING DRAWINGS SHOWING VARIOUS DRAIN LOCATIONS.
3. FOR DRAIN SYMBOLS SEE 2323-M1-0234, NOTE 4.
4. ALL PIPING IS NON-NUCLEAR SAFETY CLASS.
5. FOR MECHANICAL SYMBOLS & NOTES SEE DWG. 2323-M1-0200.
6. ALL COMPONENT COOLING WATER EQUIPMENT DRAINS HAVE CAPPED FERRULED HUBS EXCEPT THE COMPONENT COOLING WATER PUMP EQUIPMENT DRAINS.
7. PIPING DESIGNATED FD IS PROCURED BY SPECIFICATION 2323-MS-75 AND PIPING DESIGNATED VD IS PROCURED BY SPECIFICATION 2323-MS-44D.

AMENDMENT 36
DECEMBER 10 1982

THE FOLLOWING DOCUMENTS HAVE BEEN INCORPORATED

COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

FLOW DIAGRAM
VENTS & DRAINS SYSTEM
SAFEGUARDS & AUX. BLDG.

FIGURE 9.3-7 (SH. 1 OF 4)

primary access/egress routes for the areas. This lighting is provided by sealed beam DC lights powered by 8-hour rated battery packs (except in the control room). The control room has Train A and Train B DC Emergency Lights powered from the Class 1E station batteries. DC Emergency Lighting is also provided for safe egress in other areas of the plant which include the Containment, Turbine Building, non-safety related areas of the Auxiliary Building and the Fuel Building. Power is supplied from the station batteries or individual 4 or 8-hour rated 31
battery packs. Q040.14

The DC Emergency Lights connected to the station batteries are normally deenerqized. These lights are activated in a given area by a supervisory relay that senses the loss of power to the AC lighting systems. The supervisory relay is normally energized by the AC lighting system and it operates a mechanically held contactor in the DC lighting panel.

The sealed beam DC Emergency Lights are connected to individual battery packs which are normally under a float charge from the AC lighting power supplies for their respective areas. If an AC lighting power supply is lost, the respective sealed beam DC lights are activated from their individual battery packs by their respective supervisory relays.

The lighting (AC and DC) in the Control Room, the Instrument Room, and 1
the essential control areas and primary access routes between them is Q040.14
arranged in a staggered pattern and alternately fed from the redundant trains. Where the engineered safety features (ESF) equipment is energized from a particular train, AC essential and DC emergency lighting from the same train is provided in that area and in the 8
primary access route to it. These practices ensure adequate lighting Q040.14
in these areas under any possible electrical single-failure condition.

9.5.3.2.2 Design Criteria

The AC Essential Lighting Systems are connected to Class 1E power
12 system buses through Class 1E circuit breakers in 480 volt Motor
Q040.14 Control Centers, Class 1E 480 volt-120/208 volt step-down transformers, and Class 1E lighting distribution panels. The power cables from the 480 volt Motor Control Center to the primary side of the 480 volt step-down transformer and from the secondary side of the step-down transformer to the lighting distribution panel are Class 1E. The cable raceway system for these power cables is designated Seismic Category I.
36 The feeder cables from the lighting distribution panels to the lighting fixture outlet boxes are Class 1E and the conduit system for these feeder cables is Seismic Category II. Operability of these feeder cables is adequately ensured through analysis that shows that the combined conduit stresses do not exceed 90% of their yield valve.

12 The AC essential lighting fixtures, fixture primary supports, lamps and
Q040.14 bulbs are classified non-nuclear safety grade (NNS).

Restraining cables, with associated hardware (clips, clamps, "U" bolts, clevises and turnbuckles) are used to prevent the AC essential lighting
31 fixtures and fixture primary supports from falling during a safe
Q040.14 shutdown earthquake. These items are supplied with certificates of compliance by the manufacturer that they meet applicable standards, including MIL-specification for the cable. The design and construction of the restraints meet the requirements of Seismic Category II criteria. They are installed in accordance with quality assurance procedures.

12 The DC Emergency Lighting Systems connected to the station battery have the following Class 1E components: station batteries, DC switchboard,
Q040.14 power cable from the DC switchboard to the DC lighting distribution panels and from the DC lighting distribution panels to the lighting fixture outlet boxes. The raceway system for the power cable is seismic Category I.

determined by applying appropriate factors for dilutions, dispersion, and decay between the point of discharge and the boundary.

2. Dose to Individuals

Except for occasional variation, as specified in 10 CFR Part 20, the permissible dose of exposure for an individual is as follows:

a. Restricted Area
1-1/4 rems per calendar quarter to the whole body

b. Unrestricted Area
0.5 rem per calender year to the whole body

An evaluation showing that the proposed systems are capable of controlling releases within the numerical design objectives of Appendix I of 10 CFR Part 50 may be found in Appendix 11A.

A cost-benefit analysis for the LWPS is not performed because the limitations imposed by the annex Appendix I of 10 CFR Part 50 are met.

11.2.1.3 Release Control Capability

The system is capable of controlling release and reducing doses. The subsystems have components such as filters, evaporators, demineralizers, and reverse osmosis equipment, all of which remove radioactivity to various degrees. The recycling capability of the system also ensures a complete control over the release.

The decontamination factors for the processing equipment, which were used to determine isotopic release quantities, are listed in Table 11.2-6.

Ample surge capacity of the system, and the low-load factor of the processing equipment permit the system to accommodate waste until failures can be fixed and normal plant operation resumed.

11.2.1.4 Seismic Design Classification and Quality Group Classification

The seismic design classification and quality group classification for
36 the LWPS components and structures are listed in Appendix 17A, Table 17A-1.

11.2.1.5 Special Features

The system has been designed and the equipment chosen to effect an appreciable reduction in maintenance downtime, leakage, gas release, cleaning, and so forth.

All the tanks provided have a sufficient capacity to absorb any abnormal surge load. They are equipped with high-level and low-level alarm switches to prevent overflowing and emptying.

A majority of the pumps are of the canned-rotor design with flanged connections so that a spare pump can be used to replace most pumps in the system. This simplifies the spare parts inventory and eventually reduces downtime. A provision for standby pumps, such as the one for the reactor coolant drain tank, reduces maintenance. The canned-rotor design minimizes overall liquid leakage and also minimizes the release of entrained radioactive gases in the leaking fluid to the plant building atmosphere.

Provisions are made in the reactor coolant drain tank to maintain a constant level by recirculation of a portion of the pump discharge back to the reactor coolant drain tank. Maintenance of a constant level

5. Baler

The baler is a commercially available assembly used in conjunction with standard 55-gal drums which receive the low radiation level, solid, compressible wastes. Compaction of the wastes is performed after an amount of waste sufficient to fill a drum has been accumulated.

11.4.2.6 Other Design Features

1. Process Control Program

Refer to ATCOR topical report No. 132A.

2. Free Liquid

4
Q320.8

Refer to ATCOR topical report No. 132A.

3. Overflow of Tanks

Refer to ATCOR topical report No. 132A.

4. Tanks Using Compressed Gases

Compressed gas is not directly used in any SWMS tank. However, the waste blending tank is vented to the plant ventilation system since bead resin slurry is transported to the waste blending tank from the spent resin storage tanks by means of nitrogen gas pressure. The volume and flow rate of the gas used for transferring one batch is estimated to be 1200 scf at 50 scfm, with two of batches per year. The expected radionuclide concentration of the vent gases will be negligible. The treatment provided (atmospheric cleanup system) for the vent gases is described in Section 9.4.

11.4.2.7 Packaging, Storage and Shipment

1. Packaging

4 Q320.8 The SWMS product is a burial package which is classified as a special form of packaged hazardous material in accordance with 49 CFR Part 173. A DOT permit for each container is not required, since containers are shipped as a group enclosed within a lead shielding or overpack for which a DOT permit is obtained. The contents in the containers are solid, therefore there is no danger of radioactive spills caused by dropping of containers.

2. Storage

Storage capacity for up to seventy-four 50 ft^3 containers is provided in the Drum Storage Area shown in figure 1.2-38.
26 Capacity for storage of at least one hundred forty-four 55-gallon drums is provided in Area 247 in the Fuel Building (see Figure 1.2-38).

Adequate shielding is supplied to reduce exposure to personnel
36 outside the drumming station to less than 25 mrem/hr. The locations of the solidification room and the drum storage area within the plant are shown on the general arrangement drawing,
4 Figure 1.2-38. Storage time is a variable and depends on
Q320.8 shipment schedules which the operating facility has contracted. Radioactive decay as a function of the storage interval, is considered to be minimal and credit for the decay is not taken during shielding calculations.

12.3.2.3 Shielding Material

The bulk of shielding material is ordinary concrete with a minimum
density of 141 lb/ft^3 after 28 days of curing. Commerical non shrink
grout and mortar having a minimum density of 130 pounds per cubic foot
may be utilized in congested areas when closing temporary construction 32
blockouts. In the Containment wall, however, concrete with a minimum
density of 136 lb/ft^3 is used. The density of reinforcement in
ordinary concrete shield walls and slabs is a minimum of 6.26 lb/ft^3.
Where space requirements prohibit the use of ordinary concrete, high
density concrete (220 lb/ft^3) with ilmenite and hematite aggregate is
used. Observation windows in the drumming area and filter transfer
area are of high-density glass with attenuation properties that provide
adequate shielding to reduce radiation levels to be within specified 36
dose limits (zones).

12.3.2.4 Shield Descriptions

12.3.2.4.1 Primary Shield

The primary shield is a large mass of reinforced concrete, which is a minimum of six feet thick throughout the full-length of the active reactor core and of irregular geometrical configuration. It surrounds the reactor vessel and extends upward from the Containment mat to form the walls of the refueling cavity. The primary shield is designed to perform the following services:

1. Reduce, in conjunction with the secondary shield, the radiation level from sources within the reactor vessel and allow limited access to the Containment during normal full power operation.

2. Limit the radiation level after shutdown from sources within the reactor vessel and permit limited access to the RCS equipment.

3. Limit neutron flux activation of component and structural materials over the life of the plant.

12.3.2.4.2 Secondary Shielding

The secondary shield is a reinforced concrete wall and slab structure that surrounds the RCS equipment, including pipes, pumps and steam generators. This shield, which is a minimum of 2 ft 9 in. thick, protects personnel from direct gamma radiation emanating from reactor coolant fission and activation products that are carried from the core by the reactor coolant. The secondary shield reduces the radiation from the reactor and doses from the N-16 activity in the primary loops. It supplements the primary shield to attenuate radiation to low levels; this permits limited access to the Containment for inspection during full-power operation and maintenance of equipment in certain areas during hot shutdown.

The secondary shield provides protection for refueling operations, inspection, repair and maintenance during refueling, and shutdown periods. During the transport of spent fuel assemblies, concrete shielding provides protection in the spaces close to the fuel transfer route and the reactor internals temporary storage area. The fuel transfer route is flooded with water which protects the airspace above
11 the fuel transfer path. For a more detailed discussion of fuel transfer tube shielding, see Section 12.3.2.4.7 below.

12.3.2.4.3 Containment Building

The Containment Building is a cylindrical reinforced concrete structure that completely surrounds the NSSS. At full-power operation, this shield attenuates the radiation level outside the primary-secondary shield complex to ensure that radiation levels outside the Containment correspond to zone designations of adjacent areas. The Containment wall and dome are 4-1/2 ft and 2-1/2 ft thick, respectively. These

INTEGRATED ENGINEERED SAFETY FEATURES TEST
TEST SUMMARY

OBJECTIVE

Demonstrate proper automatic alignment and operation of all safeguards systems upon a safety injection signal using both normal offsite and emergency power sources.

PREREQUISITES

1. The RCS is cold and drained down and the reactor vessel head and internals are removed.

2. The Refueling Water Storage tank has an adequate supply of demineralized water or boric acid at refueling concentration. 36

3. The containment spray pump discharge lines are manually isolated to prevent spraying into the Containment Building.

4. Actuation circuitry has been tested and is capable of actuating all equipment upon a manually initiated safety injection signal.

5. All systems required to actuate on a safety injection signal are operational and are aligned for normal operation.

TEST METHOD

1. With the SIS aligned for normal power operation, manually initiate a safety injection signal ("S" signal) and verify proper starting of all safeguards pumps and measure flowrates. Verify proper alignment of valves, including centrifugal charging pump suction and discharge valves, and pump mini-flow lines. 36

2. Verify proper actuation, alignment, and operation of all systems which are required to operate or change mode of operation upon receipt of a safety injection signal including the Station Service Water System, Component Cooling Water System, and required ventilation systems.

3. Repeat test with no offsite power supply available and verify proper operation of emergency diesel generators in conjunction with safety injection actuation including proper load sequencing, pump start times, and valve sequencing.

ACCEPTANCE CRITERIA

The safeguards systems actuate in the proper sequence, automatically align in the proper manner, and provide required flowrates to the RCS injection points in accordance with design requirements.

show a rapid pressure drop and a decrease in system water mass due to the break. The Safety Injection System is actuated on low pressurizer pressure within 1 minute after the break. The RCS pressure continues to drop and reaches saturation (1200 psi) in about 2 to 3 minutes. Due to the large thermal inertia of primary and secondary system, there has been no significant decrease in the RCS temperature below no-load by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2 percent Δk due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than 10 minutes after the break. The addition of borated (2,000 ppm) safety injection flow starting 1 minute 36
after the break is sufficient to ensure that the core remains subcritical during the cooldown.

Reactor Protection

As discussed in Section 15.4.8.1.1, reactor protection for a rod ejection is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are part of the Reactor Trip System. No single failure of the Reactor Trip System will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident.

Results

Cases are presented for both beginning and end-of-life at zero and full power.

1. Beginning of cycle, full power

   Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.20 percent Δk and 4.43, respectively.

The peak hot spot clad average temperature was 1971^{o}F. The peak hot spot fuel center temperature reached was 4764^{o}F.

2. Beginning of cycle, zero power

   For this condition, control bank D was assumed to be fully inserted and banks B and C were at their insertion limits. The worst ejected rod is located in control bank D and has a worth of 0.68 percent Δk and a hot channel factor of 13.0. The peak hot spot clad temperature reached 1751^{o}F, the fuel center temperature was 3029^{o}F.

3. End of cycle, full power

   Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.265 percent Δk and 5.41, respectively. This resulted in a peak clad temperature of 2039^{o}F. The peak hot spot fuel center temperature reached melting at 4800^{o}F. However, melting was restricted to less than 10 percent of the pellet.

4. End of cycle, zero power

   The ejected rod worth and hot channel factor for this case were obtained assuming control bank D to be fully inserted with banks C and B at their insertion limits. The results were 0.95 percent Δk and 17.44 respectively. The peak clad and fuel center temperatures were 2622 and 4376^{o}F. The Doppler weighting factor for this case is significantly higher than for the other cases due to the very large transient hot channel factor.

A summary of the cases presented above is given in Table 15.4-3. The nuclear power and hot spot fuel and clad temperature transients for the

WREFLOOD is also linked to the LOCTA-IV code in that thermal-hydraulic parameters from WREFLOOD are used by LOCTA-IV in its calculation of the fuel temperature. LOCTA-IV is used throughout the analysis of the LOCA transient to calculate the fuel clad temperature and metal-water reaction of the hottest rod in the core.

The analysis presented here was performed with the February 1978 version of the evaluation model which includes modifications delineated in References [15], [16], [17] and [17a].

The analysis in this section was performed with the upper head fluid 6
temperature equal to the reactor coolant system cold leg fluid temperature.

The upper head fluid temperature has been made equal to the cold leg temperature by increasing the upper head cooling flow [20].

Small Break LOCA Evaluation Model

The WFLASH program used in the analysis of the small break LOCA is an extension of the FLASH-4 code [13] developed at the Westinghouse Bettis Atomic Power Laboratory. The WFLASH program permits a detailed spatial representation of the RCS.

The RCS is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicityly with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy and momentum applied throughout the system. A detailed description of WFLASH is given in Reference [14].

The use of WFLASH in the analysis involves, among other things, the representation of the reactor core as a heated control volume with the associated bubble rise model to permit a transient mixture height calculation. the multi-node capability of the program enables an explicit and detailed spatial representation of various system components. In particular it enables a proper calculation of the behavior of the loop seal during a loss of coolant transient.

Clad thermal analyses are performed with the LOCTA-IV code [9] which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core and mixture height history from the WFLASH hydraulic calculations as input.

Schematic representations of the computer code interfaces are given in Figures 15.6-5 and 15.6-5.

The small break analysis was performed with the October, 1975 version
6 of the Westinghouse ECCS Evaluation Model (refer to References [9],
Q212.80 [14], [14a] and [14b]).

15.6.5.3.2 Input Parameters and Initial Conditions

Table 15.6-5 lists important input parameters and initial conditions used in the analysis.

The analysis presented in this section was performed with a reactor
vessel upper head temperature equal to the RCS cold leg temperature.
6 The effect of using the cold leg temperature in the reactor vessel
upper head is described in Reference [20]. In addition, the large
36 break analysis in this section utilized the upflow barrel-baffle
methodology described in Reference [25].

The bases used to select the numerical values that are input parameters
6 to the analysis have been conservatively determined from sensitivity
studies (refer to References [19] and [20]. In addition, the
requirements of Appendix K regarding specific model features were met
by selecting models which provide a significant overall conservatism in

the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time that the

14. Esposito, V. J., Kesavan, K. and Maul, B. A., "WFLASH, A FORTRAN-IV Computer Program for Simulation of Transients in a Multi-Loop PWR," WCAP-8200, Revision 2 (Proprietary) and WCAP-8261, Revision 1 (Non-Proprietary), July 1974.

14a. Skwarek, R., Johnson, W., Meyer, P., "Westinghouse Emergency Core Cooling System Small Break October 1975 Model", WCAP-8970 (Proprietary) and WCAP-8971 (Non-Proprietary), April, 1977.

14b. Letter NS-CE-1672, dated January 1978, C. Eicheldinger (Westinghouse) to J. F. Stolz (NRC).

15. Kelly, R.D., Thompson, C.M., et. al., "Westinghouse Emergency Core Cooling system Evaluation Model for Analyzing Large LOCA's During Operation with One Loop Out of Service for Plants Without Loop Isolation Valves," WCAP-9166, February 1978. 6

Q212.80

16. Eicheldinger, C., "Westinghouse ECCS Evaluation Model, February 1978 Version," WCAP-9220-P-A (Proprietary Version), WCAP-9221-P-A (Non-Proprietary Version), February 1978.

17. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, letter number NS-TMA-1981, November 1, 1978.

17a. Letter from T. M. Anderson of Westinghouse Electric Corporation to R. L. Tedesco of the Nuclear Regulatory Commission, letter number NS-TMA-2014, December 11, 1978.

18. "Westinghouse ECCS Evaluation Model Sensitivity Studies," WCAP-8341 (Proprietary) and WCAP-8342 (Non-Proprietary), July 1974.

19. Salvatori, R., "Westinghouse ECCS - Plant Sensitivity Studies," WCAP-8340 (Proprietary) and WCAP-8356 (Non-Proprietary), July 1974.

6 20. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, letter number NS-TMA-2030, January 1979.

21. DiNunno, J.J., Anderson, F. D., Baker, P.E. and Waterfield, R. L., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, U. S. Nuclear Regulatory Commission, Division of Licensing and Regulation.

22. Postma, A. K., and Pasedag, W. F., "A Review of Mathematical Models for Predicting Spray Removal of Fission Products in Reactor Containment Vessels," WASH-1329, U. S. Nuclear Regulatory Commission, Accident Analysis Branch.

23. "Meteorology and Atmoic Energy," U. S. Nuclear Regulatory Commission, Division of Technical Information, 1968.

24. Murphy, K. G. and Campe, Dr. K. M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," U. S. Nuclear Regulatory Commission.

36 25. Johnson, W. J. and Thompson, C.M., "Westinghouse Emergency Core Cooling System Evaluation Model - modified October 1975 version," WCAP-9168 (Proprietary) and WCAP-9169 (Non-Proprietary), September 1977.

and rise time both as a function of the volume of gas released. Having measured the characteristics of large scale gas releases the decontamination factor for iodine was obtained using the analytical expression from small scale iodine testing.

$$\text{Decontamination Factor} = 7.3\ e^{0.313\ t/d}$$

where

t = rise time (sec)
d = effective bubble diameter (cm)

The overall test results clearly indicate that iodine will be readily removed from the gas rising through the fuel storage area solution and that the efficiency of removal will depend on the volume of gas released instantaneously from the full void space

With consideration given to the total quantity of gas released from a full assembly i.e., 6.9 standard cubic feet (scf) for the 17 x 17 array the pool decontaminaction factor for iodine is indicated to be a minimum of 760 for the 26 foot depth. Thus a decontamination factor of 760 constitutes one parameter of the expected case. In the conservative case a much lower decontamination factor of 100 is selected to provide for deviation in the factors which control iodine absorption by the pool water. For realistic analysis a decontamination factor value of 500 is used which is a reduction to 66 percent of the expected value. This factor of conservatism is selected on the basis that more than 90 percent of the experimental data for iodine would support such a value. 36

15.7.4.3 Environmental Consequences

15.7.4.3.1 Postulated Fuel Handling Accident Outside Containment

The analyses of a postulated fuel handling accident are performed as follows

1. Conservative analysis based on Regulatory Guide 1.25.

2. Realistic analysis

   The parameters used for each of these analyses are listed in Table 15.7-7.

The bases for the Regulatory Guide 1.25 evaluations are as follows:

1. The accident is assumed to occur 100 hr after plant shutdown.

2. All of the rods in one fuel assembly are ruptured.

3. The damaged assebmly is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full power operation at the end of core life immediately preceding shutdown. A radial peaking factor of 1.65 is used.

4. The maximum fuel rod pressurization if $\leq$ 1200 psig.

5. The minimum water depth between the top of the damaged fuel rods and the spent fuel pool surface is 23 ft.

6. All of the gap activity in the damaged rods is released to the spent fuel pool and consists of 10 percent of the total noble gases other than krypton-85, 30 percent of the krypton-85, and 10

percent of the total radioactive iodine in the rods at the time of the accident.

7. Noble gases released to the spent fuel pool are then immediately released at ground level to the environment.

8. The iodine gap inventory is composed of inorganic species (99.75 percent) and organic species (0.25 percent).

9. The overall decontamination factor for the spent fuel pool is 100.

10. Noble gases are not held up in the fuel pool water.

11. All iodine escaping from the pool is immediately exhausted at ground level to the environment over a 2 hr period through the building ventilation system. The iodine adsorber removal efficiency is assumed to be 95 percent for inorganic species and organic species.

12. Atmospheric diffusion conditions are assumed to be the 0 to 2 hr ground level case.

Based on foregoing model, the whole body dose and thyroid dose at the EAB are conservatively calculated to be 0.2 rem and 3.62 rem, respectively. 36
The doses from this accident are well within the limits defined in 10CFR100 (25 rem whole body and 300 rem thyroid EAB dose). Based on all but item 11 of the foregoing assumptions the whole body dose and thyroid dose at the EAB are conservatively calculated to be 22
0.213 Rem and 72.4 REM, respectively. The doses from this accident even without iodine removal by filtration are well within the limits defined by 10 CFR 100.

15.7.4.3.2 Postulated Fuel Handling Accident Inside Containment

The possibility of a fuel handling accident inside Containment during refueling is relatively small due to the many physical, administrative, and safety restrictions imposed on refueling operations. Nevertheless, for analytical purposes, consideration is given to one accident, a drop of a fuel assembly into the refueling cavity by the refueling machine inside Containment. The impact would result in breaching of the fuel rod cladding and release of a portion of the volatile fission gases from the damaged fuel rods to the refueling cavity.

The following conservative assumptions are based on Regulatory Guide 1.25 and inherent plant design parameters used to calculate the activity releases and offsite doses for the postulated fuel handling accident inside Containment.

1. The accident assumed to occur 100 hr following reactor shutdown for refueling.

2. All rods in one fuel assembly are ruptured.

3. The assembly damaged is assumed the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full power operation at the end of core life immediately preceding shutdown. A radial peaking factor of 1.65 is used.

4. All of the gap activities in the damaged rods is released and consists of the 10 percent of the total noble gases other than krypton-85, 30 percent of the krypton-85, and 10 percent of the total radioactive iodine in the rods at the time of the accident.

5. The iodine gap inventory is composed of inorganic species (99.75 percent) and organic species (0.25 percent).

TABLE 17A-1

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
32a. Post Accident Sample System								
Sample heat exchanger	NNS	Mfrs Stds	-	NONE	Note C	II.B.3	Note 13g	
Sample panel	NNS	Mfrs Stds	-	NONE	Note C	II.B.3	Note 13g	
Piping and Valve (Containment isolation portion)	2	ASME III	2	I	Note 26,A	9.3.2, II.B.3	Note 13	36
Supports for Class 2 piping	2	ASME III	2	I	Note 27,A	3.9B	Note 13c	
Supports for Class 5 piping	NNS	ANSI B31.1	-	II	Note 44,B	3.6B	Note 13e	
Isolation Valve Control Panel	1E	IEEE-323	-	I	Note 26,A	9.3.2, II.B.3	Note 13c	36
33. Fuel Handling Equipment								
Refueling machine	NNS	Mfrs Stds	-	NONE	Note 4,B	9.1	Note 16, 13g	33
Containment Fuel Handling Bridge Crane	NNS	CMAA 74	-	II	Note 26,B		Note 23, 24, 13g	
Fuel Handling Bridge Crane	NNS	CMAA 70	-	II	Note 4,9,B		Note 13a, 23, 24,	
Rod cluster control changing fixture	NNS	Mfrs Stds	-	NONE	Note 4, 28		Note 16, 13g	
Reactor vessel stud tensioner	NNS	Mfrs Stds	-	NONE	Note 5		Note 15, 13g	
Spent fuel handling tool	NNS	Mfrs Stds	-	II	Note 4,B		Note 13a, 10	
Fuel Transfer System:								
- Fuel Transfer Tube & Flange	2	ASME III	MC	I	Note 4,A		Note 13c - (also evaluated as part of containment structure)	33
- Conveyer System & Controls	NNS	Mfrs Stds	-	NONE	Note 4, 28		Note 10, 13g	
- Remainder of System	NNS	Mfrs Stds	-	NONE	Note 4, 28		Note 10, 13g	
Refueling gates	NNS	AISC	-	I	Note 27,A		Note 13	
Fuel transfer tube expansion joint	2	ASME III	MC	I	Note 26,A		Note 13c	
Stud hole plug handling fixture	NNS	Mfrs Stds	-	NONE	Note 4, 28		Note 10, 13g	
Stud hole plugs	NNS	Mfrs Stds	-	NONE	Note 4, 28		Note 10, 13g	
Lower internals storage stand	NNS	Mfrs Stds	-	NONE	Note 4, 28		Note 10, 13g	
Upper internals storage stand	NNS	Mfrs Stds	-	NONE	Note 4, 28		Note 10, 13g	

TABLE 17A-1

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Electrical equipment supports (associated with safety related equipment)	N/A	AISC Code	-	I	Note 27,A	8.3.1	Note 13c	
Motors (safety related)	1E	IEEE-323	-	I	Note 26,A	8.3.1	Note 13c, 13d	
Power cables (associated with safety related equipment)	1E	IEEE-323	-	N/A	Note 26,A	8.3.1	Note 13d	
Lighting Instrumentation and control cable (associated with safety related equipment)	1E	IEEE-323	-	N/A	Note 26,A	8.3.1	Note 13d	
Wire and Cable Raceway System								
Cable trays (1E)	1E	-	-	I	Note 26,A	8.3	Note 13d	
Cable trays (non-1E)	N/A	-	-	II	Note 26,B		Note 13d	
Conduit (1E)	1E	-	-	I	Note 49,A		Note 13	
Conduit (non-1E)	N/A	-	-	II	Note 49,B		Note 13	
Conduit for AC Essential Lighting feeder cable	N/A	-	-	II	Note 49,B	9.5.3.2.2	Note 13	36
Supports (1E)	1E	AISC	-	I	Note 32,A		Note 13e	
Supports (non-1E)	N/A	AISC	-	II	Note 32,B		Note 13e	
Supports of AC Essential Lighting feeder cable conduit	N/A	AISC	-	II	Note 32,B	9.5.3.2.2	Note 13e	36
480/120V Bypass transformers	1E	IEEE-323	-	I	Note 26,A	8.3.1	Note 13d	
480/208-120V isolation transformers	1E	IEEE-323	-	I	Note 26,A	8.3.1	Note 13d	
Priority Panels	N/A	-	-	I	Note 27,A	8.3.1		
Prefabricated Cables and Connectors (associated with Safety Related equipment)	1E	IEEE-323	-	I	Note 26,A	8.3.1	Note 13d	

I.D CONTROL ROOM DESIGN

OBJECTIVE:

"Improve the ability of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them."

- NUREG 0660, Pg. I.D-1

I.D.1 Control Room Design Review

Action Plan Requirements:

"In accordance with Task Action Plan I.D.1, Control Room Design Reviews (NUREG-0660), all licensees and applicants for operating licenses will be required to conduct a detailed control-room design review to identify and correct design deficiencies. This detailed control-room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor Regulation (NRR) requires that those applicants for operating licenses who are unable to complete this review prior to issuance of a license make preliminary assessments of their control rooms to identify significant human factors and instrumentation problems and establish a schedule approved by NRC for correcting deficiencies. These applicants will be required to complete the more detailed control room reviews on the same schedule as licensees with operating plants."

- NUREG 0737

CPSES Response

The preliminary assessment is complete and the report has been submitted to the NRC. The NRC onsite audit/review was conducted.

The comprehensive review of the Control Room has been completed in
36 accordance with NUREG-0700. A report was submitted to the NRC in
December, 1982, to document the findings of this review.

Implementation of required modifications should be completed by January, 1983 or fuel load, whichever is later.

I.D.2 Plant Safety Parameter Display Console

Action Plan Requirements:

"In accordance with Task Action Plan 1.D.2, Plant Safety Parameter Display Console (NUREG-0660), each applicant and licensee shall install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status."

- NUREG 0737

Also see NRC Letter dated August 1, 1980.

CPSES Response

A plant safety parameter display console will be provided for CPSES to assess plant safety status. This system will be designed when NUREG-0696 is issued in final form.

under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet criteria.

"A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (in less than 2 hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

"In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift)."

CPSES Response

II.B.3.1 General Description

FSAR Section 9.3.2 describes the existing process sampling system presently installed in CPSES. This system was designed for the capability of obtaining samples during normal operating conditions. During postulated accident conditions, radiation levels would be significantly increased. Therefore, additional sampling capability is being added to CPSES to specifically address the requirements for post accident sampling. This new sampling system will be installed and operational prior to operation above 5% power.

36 A review of the reactor coolant, containment atmoshpere and containment sump sampling systems and the radiological spectrum and chemical analysis facilities has been conducted. Plant modifications are being implemented to permit personnel to obtain and analyze samples within three hours after a decision is made to take a sample (without incurring an exposure to an individual in excess of five rem whole-body or 75 rem to the extremities). Provisions will be made to allow a chloride analysis within four days.

The Post-Accident Sampling System (PASS) is comprised of two independent subsystems; the Reactor Coolant PASS (RC PASS) and the Containment Air PASS (CA PASS). Figures II.B.3-1 and II.B.3-2 illustrate the basic system flow diagrams for the RC PASS and CA PASS respectively.

II.B.3.2 Design Criteria and Functional Requirements

The following design criteria are applied to the PASS:

1. Provisions are included to make it possible to obtain and analyze a sample without incurring a radiation dose to any individual in excess of GDC 19 (i.e., five rem wholebody or 75 rem to the extremities). To assure that these limits are met, facility design and procedures meet the following criteria:

   a. For shielding calculations, PASS source terms are for undiluted reactor coolant or containment atmosphere as delineated in Section II.B.2. These source terms exceed those resulting from a Regulatory Guide 1.4, Revision 2, release. 36

   b. Access to the sample station and the analyses facilities is through areas which are accessible in post-accident situations and which are provided with sufficient shielding to assure that the radiation dose criteria are met.

   c. Operations in the sample station, handling of highly radioactive samples from the sample station to the hot laboratory, and handling while working with the samples in the laboratory are such that radiation dose criteria are met. To this end sufficient shielding of personnel from 1) radioactive sample lines, 2) the samples themselves, and 3) other radioactive lines, tanks and equipment in the vicinity of the sampling station and hot laboratory is provided. Remote operation and the handling of small volumes of radioactive material also help to meet the dose criteria. Capabilities for remote handling and dilution of samples for analysis also are provided.

d. Portable survey instruments and personnel dosimeters are provided to permit rapid assessment of exposure rates and accumulated personnel exposure.

2. The RC PASS and CA PASS are totally self-contained, enclosed systems which do release sample fluid to the interior of the sample module during operation. Each sample module is connected to the Safeguards Building exhaust ventilation system and each has an exhaust fan which is actuated upon manual access to the sample module door (i.e., during removal of the sample). This exhaust passes through the Safeguards Building exhaust filtration package (HEPA and charcoal filters) and is monitored prior to release to the atmosphere.

36 3. a. All unused volumes of reactor coolant or sump water samples are flushed out of the RC PASS Sample Module and transported to the Reactor Coolant Drain Tank.

b. Internal drainage from the Reactor Coolant Sample Module spray system is pumped to the floor drain tank at Elevation 778' in the Safeguards Building.

c. All unused volumes of containment air samples are flushed out of the CA PASS sample module and are returned to the Containment at a different elevation than the sampling locations.

4. An area radiation monitor is included within each sample module to monitor local activity levels within the module.

5. A dedicated sample cooler is provided for the RC PASS function.

6. Post accident sampling does not require an isolated auxiliary system to be placed in operation in order to use either PASS subsystem.

7. In order to permit post-accident sampling, the capability to override the containment isolation signal for the appropriate sample isolation valves is provided.

8. PASS samples can be obtained from the following sample points in each unit.

   a. Reactor Coolant System hot leg No. 1 and hot leg No. 4 normal sample lines. 36

   b. Containment sump via a connection to the ECCS recirculation (RHR) system which can be used when in the recirculation mode of operation.

   c. Containment atmosphere from two different elevations (sample points) in the containment.

9. The containment penetrations and containment isolation valves for PASS sample lines are Class 2 and Seismic Category 1. Other portions of the PASS sample lines are Class 5.

10. Provisions are included for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing loss of sampling capability due to loose material in RCS or Containment, and for flow restriction to limit reactor coolant loss from a rupture of a sample line. Sample lines are as short as practicable to minimize the volume of sample fluid outside containment.

11. In order to assure that a representative sample reaches the PASS sample station, the Containment atmosphere sample lines are heat traced so that the temperature is maintained and condensation is minimized until the sample reaches the sample panel. Provisions are included to permit containment atmosphere sampling under both positive and negative containment pressures.

12. In order to minimize the amount of radioactivity outside containment in a post accident situation, the PASS will be capable of flushing all lines from the external containment isolation valve to the sample return point. Liquid sample lines are flushed with demineralized water to the reactor coolant drain tank inside containment. Gas sample lines are flushed back to the containment using nitrogen gas.

36 13. Electrical power to the non-safeguards components of the PASS are powered from a Class 1E isolation transformer connected to a Safety-Train A motor control center. This highly reliable non-Class 1-E, electrical system (backed-up by the Train A diesel generator) provides assurance that samples can be obtained within three hours of the decision to sample, assuming loss of offsite power. Additionally, indication of sample pressure, temperature, pH, conductivity and gross radioactivity can be obtained at the PASS Remote Operation Modules assuming a loss of off-site power (see Section II.B.3.3). Operation of tne plant hot laboratory and counting room requires a normal AC power. The Containment Isolation valves which are part of the PASS are controlled by a class 1E valve control panel, located in the Cable Spread Room. (Electrical and Control Building, elevation 807'- 0"). This location provides a low post-accident radiation exposure to the panel operator.

14. When plant modifications are completed and procedures implemented, testing will be conducted to demonstrate the capability to obtain and analyze a sample according to the design criteria listed above.

Onsite facilities (including PASS and hot laboratory facilities) and procedures are being developed which provide the capability to quantify the following.

1. Gross radioactivity and certain isotopes that are indicators of the degree of core damage (i.e., noble gases, iodines, cesiums and significant non-volatile isotopes).

2. Dissolved gases (i.e., H_2 and O_2), boron concentrations, chloride and pH of liquids. 36

3. Hydrogen and oxygen levels in the containment atmosphere. The CPSES design includes additional provisions for monitoring of hydrogen concentration via the Containment Hydrogen Monitoring System as described in FSAR Section 6.2.5.

II.B.3.3 PASS Equipment

As stated above, the CPSES PASS is comprised of two independent subsystems, the Reactor Coolant PASS (RC PASS) and the Containment Air (CA PASS). A discussion of the major components of each of these subsystems is provided below.

II.B.3.3.1 Reactor Coolant PASS

The Reactor Coolant PASS (RC PASS) will be used to obtain micro-volume samples of undiluted reactor coolant, reactor coolant off-gas, and Containment sump water. The Containment sump sample will be taken from the RHR system via ECCS recirculation. The RC PASS, as shown in Figure II.B.3-1, includes the following major components:

- RC PASS Sample Module
- RC PASS Remote Operating Module
- RC PASS Flush and Diversion Manifold
- RC PASS Auxiliary Module
- RC PASS Flush Module
- RC PASS Sample Cooler

36 The RC PASS Sample Module is located in the primary sample room, it contains all the equipment required to perform the sample acquisition process. This process is remotely controlled at the RC PASS Remote Operating Module. The sample module traps small volumes of undiluted reactor coolant, reactor coolant off-gas, and Containment sump water in two shielded, pressure vessels. One vessel traps a liquid sample and the other traps an off-gas sample.

The actual sample removal is a manual task, accomplished by the use of pressure lock, micro-volume sygriges. These syringes are inserted into the pressure vessels through needle ports in the shields. The samples contained in the syringes are either immediately injected into a septum bottle (contained in a cart mounted, shielded cask) or are placed into a shielded syringe cask, also mounted on a transport cart. The micro-volume samples acquired by the above process supply sufficient sample volume to perform the required radiological analyses and the boron analysis above 1000 ppm.

The chloride analysis requires the acquisition of a larger volume than the above analyses. However, the chloride sample is not required until four days after the accident, when radiation levels in the primary coolant are significantly lower. To obtain the necessary volume of undiluted reactor coolant for this analysis a 5ml grab sample can be acquired in the RC Sample Module by operations at the RC PASS Remote Operating Module. This sample is trapped in a lead "pig" with nominal 2" lead shielding. The pig is manually disconnected from the sample module (after flushing) and is loaded into a shielded transport cart for transport to the hot laboratory. The pig is placed in the hot cell where the sample is removed from the pig and prepared for analysis. This grab sample capability, coupled with the hot cell, also provides flexibility in acquiring a larger volume sample for other analyses, such as alternate methods of pH, conductivity, and boron analysis, as necessary.

For the pH and conductivity analyses, the Sample Module contains in-line instrumentation with remote read-out at the Remote Operating Module. In addition, the Sample Module contains temperature and pressure sensing instrumentation necessary to permit an analytical determination of the quantity of dissolved gas contained in a reactor coolant sample, and it has an emergency spray system to aid in the removal of radioactive contamination inside the sample module, should a leak occur in the closed system. Indication of such a failure can be determined from the radiation monitor located inside the sample module. This monitor has a remote read-out in the RC PASS Remote Operating Module. 36

The RC PASS Remote Operating Module (ROM) contains the equipment necessary to control the acquisition of the required samples in the sample module. The ROM contains the display units for the instrumentation in the sample module. The ROM is located in the Turbine Building Switchgear Room; an area where Post-Accident radiation levels are expected to be minimal.

The RC PASS Flush and Diversion Manifold (F&DM) is located in close proximity to the Containment penetration of the reactor coolant sample line. The F&DM is remotely operated from the RC PASS Auxiliary Module. The operation of the F&DM permits the flushing of the RC PASS sample path from the inlet Containment isolation valve to the Reactor Coolant Drain Tank, (in Containment). The source of demineralized water for the F&DM is the RC PASS Flush Module.

The RC PASS Auxiliary Module is located immediately adjacent to the RC PASS ROM. This module houses the valve controls and associated equipment for the F&DM as well as for the Sample Module's internal spray system. The Auxiliary Module also contains digital displays for the inlet flow rate, exit temperature and exit pressure for the Component Cooling Water entering and exiting the PASS Sample Cooler.

36 The RCS PASS Flush Module is also located in close proximity to the ROM. The Flush Module is primarily comprised of a 30 gallon stainless steel tank (with an open connection to the plant's demineralized water system) and a positive displacement pump capable of pumping 0.5 gpm of demineralized water to the F&DM. The primary functions of this device are to provide a source of flush water of sufficient pressure and to eliminate a direct connection between the plant Demineralized Water System and the Reactor Coolant System.

The RC PASS Sample Cooler reduces the temperature and pressure of the inlet liquid samples prior to entry into the Sample Module. The sample cooler is located in the sample conditioning rack, which is in a room adjacent to the Sample Module location.

II.B.3.3.2 Containment Air PASS

The Containment Air PASS (CA PASS) will be used to obtain micro-volume samples of the containment atmosphere. The CA PASS, as shown in Figure II.B.3-2, includes the following major components:

- CA PASS Sample Module
- CA PASS Remote Operating Module
- CA PASS Sample Diversion Valve

The CA PASS Sample Module is also located in the sample room, adjacent to the RC PASS Sample Module, and functions in a similar manner. The CA PASS Sample Module also contains particulate and iodine filter cartridges for radiological analysis. The CA PASS Sample Module is remotely controlled at the CA PASS Remote Operating Module.

The CA PASS Remote Operating Module (ROM) is located adjacent to the RC PASS RCM: it contains the equipment necessary to control the acquisition of the containment air sample in the CA PASS Sample Module. The CA PASS also contains the display units for the flow meter, temperature, pressure and radiation detection instrumentation located in the CA PASS sample module.

The CA PASS Sample Diversion Valve (SDV) is a three-way motor operated valve that controls the sampling and flushing processes. Nitrogen is used for the flushing gas; and it reaches the SDV from a connection with the nitrogen system contained in the ROM. The SDV is located near the containment penetrations for the two containment air sample intake lines. 36

II.B.3.4 Sample Analyses

After the post accident samples are obtained by the sample modules, and are placed in a shielded cask, the cask is transported via a transport cart to the hot laboratory for analyses. The laboratory will be equipped with a shielded hot cell for sample handling, sample preparation and various analyses. The hot cell is constructed so that the sample can be transferred from the shielded sample cask to inside the hot cell without direct exposure to the operator.

The sample will be handled by the master-slave manipulators inside the hot cell. Design features of the hot cell include lead-glass viewing window, interior illumination and ventilation so that radioactive samples can be safely handled. The hot cell shall have a remote controlled pipetter for removing measured quantities of sample and dispensing for analysis. Radiological analyses of the samples will be performed in the Counting Room (adjacent to the Hot Laboratory). As shown in Figure II.B.2-12, the post-accident radiation analysis shows that the maximum radiation level to be expected in the plant counting room is 15 mRem/hr. This is on the order of 1500 times the normal background. High activity samples such as post-accident coolant and containment air samples are expected to yield instrument count rates 10 times greater than that resulting from the elevated background. Therefore, the results of the counting analyses should have an accuracy within a factor of two. However, samples having lower activity and requiring the analyses of specific isotopes that may have excessive background interference in the plant counting room can be transferred to the Emergency Operations Facility counting room for analyses, where the background radiation level should be several orders lower than the plant counting room. The PASS will be used to acquire routine samples, such that it will be used at least once every two weeks. The routine use of the PASS will assure its operability and will also assure proper operator training and familiarity with the system.

36 Table II.B.3-1 lists the chemical and radiological analyses capabilities that will be utilized to perform the required post accident sample analyses. A discussion of these sample analyses is provided below.

II.B.3.4.1 Gross Activity

A 50 ul sample (0.5 uCi to 500 mCi) taken with a micro-syringe at the sampling port of the Post Accident Sampling System (PASS) will be placed in a shielded container and transferred to the hot cell. In the hot cell a gross estimate of the sample activity will be made by a radiation measuring device such as an ion chamber. The sample will then be diluted in an analytically acceptable manner to a factor that will allow conveniently placing approximately 500,000 - 1,000,000 dpm of activity in a counting planchet for evaporation. After evaporation of the sample in the planchet it will be transferred to a calibrated proportional counter in the counting room. With proper care this procedure can be accomplished in such a manner that a maximum error of 10% would be introduced. For post-accident samples, sensitivity would not be of concern.

II.B.3.4.2 Gamma Spectrum (Liquid)

A volume of diluted sample (prepared for the gross activity sample) with an activity of approximately 500,000 to 10,000,000 dpm will be placed in a standard gamma counting geometry and transferred to a calibrated Germanium detector gamma spectroscopy system in the counting room. With proper care this procedure can be accomplished in such a manner that a maximum error of 10% would be introduced.

II.B.3.4.3 RCS OFF-Gas Gamma Analysis 36

A minimum volume (10 ul) of reactor coolant off-gas will be removed from the RC PASS Sample Module using a micro-syringe. Larger volumes may be taken, depending on the level of sample activity, to optimize the analysis time. The sample will be injected into a 6 ml vial for standard gamma analysis.

II.B.3.4.4 Boron Content

Two ml of a primary coolant sample (0.02 uCi) that has been diluted 1000:1, as described in the gross activity section, will be processed for boron concentration using ASTM Carminic acid method. With this method the error of analysis should not exceed 15%, in the 1000 ppm to 6000 ppm range.

For a sensivity range of 0.5 ppm to 6000 ppm, the following method can be used. One ml of undiluted reactor coolant sample (10 uCi to 10 Ci) that has been transferred to the hot cell will be used to synthesize tetrafluoroborate. The concentration of tetrafluoroborate will then be determined using a tetrafluoroborate specific ion electrode. With proper care this procedure can be accomplished in such a manner that a maximum error of 10% would be introduced. Due to the larger volume of sample required, this procedure can be implemented only when the 5 ml grab sample container can be used.

II.B.3.4.5 Chloride Content

A small volume (3 uCi to 3 Ci) of undiluted sample will be transferred to an ion chromatograph located in the hot cell. The analysis for Chloride using this instrument should result in a maximum error of 10% in the 0.05 ppm to 20 ppm range. The 5 ml graph sample container is also required to obtain the necessary volume.

36 II.B.3.4.6 Total Dissolved Gas

The analysis for total dissolved gas is performed analytically by use of the pressure and temperature instrumentation of the RC PASS, and the gas laws, since the volume of the gas sample loop is known.

As an optional analytic technique, a sample of reactor coolant gas can be removed from the PASS using a shielded syringe. The sample can then be analyzed for hydrogen gas using standard gas chromatographic techniques. The analysis for hydrogen gas, by gas chromatography, should result in a maximum error of 10%, in the 0.5 to 2000 cc/kg (at STP) range.

The same technique (gas chromatography) that can be used for hydrogen gas analysis can be used for oxygen gas analysis. The maximum error for this analysis should also be 10%, in the 0.5 to 20 ppm range.

II.B.3.4.7 Conductivity

The preferred method of conductivity analysis is by means of an in-line conductivity cell. This cell (with a range of 0 to 20,000 micromho) is included in the RC PASS sample module; remote instrumentation on the RC PASS ROM will indicate the measured value. Additionally, manual, back-up capability is provided by a micro-conductivity cell or probe, which can be used in the hot cell. The required sample volume is provided by the 5 ml grab sample pig.

II.B.3.4.8 pH Measurement

The preferred method of pH analysis is by means of an in-line pH probe. This probe (with a range of 1 to 14 pH units) is included in the RC PASS Sample Module; remote instrumentation on the RC PASS ROM will indicate the measured value. Manual, back-up capability is provided by a micro pH electrode, which can be set up in the hot cell to measure the pH of undiluted reactor coolant samples (3 uCi to 3 Ci) with an accuracy of at least ± 0.2 pH units, in the range of 1 to 13 pH units. The required sample volume is provided by the 5 ml grab sample pig. 36

II.B.3.4.9 Containment Air Hydrogen Content

A sample of containment air will be removed from the CA PASS using a shielded syringe. The sample will be analyzed for hydrogen gas using standard gas chromatographic techniques. The analysis for hydrogen gas by gas chromatography should result in a maximum error of 10%, in the range of 0.1% to 10%.

II.B.3.4.10 Containment Air Oxygen Content

the same technique (gas chromatography) that is used for hydrogen gas analysis will be used for oxygen gas analysis. The maximum error for this analysis should also be 10%, in the range of 0.1% to 30%.

II.B.3.4.11 Containment Air Gamma Spectrum

The PASS will collect particulate activity and radioiodine isotopes on filters for analysis. The radiological measurements can be accomplished with a maximum error of 5-15%. By acquiring appropriate sized volumes of sample, the sensitivity of the analysis can exceed the MPC valves of 10CFR20.

II.B.3.4.12 Containment Air Noble Gas Activity

36 A minimum volume (10 ul) of Containment air will be removed from the CA PASS Sample Module using a micro-syringe. Larger volume samples may be taken, depending on the level of sample activity, to optimize the analysis time. The sample will be injected into a 6ml vial for standard gamma analysis.

II.B.4 TRAINING FOR MITIGATING CORE DAMAGE

Action Plan Requirements:

"A program is to be developed to ensure that all operating personnel are training in the use of installed plant systems to control or mitigate an accident in which the core is severly damaged. The training program should include the following topics."

"1. Incore Instrumentation
   a. Use of fixed or movable incore detectors to determine extent of core damage and geometry changes.
   b. Use of thermocouples in determining peak temperatures, methods for extended range readings; methods for direct readings at terminal junctions.
   c. Methods for calling up (printing) incore data from the plant computer."

"2. Excore Nuclear Instrumentation (NIS)

a. Use of NIS for determination of void formation; void location basis for NIS response as a function of core temperatures and density changes."

"3. Vital Instrumentation 36

a. Instrumentation response in an accident environment; failure sequence (time to failure, method of failure); indication reliability (actual vs indicated level).
b. Alternative methods for measuring flows, pressures, levels, and temperatures.
    1) Determination of pressurizer level if all level transmitters fail.
    2) Determination of letdow flow with a clogged filter (low flow).
    3) Determination of other Reactor Coolant System parameters if the primary method of measurement has failed."

"4. Primary Chemistry

a. Expected chemistry results with severe core damage; consequences of transferring small quantities of liquid outside containment; importance of using leak tight systems.
b. Expected isotopic breakdown for core damage; for clad damage.
c. Corrosion effects of extended immersion in primary water; time to failure."

"5. Radiation Monitoring

a. Response of Process and Area Monitors to severe damages; behavior of detectors when saturated; method for detecting radiation readings by direct measurement at detector output (overranged detector); expected accuracy of detectors at different locations; use of detectors to determine extent of core damage.

b. Methods of determining dose rate inside containment from measurements taken outside containment."

"6. Gas Generation

a. Methods of H_2 generation during an accident; other sources of gas (Xe, Ke); techniques for venting or disposal of non-condensibles.

b. H_2 flammability and explosive limit; sources of O_2 in containment or Reactor Coolant System.

- NRC Letter dated March 28, 1980, ENCLOSURE 3

36 "Complete the training of all operating personnel in the use of installed systems to monitor and control accidents in which the core may be severly damaged."

- NUREG 0694, Pg. 22

Also see NUREG 0737

CPSES Response

CPSES commits to develop training for mitigating core damage prior to fuel load and complete the training program prior to full-power operation. Shift Technical Advisors and all operations personnel who are license applicants will receive the training indicated in letter from H. R. Denton, NRC, to All Power Reactor Applicants and Licensees, Enclosure 3, dated March 28, 1980. Managers and technicians in the Instrumentation and Control Department and in the Chemistry and Health Physics Department will receive training commensurate with their responsibilities.

TABLE II.B.3-1

POST ACCIDENT SAMPLING ANALYSES

SAMPLE SOURCE	TYPE	RANGE	
Reactor Coolant	Gross Activity	10 uCi/ml - 10 Ci/ml	
	Gamma Spectrum	Identify and quantify isotopes	
	Boron Content	0.5 - 6000 ppm	
	Chloride	0.05 - 20 ppm	
	Dissolved Hydrogen	0.5 - 2000 cc/kg (STP)	36
	Dissolved Oxygen	0.5 - 20 ppm	
	pH	1 - 14	
	Conductivity	0 - 20,000 micromho	
Containment Sump	Same as Reactor Coolant, except for dissolved gases which are not required.		
Containment Atmosphere	Hydrogen	0.1 - 10 percent	36
	Oxygen	0.1 - 30 percent	
	Gamma Spectrum	Identify and quantify isotopes	

FIGURE II.B.3-1
([illegible]) 36

FIGURE II.B.3-2
(LATER) 36

(1) Noble Gas Monitor

The CPSES design will include wide range noble gas monitors for the plant vent stack which will detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident.

An adjacent-to-line monitor will be provided for each main steam line to monitor the concentration in steam that may be released to the environment by the safety or relief valves.

For description of these monitors, see Section 11.5.

(2) Iodine/Particulate Sampling

The wide range noble gas monitor discussed above provides the capability to sample the plant vent stacks as required. 27

For description, see Section 11.5

(3) Containment High Range Radiation Monitor

The redundant Category 1 monitors will be located in each Containment Building at Elevation 905'-6". To ensure valid data, these monitors will be located at least 90° apart and will not be located adjacent to process piping.

For further discussion, see Section 12.3.

(4) Containment Pressure

The CPSES design will include redundant wide range pressure indication (0 - 160 psig) on the Main Control Board meeting Regulatory Guide 1.97 Rev. 2 requirements. In addition this parameter will be provided as input to the SPDS high level display.

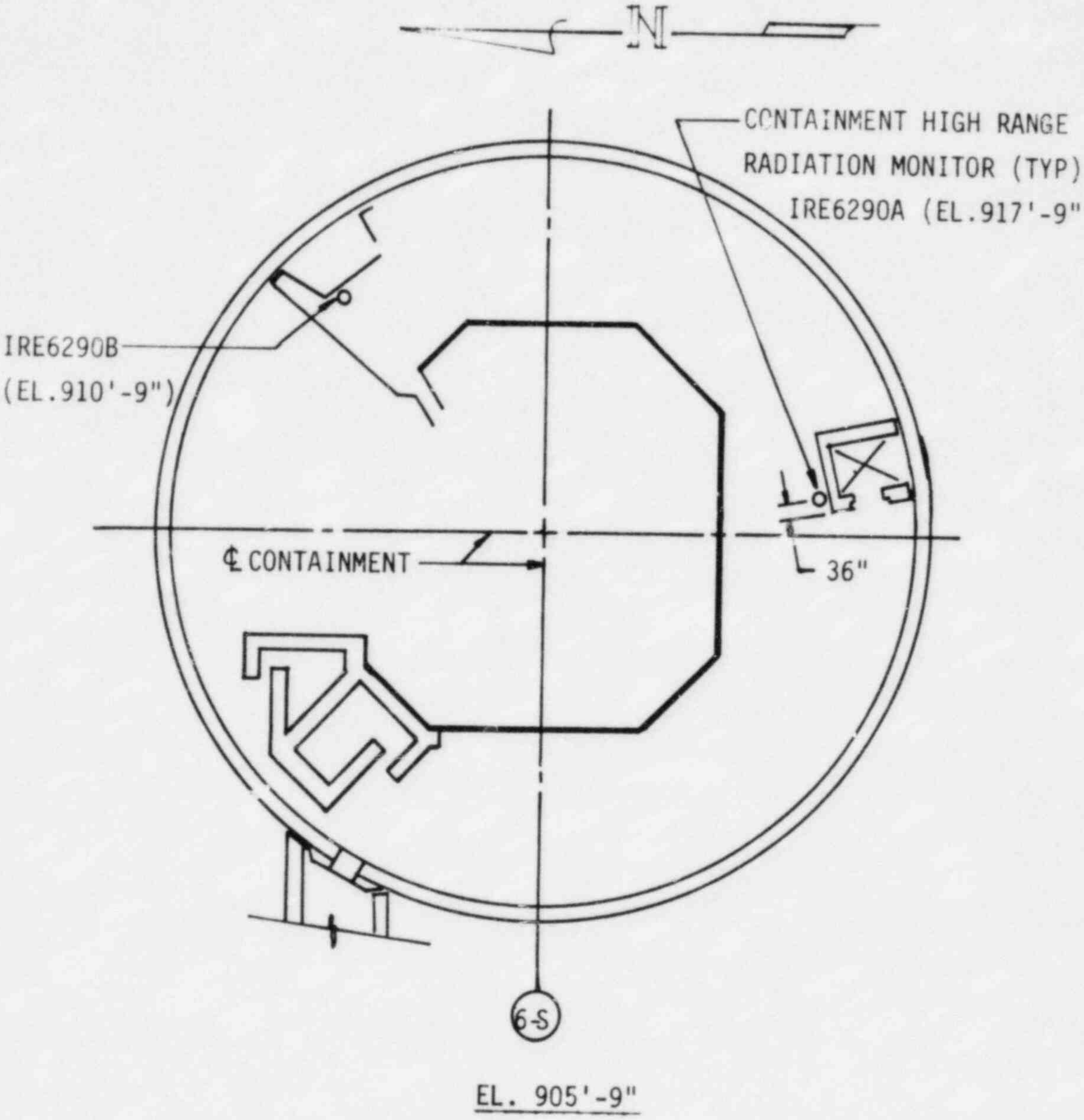
The present CPSES design includes four channels of narrow range pressure indication (-5 to + 60 psig) on the Main Control Board meeting Regulatory Guide 1.97 Rev. 2 requirements. In addition this parameter will be provided as input to the SPDS high level display.

(5) Containment Sump Level

CPSES design will include redundant wide range containment level
indication (0 - 160 inches) on the Main Control Board meeting
Regulatory Guide 1.97 Rev. 2 requirements. In addition this
22 parameter will be provided as input to the SPDS high level
display. These transmitters measure volume in excess of 600,000
gallons. CPSES design will also include normal sump level
indications (0 - 30 inches) on the Main Control Board meeting
Regulatory Guide 1.97 Rev. 2 requirements.

(6) Containment Hydrogen

The CPSES design will include H_2 concentration indication (0 - 10%) on the Main Control Board meeting Regulatory Guide 1.97 Rev. 2 requirements. In addition this parameter will be provided as input to the SPDS high level display.



NOTE: UNIT 1 MONITORS SHOWN
SAME FOR UNIT 2

REFERENCE:

1. FSAR FIG. 1.2-15

DECEMBER 10, 1982

COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

CONTAINMENT HIGH RANGE
RADIATION MONITOR LOCATIONS

FIGURE II.F - 1

III. EMERGENCY PREPAREDNESS AND RADIATION EFFECTS

III.A NRC AND LICENSEE PREPAREDNESS

III.A.1 IMPROVE LICENSEE PREPAREDNESS-SHORT TERM

OBJECTIVE:

"Promptly improve and upgrade licensee emergency preparedness by requiring improvements in facilities, plans, procedures, offsite support, technical assistance, equipment, and supplies required to adequately respond to and manage an accident."

- NUREG 0660, Pg. III.A.1-1

III.A.1.1 Upgrade Emergency Preparedness

Action Plan Requirements:

"Licensees will upgrade emergency preparedness in accordance with the requirements described in the NRC "Action Plan for Promptly Improving Emergency Preparedness" (SECY 79-450), which was distributed to all licensees during regional meetings in August 1979, and in accordance with subsequently issued acceptance criteria (NUREG-0654). These actions include:

(1) Preparing and submitting upgraded plans which satisfy the NRR supplemental acceptance criteria provided by the NRC emergency preparedness review teams, with special attention to the establishment of emergency action levels in accordance with NUREG-0610, "Basis for Emergency Action Levels for Nuclear Power Facilities."

(2) Implementing the short-term emergency planning recommendations of NUREG-0578.

(3) Establishing an onsite Technical Support Center, an onsite Operational Support Center, and a near-site Emergency Operations Facility.

(4) Establishing improved offsite radiological monitoring capability in accordance with the NRR/RAB technical position.

(5) Providing planning assistance to appropriate Federal, State, and local governments to assure that their emergency response roles are properly coordinated with the facility plan and that such plans satisfy the NRC acceptance criteria.

(6) Providing resources as necessary to State and local governments for implementing the emergency planning zone concept, in accordance with NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants."

(7) Participating in periodic joint exercises involving Federal, State, and local government emergency response organizations.

- NUREG 0660, Pg. III.A.1-9

"Provide an emergency response plan in substantial compliance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (which may be modified after May 13, 1980 based on public comments) except that only a description of and completion schedule for the means for providing prompt notification to the population (App. 3), the staffing for emergencies in addition to that already required (Table B.1), and an upgraded meteorological program (App. 2) need be provided. NRC will give substantial weight to FEMA findings on offsite plans in judging the adequacy against NUREG-0654. Perform an emergency response exercise to test the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations."

- NUREG 0694, Pg. 25

CPSES Response

1) The CPSES Emergency Plan has been written in accordance with the draft version of NUREG-0654 and was submitted to the NRC in October 1980 for review.

2) For implementation of NUREG-0578, 2.1.8.a, Post Accident Sampling, see FSAR Section II.B.3.

   For implementation of NUREG-0578, 2.1.8.b, Accident Monitoring Instrumentation, see FSAR Section II.F.1.

   For implementation of NUREG-0578, 2.1.8.c, Improved Inplant Iodine Instrumentation, see FSAR Section III.D.3.3.

3) See FSAR Section III.A.1.2.

4) CPSES is reviewing NUREG-0654, Rev. 1 requirements for offsite monitoring in order to develop a program for upgrading offsite monitoring capability.

5) Planning assistance will be provided to State and Federal agencies upon request. Local planning assistance is continually provided by a consultant retained by CPSES.

6) Resources to implement the State and Local emergency plans will

be made available to those agencies.

7) The State and Local emergency plans provide for periodic joint exercises with CPSES.

III.A.1.2 Upgrade Licensee Emergency Support

Action Plan Requirements:

"A separate technical support center shall be provided for use by plant management, technical, and engineering support personnel. In an emergency, this center shall be used for assessment of plant status and potential offsite impact in support of the control room command and control function. The center should also be used in conjunction with implementation of onsite and offsite emergency plans, including communications with an offsite emergency response center. Provide at the onsite technical support center the as-built drawings of general plant arrangements and piping, instrumentation, and electrical systems. Photographs of as-built system layouts and locations may be an acceptable method of satisfying some of these needs."

"Each operating nuclear power plant should establish and maintain a separate onsite operational support center outside the control room. In the event of an emergency, shift support personnel (e.g., auxiliary operators and technicians) other than those required and allowed in the control room shall report to this center for further orders and assignment."

- NUREG 0578, Pg. 13

Also see NRC Letter, dated April 25, 1980; August 1, 1980; and September 5, 1980.

CPSES Response

I. INTRODUCTION

CPSES will establish emergency response facilities, consisting of a Technical Support Center (TSC), an Operational Support Center (OSC), and an Emergency Operations Facility (EOF), prior to issuance of an operating license. These facilities will meet the guidelines of NUREG 0578, 0654, 0694, and also meet the intent of NUREG 0696 "Functional Criteria for Emergency Response Facilities," as stated in the descriptions below. 27

Further, the Emergency Response Facilities (ERF) will be upgraded by the implementation of an integrated ERF data acquisition and display system to be provided by a dual mini-computer system. This system will provide data to the TSC and EOF and will also provide the Safety Parameter Display System (SPDS) displays to the control room, TSC, and EOF. A description of this computer system and an implementation schedule is also provided in the discussion below.

The CPSES Emergency Plan has been revised to include the existence and function of these emergency response facilities; therefore, the following information is provided to supplement the information already contained in the CPSES Emergency Plan.

II. TECHNICAL SUPPORT CENTER

A. FUNCTION

The onsite Technical Support Center (TSC) will provide the following functions:

1. Provide plant management and technical support to plant operations personnel during emergency conditions.

2. Relieve the reactor operators of peripheral duties and communications not directly related to reactor system manipulations.
3. Prevent congestion in the control room.
4. Perform EOF functions for the Site Area Emergency Class and General Emergency Class until the EOF is functional.
5. Serve as primary communication center for the plant during the emergency.
6. Provide technical support during recovery operations following an emergency.

The TSC will be the Emergency Operations Work Area for designated technical, engineering and senior management personnel, and a small staff of NRC personnel. The TSC Manager will use the resources of the TSC to provide guidance and technical assistance to the control room during an emergency. The TSC will have facilities to support plant
27 management and technical personnel who will be assigned there during an emergency.

The TSC will be activated for the following classes of emergency action levels:

1. Alert
2. Site Area Emergency
3. General Emergency

B. LOCATION

The onsite TSC at CPSES is in the observation area above the control room, on elevation 840'6" of the Control Building (refer to Figure III.A.1.2-1). The TSC and control room are connected by a common stairwell, which can provide face-to-face interaction between control room personnel and the TSC Manager. TSC personnel will have access to information in the control room that is not available through the TSC Data System.

The stairwell between the TSC and control room is located outside of the control room environment, therefore, provisions will be made for safe and timely movement of personnel under emergency conditions. These provisions will 27
include considerations of effects of direct radiation and airborne radioactivity from inplace sources.

There will be no major security barriers between the TSC and control room, other than access stations at each facility.

C. STAFFING AND TRAINING

The staffing requirements of the TSC are in accordance with Figure 1.4 of the CPSES/Emergency Plan. Figure 1.4 staffing 36
applies to all emergency action levels for which the TSC is activated.

Upon activation of the TSC, designated personnel will report to the TSC and achieve full functional operation within 60 minutes. Activation of the TSC will ensure only designated operating personnel are in the control room and that needed technical support will be provided without obstructing plant manipulations or overcrowding the control room. 27

To maintain proficiency, the TSC staff will participate in TSC activation drills that will be conducted periodically in accordance with the CPSES/Emergency Plan. Operating procedures, and staff training in the use of the TSC Data System will contain guidance on limitations of instrument readings, including whether the information can be relied upon following such events as accidents resulting from earthquakes or the release of radiation.

D. SIZE

The TSC is housed in a complex of directly adjacent areas and is large enough to provide:

1. Work space for the personnel assigned to the TSC.
2. Space for the TSC Data Display Equipment.
3. Space for unhindered access to communication equipment by all TSC personnel.
4. Space for storage of and/or access to plant records and historical data.
5. A separate room for private NRC consultations.

The TSC working space is sized for 25 persons, including five NRC personnel. The TSC floor space is approximately 1500 sq. ft.

27

E. STRUCTURE

The TSC is located in a seismic category I structure, which is able to withstand the most adverse conditions expected, including earthquakes, high winds, and floods.

F. HABITABILITY

The Habitability System for the TSC is the same system that is provided for the control room. (Applicable criteria is as specified in General Design Criterion 19).

Potassium iodine will be provided for use by TSC and control room personnel. Detectors will be able to distinguish the presence or absence of radioiodines at concentrations as low as 10^{-7} microcuries/cc.

Equipment that protects personnel will be provided in the TSC for persons who must travel between TSC and control room or EOF under adverse radiological conditions. Protection equipment will be provided to allow TSC personnel to continue to function during the presence of low-level airborne radioactivity or radioactive surface contamination. Anti-contamination clothing and respiratory protective gear will be provided and properly maintained to assure availability during an emergency.

G. COMMUNICATIONS

The TSC will be the primary onsite communications center for CPSES during an emergency. Primary and backup voice
communications will be provided in the TSC to communicate 27
with the following:

1. Control Room
2. EOF
3. OSC
4. State and Local Emergency Operations Centers
5. NRC

The TSC Communication System will consist of the following:

1. Telephone (normal)
2. Telephone (leased line)
3. Sound Power System
4. Gaitronics (intercom/paging system)
5. Radio (portable)

The TSC will also include two designated telephones for NRC personnel.

27 Facsimile transmission capability between the EOF, the TSC, and the NRC Operations Center will be provided.

The communication system for the emergency facilities is
36 presented in CPSES/Emergency Plan, Section 4.0.

H. INSTRUMENTATION, DATA SYSTEM EQUIPMENT, AND POWER SUPPLIES

The TSC Data System is provided as part of the integrated Emergency Response Facility (ERF) Computer System. As shown in Figure III.A.1.2-2, the integrated ERF Computer System
27 will gather, store, and display data needed in the TSC to
analyze the plant conditions. The TSC Data System will perform its function independent of action in the control room and without degrading or interfering with control room and plant functions. TSC displays will be powered from a Non-1E battery, Uninterruptable Power Supply (UPS) System.

Signals to the integrated ERF Computer System which are received from sources providing signals to safety-related equipment or displays will be isolated to ensure that the TSC Systems will not degrade performance of the safety-related equipment or display.

Power supplied to the ERF computer will be provided from a Non-1E battery, UPS System to insure the continuity of TSC functions if temporary loss of primary TSC Power sources
36 occurs. Upon loss of all A/C power or a seismic event, the
ERF computer will not be functional, however, data acquisition remains available from other sources.

The design goal for the Data System reliability is to achieve an operational unavailability goal of 0.01 during all plant operating conditions above cold shutdown.

The plant process computer will not be used to provide data to the ERF Computer System.

The SPDS display equipment used in the TSC will not be seismically qualified; it will have the same design goal of meeting the TSC Data System, reliability, and performance criteria. The design of the TSC Data System equipment will incorporate human factors engineering with consideration for both operating and maintenance personnel.

I. TECHNICAL DATA AND DATA SYSTEM

The ERF Computer System will receive, store, process, and display information acquired from different areas of the plant as needed to perform the TSC function. The ERF Computer System will provide access to accurate and reliable information sufficient to determine: 27

1. Plant steady-state operating conditions prior to an accident.
2. Transient conditions producing the initiating event.
3. Plant systems dynamic behavior throughout the course of an accident.

The data set available to the TSC Data System will be complete enough to permit accurate assessment of an accident without interference with the control room emergency operation.

36 As a minimum, the CPSES variables as listed in FSAR Section
7.5 will be available for display in the TSC. In addition,
all sensor data and calculated variables used in data set for
SPDS, EOF, or for transmission to offsite locations will be
27 available for display. The accuracy of the data displayed
will be substantially the same as the accuracy of comparable
data displayed in the control room. The time resolution of
data acquisition will be sufficient to provide data without
36 loss of information during transient conditions where that
data is necessary for operator action. The time resolution
of each sensor signal will respond on the potential transient
behavior of the variable being measured.

Disk and tape storage and recall capability will be provided
for the TSC data set. Two hours of pre-event and 12 hours of
27 post event data will be recorded.

The sample frequency will be consistent with the use of the
data. Capacity to record two weeks of additional post event
data will be provided. Archival data storage will be
provided automatically, and retrieval will be accomplished
without interrupting the TSC data acquisition function. The
36 ERF displays (3 of which will be provided in the CPSES TSC)
if used for data retrieval, will not be available for real
time parameters, but can be returned to on line display
service very quickly.

A sufficient number of display and printout devices will be
27 provided in the TSC to allow all TSC personnel to perform
their assigned tasks. The TSC displays will include
alphanumeric and graphical representations of:

1. Plant system variables
2. In-plant radiological variables
3. Meterorological information
4. Offsite radiological information

Trend information and time history display capability will be available to TSC personnel. The TSC displays will be designed so that call up, manipulation, and presentation of data can easily be performed. The TSC data display will present information in easily understood formats.

The SPDS can also be displayed in the TSC.

J. RECORDS AVAILABILITY AND MANAGEMENT

The TSC will have a repository of plant records and procedures at the disposal of the TSC personnel to aid in their technical analysis and evaluation of emergency conditions. The following reference material will be provided in the TSC:

1. CPSES FSAR
2. Plant Technical Specifications 27
3. Operating Instructions, Both Normal and Emergency
4. Technical Manuals
5. As-Built Drawings

III. OPERATIONAL SUPPORT CENTER

A. FUNCTIONS

The Operational Support Center (OSC) will be an onsite area where CPSES Operations will support personnel will assemble in an emergency. The OSC will:

1. Provide a location where plant support can be coordinated during an emergency.
2. Restrict control room access to support personnel specifically requested by the shift supervisors.

36 Supervision of the OSC will be as specified in the CPSES/Emergency Plan, Section 1.2.1.14.

The OSC will be activated for the following classes of Emergency Action Levels:

27
1. Alert
2. Site Area Emergency
3. General Emergency

The Operational Support Center will be located in the plant, specifically outside of and on the south side of the Turbine
36 Building at elevation 810'. The OSC will provide approximately 960 sq. ft. for gathering of personnel and storage of emergency equipment (refer to Figure III.A.1.2-3).

B. HABITABILITY

The OSC will have the same habitability conditions as the turbine building. Provisions will be made for the evacuation
27 of OSC personnel to a safe area in the event of a hazardous radioactive release.

C. COMMUNICATION

The OSC will have reliable voice communication facilities to the following:

1. Control Room
2. TSC

The voice communication facilities will be composed of:

1. Telephone (Normal)

2. Gaitronics (intercom/paging system) 27

The communication system for the Emergency Facilities is presented in CPSES/Emergency Plan, Section 4.0. 36

IV. EMERGENCY OPERATIONS FACILITY

A. FUNCTION

The Emergency Operations Facility (EOF) is a TUGCO controlled and operated offsite support facility. The EOF has facilities for: 27

1. Management of overall CPSES response during emergency conditions
2. Coordination of radiological and environmental assessment
3. Determination of recommended public protective actions
4. Coordination of emergency response activities with federal, state, and local agencies

Facilities will be provided in the EOF for the acquisition, display, and evaluation of all radiological, meterological, and pertinent plant system data to determine off-site protective measures. The EOF will be activated for the following classes of Emergency Action Levels:

1. Site Area Emergency
2. General Emergency 36

The EOF is located in the Nuclear Operations Support Facility (NOSF) and is provided with special shielding and HVAC. Also located in the NOSF are other emergency related facilities. 27 These include a decontamination facility and an area for representatives of the news media to gather and be briefed.

When the EOF is activated, security protection will be upgraded to restrict access to those personnel assigned to the facility.

B. LOCATION

27 The EOF is a part of the Nuclear Operations Support Facility (NOSF) which is located 1.2 miles west of the plant site. The NOSF is a 55,000 square foot facility containing a plant specific simulator and training department including classrooms and laboratories, public information and visitor's center, engineering department for offsite technical support, and the EOF.

C. STRUCTURE

36 The EOF is a well engineered building meeting the Uniformed Building Code. It is designed for the expected life of the plant.

D. HABITABILITY

27 The EOF has special shielding and HVAC provisions for habitability. The walls and ceilings are approximately eight
36 (8) inches of concrete providing a protection factor of $\geq$15. The HVAC may be manually isolated providing a positive pressurization of the EOF and HEPA filtration of any makeup
27 air. The ventilation system and structure are not seismically qualified.

E. STAFFING AND TRAINING 27

The staffing requirements of the EOF are in accordance with Figure 1.4 of the CPSES/Emergency Plan. Figure 1.4 staffing 36 applies to all emergency conditions for which the EOF is activated.

Upon EOF activation, designated personnel will report directly to the EOF to achieve full functional operation within 60 minutes. Operating procedures and staff training in the use of data systems and instrumentation will contain guidance on the limitations of instrumentation, including whether the information can be relied upon following a 27 serious accident.

To maintain proficiency, the EOF staff will participate in an EOF activation drill that will be conducted periodically in accordance with the CPSES/Emergency Plan.

F. SIZE

The EOF is large enough to provide the following:

1. Working space for the personnel assigned to the EOF.
2. Space for the EOF data display equipment
3. Space for unhindered access to communication equipmer all EOF personnel
4. Space for storage of and/or access to plant records and historical data
5. A separate room for private NRC consultations

The EOF working space is sized for 35 persons, including 9 NRC personnel, and one person from FEMA. The EOF floor space is approximately 3200 sq. ft. for the area which has special shielding and HVAC. This 3200 sq. ft. is the area dedicated to dose assessment, communications and decision making. Refer to Figure III.A.1.2-4, "Emergency Operations Facility". Also located in the NOSF is a 2000 sq. ft. area dedicated to
27 decontamination of evacuated personnel and handling of contaminated and/or injured personnel. This area is equipped with decontamination facilities including showers with controlled drains.

Also located in the NOSF is an area for the news media and for briefings of the news media. This area will be outside of the controlled access area to prevent unauthorized persons from entering the EOF.

G. RADIOLOGICAL MONITORING

To ensure adequate radiological protection of EOF personnel,
36 portable radiation monitors will be provided in the EOF. These monitors will continuously indicate radiation dose rates and airborne radioactivity concentration inside the EOF during emergency conditions and will distinguish the presence or absence of radioiodines (at concentrations as low as 10^{-7} microcuries/cc). An early warning to EOF personnel will be provided in the event of adverse conditions.

27 H. COMMUNICATIONS

The EOF will have reliable voice communications facilities to the following:

1. TSC
2. Control Room
3. NRC
4. State and Local Emergency Operations Centers

The normal communication path between the EOF and control room will be through the TSC.

The voice communications between facilities will be composed of:

1. Telephone (Normal)
2. Telephone (Leased Line) 27
3. Two-Way Radio

The EOF will also include three designated telephones for NRC personnel.

Facsimile transmission capability between the EOF, the TSC, and the NRC Operations Center will be provided. The communication system for the emergency facilities is presented in CPSES/Emergency Plan, Section 4.0. 36

I. INSTRUMENTATION, DATA SYSTEM EQUIPMENT, AND POWERS SUPPLIES

The data display equipment located in the EOF is the same type equipment provided in the TSC, and will be a part of the integrated ERF Computer System. All comments made in II.H of this section apply, except for the display power supply. The displays provided for use in the EOF will be powered from a 27 commercial AC source. Should power fail, no data will be lost, as the displays can be regenerated when the power is restored.

J. TECHNICAL DATA AND DATA SYSTEM

The entire data set stored within the integrated ERF Computer System will be available for display to the EOF personnel. In addition to plant parameters, measurements of radiological, meteorological, and environmental conditions will also be available to the EOF personnel. This data set can be displayed at the EOF work stations and will be sufficient to perform assessments of the actual and potential onsite and offsite environmental consequences of an emergency condition.

The accuracy of data in the EOF will be the same as that in the TSC and control room displays which are all part of the integrated ERF Computer System.

All other comments in II.I of this section also apply to this
27 section. However, no computer logging of offsite radiological information is anticipated, as these measurements will be made by mobile monitoring teams and will be manually recorded.

No display capabilities will be provided for the news media.

The SPDS will be available for display in the EOF.

K. RECORDS AVAILABILITY AND MANAGEMENT

The EOF will have ready access to plant records, procedures, and emergency plans needed to excercise overall management of CPSES emergency response resources. The EOF records will include:

1. CPSES FSAR
2. Plant Technical Specifications
3. Operations Instructions, both Normal and Emergency
4. Technical Manuals
5. As-Built Drawings
6. Off-Site Population Distribution Data
7. Evacuation Plans
8. Employee Radiation Exposure History

V. SAFETY PARAMETER DISPLAY SYSTEM

A. FUNCTION

The purpose of the Safety Parameters Display System (SPDS) 27
will be to assist control room personnel in evaluating the safety status of the plant. The SPDS will provide a continuous high level graphical display of plant parameters or derived variables representative of the safety status of the plant.

As an operator aid, the SPDS serves to concentrate a minimum set of plant parameters from which the plant safety status can be assessed. More detailed plant information will be provided by several secondary displays.

Human factor engineering will be incorporated in the SPDS design to enhance the functional effectiveness of the control room personnel.

The SPDS will be operational during normal, as well as, emergency conditions except for loss of all A/C power or
seismic events. However, data acquisition will remain 36
available from other sources. The SPDS will be able to
display pertinent information concerning steady-state and 27
transient conditions.

B. LOCATION

SPDS displays will be located in the control room, TSC, and EOF. The SPDS location in the control room will be visible to both the control room operator and the Senior Reactor Operator.

C. SIZE

The SPDS will be compatible with the existing space in the control room area and will not interfere with normal movement or with full visual access to other control room operating systems or displays.

D. STAFFING

The SPDS will be of such design that no operating personnel,
27 in addition to the normal control room operating staff, are
required for its operation.

E. DISPLAY CONSIDERATIONS

Texas Utilities is a member of the Safety Assessment System committee to develop and implement the necessary displays to meet the requirements for the SPDS. The committee consists of Quadrex Corporation and eleven other utilities owning Westinghouse PWRs. The committee presented to the NRC staff a description of the software and displays being developed for the Safety Assessment System. This presentation was conducted at a meeting held on May 14, 1981 in Bethesda, Maryland. Subsequently, slides used in the presentation were submitted by letter to Mr. L. Beltracchi. The following discussion serves as a general description of the Safety Assessment System in relation to SPDS requirements.

1. General Considerations

The Safety Assessement System (SAS) meets the requirements of the Safety Parameter Display System (SPDS). This report describes that portion of the SAS which meets the SPDS requirements of NUREG-0696. It provides a centralized, flexible, computer-based data and display system to assist control room personnel in evaluating the safety status of the plant. This assistance is accomplished by providing the operator and also Emergency Response Facilities (ERFs) a high-level graphical display containing a minimum set of key plant parameters representative if the plant safety status. More detailed plant information is provided by several secondary displays. All graphical displays are presented to the control room operator on a high resolution multiple-color CRT.

All data displayed by the SAS is validated by comparing redundant sensors, checking the value against reasonable limits, calculating rates of change, and/or checking 27
temperature versus pressure curves.

All displays of the SAS have been carefully designed by persons with plant operating experience and evaluated against human factors design criteria. The concepts used in the SAS design will be verified using data recorded from power plant simulator. The intent of the SAS is to present to the control room personnel a few easily understandable displays which use color coding and pattern recognition techniques to indicate off-normal values. These displays are updated and validated on an essentially real-time basis.

The SAS will be operable during normal and abnormal plant operating conditions. The SAS will operate during all SPDS required modes of plant operation. The normal operation mode will encompass all plant conditions at or above normal operating pressure and temperature. When the reactor coolant system is intentionally cooled below normal operating values, the operator will select the Heatup-Cooldown mode which alters the limit checking algorithm for the key parameters. An additional mode may be provided to address concerns of cold shutdown plant conditions.

2. Display Hardware Locations and Operation

The SPDS portion of the SAS may be implemented on a single CRT located in a control location of the control room visible to the control room operator and the Senior Reactor Operator. This CRT contains the high-level
27 display from which the overall safety status of the plant may be assessed. A dedicated function button panel allows operator selection of several predetermined second level (trend) displays at any time.

The SAS has been designed such that control room personnel can utilize its features without requiring additional operations personnel.

The SAS displays will be provided to other ERFs such as the Technical Support Center and Emergency Operations Facility.

3. <u>Display Contents</u>

The primary display consists of bar graphs of selected
parameter values, digital status indicators for important
safety system parameters and digital values. The
parameters indicated by bar graphs and digital values 27
include: RCS pressure, RCS temperature, pressurizer
level, steam generator levels and steam generator
pressures. Status indicators are provided for
containment environment and secondary system radiation.
Core exit temperature, amount of subcooling and 36
containment radiation are indicated by digital values.

These displays continue to envolve as the SAS development group reviews their effectiveness. The final display designs will be verified and validated as described later.

In addition, there is a message area which will be used to indicate that an appropriate secondary display provides further information in case an off-normal value is detected or an event is occurring.

Each of the bar graphs indicate wide-range values. If a
parameter's value is outside the normal range, the bar
color will turn red. Arrows next to the bar will
indicate the trend direction (increasing or decreasing) 27
based on data smoothing algorithms.

Secondary displays may be selected by the operator. Trend graph groups of selected parameters, showing the last thirty minutes of plant operation are available. These trend groupings were chosen to keep like parameters or related paramters on one display "page".

4. Human Factor Engineering

Human factors engineering and industrial design techniques have been effectively combined to establish man-machine interface design requirements, maximize system effectiveness, reduce training and skill demands, and minimize operator error.

The CRT color graphic formats and functional key board designs have been developed through an interdiscipliany team of senior operational, human factors, industrial design and computer interface personnel.

Minimum use of color combined with simplified format through the CRT presentation have key design features to provide both normal and off-normal pattern recognition. The operator, who is the end user, has been directly
27 involved from the conception to insure that man-machine interface goals of SAS have been satisfied. Human factor engineering standards and testing verification have been used which are consistent with accepted practices.

5. Validation and Verification

The SAS is implemented on a digital computer system which includes a peripheral display generator computer for color graphic displays. The software that controls the sensor data validation, key parameter construction, and display formats has been developed under strict quality assurance procedures similar to those defined in Position 19 of NUREG-0308, "Safety Evaluation Report Related to Operation of Arkansas Nuclear One, Unit 2", Revision 1 and supplements. The original development of the SAS software began with a functional specification that was

developed over a period of 18 months by a technical committee comprised of members from a number of utilities and consultants. These functional specifications are transformed into a design specification. Reviews of the design specification will assure conformance to the SPDS portion of the SAS to those functions discussed in NUREG-0696. The basis for selection of the primary display parameters will be a part of the final project 27
documentation.

F. DESIGN CRITERIA

The total SPDS will not be Class 1E or meet the single-failure criterion. The sensors and signal conditioners (such as preamplifiers, isolation devices, etc.) will be designed and qualified to meet Class 1E standards for those SPDS parameters that are also used by safety systems.

36

The control room operations staff will be provided with sufficient information and criteria to allow for performance of an operability evaluation of SPDS if an earthquake should occur.

The SPDS as used in the control room will be designed to an operational unavailability goal of 0.01. The cold shutdown unavailability goal for the SPDS during the cold shutdown and refueling modes for the reactor will be 0.2.

The unavailability goal of 0.01 is more stringient than can 27
be reasonably achieved without some redundancy. Therefore, dual minicomputers, data de-multiplexors, and other critical peripherals will be installed.

Isolation of signals, derived from safety systems will be qualified isolation devices.

Power supply will be from Non-1E Battery, UPS supply.

Expandability in both numbers of parameters and processing power will be provided.

Verification and validation will be applied as appropriate.

VI. INTEGRATED ERF COMPUTER SYSTEM

A. GENERAL DESCRIPTION

In general, the integrated ERF computer system proposed for CPSES consists of a system configuration as shown in Figure III.A.1.2-2. The overall system principally consists of:

27

1. Isolation devices with integral digiting equipment which will provide 12 bit resolution of the parameter ranges. Fiber optic cable runs between the remote multiplexors/isolators and the de-multiplexors which will be located near the ERF computer. This system is considered to be the Data Acquisition System (DAS). The DAS will be powered by a non-1E, highly reliable power source. (UPS-Battery System)

2. Redundant minicomputers will provide the data processing/distribution/and record keeping functions required. The mini computers will be located in the same room as the plant process computer, but will not rely on the process computer, for any of its ERF System functions. The mini computers will be powered from a highly reliable Non-1E Battery/UPS System. This system is the ERF computer referred to in this submittal.

3. The display system will consists of seven (7) human factored, color graphics display units implementing the display generated by the Westinghouse Ad-hoc Safety Assessment System Group. Two displays will be located in the control room, three in the TSC, and two in the EOF. One of the control room displays will be dedicated to the display of SPDS type parameters. The other displays will have full display capability, including the SPDS type parameters, in addition to all other parameters available to the computer. This system is referred to as the 27
display system. The control room, and TSC displays will be powered from Non-1E, Battery/UPS power supplies.

The integrated ERF Computer System reliability design goal is to achieve 0.01 unavailability during all plant operating modes above cold shutdowns.

B. IMPLEMENTATION SCHEDULE

The ERF computer hardware and software will be installed with a sufficient set of input variables to accomplish the SPDS and Regulatory Guide 1.23 functions by fuel load. 36

III.A.2 IMPROVING LICENSEE EMERGENCY PREPAREDNESS--LONG-TERM

<u>OBJECTIVE:</u>

To upgrade the emergency preparedness of nuclear power plants.

- NUREG 0660

Action Plan Requirements:

Each nuclear facility shall upgrade its emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement is delineated in NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants."

- NUREG 0737

CPSES Response

The CPSES Emergency Plan is currently being revised to reflect requirements expressed in NUREG-0654, Revision 1.

Prior to fuel load of Unit One, the meteorological measurements program for CPSES shall consist of the following:

1) A primary meteorological measurements program.

2) A backup meteorological measurements system.

3) A system for making near real-time predictions of the atmospheric effluent transport and diffusion.

4) A capability for remote interrogation, on demand, of the atmospheric measurements and prediction systems by CPSES, emergency response organizations, and the NRC staff.

III.A.3 IMPROVING NRC EMERGENCY PREPAREDNESS

OBJECTIVE:

To enable NRC, in the event of a nuclear accident at a licensed reactor facility, to (1) monitor and evaluate the situation and potential hazards, (2) advise the licensee's operating staff as needed, and (3) in an extreme case, be able to issue orders governing such operations.

III.A.3.3 Communications

Action Plan Requirement:

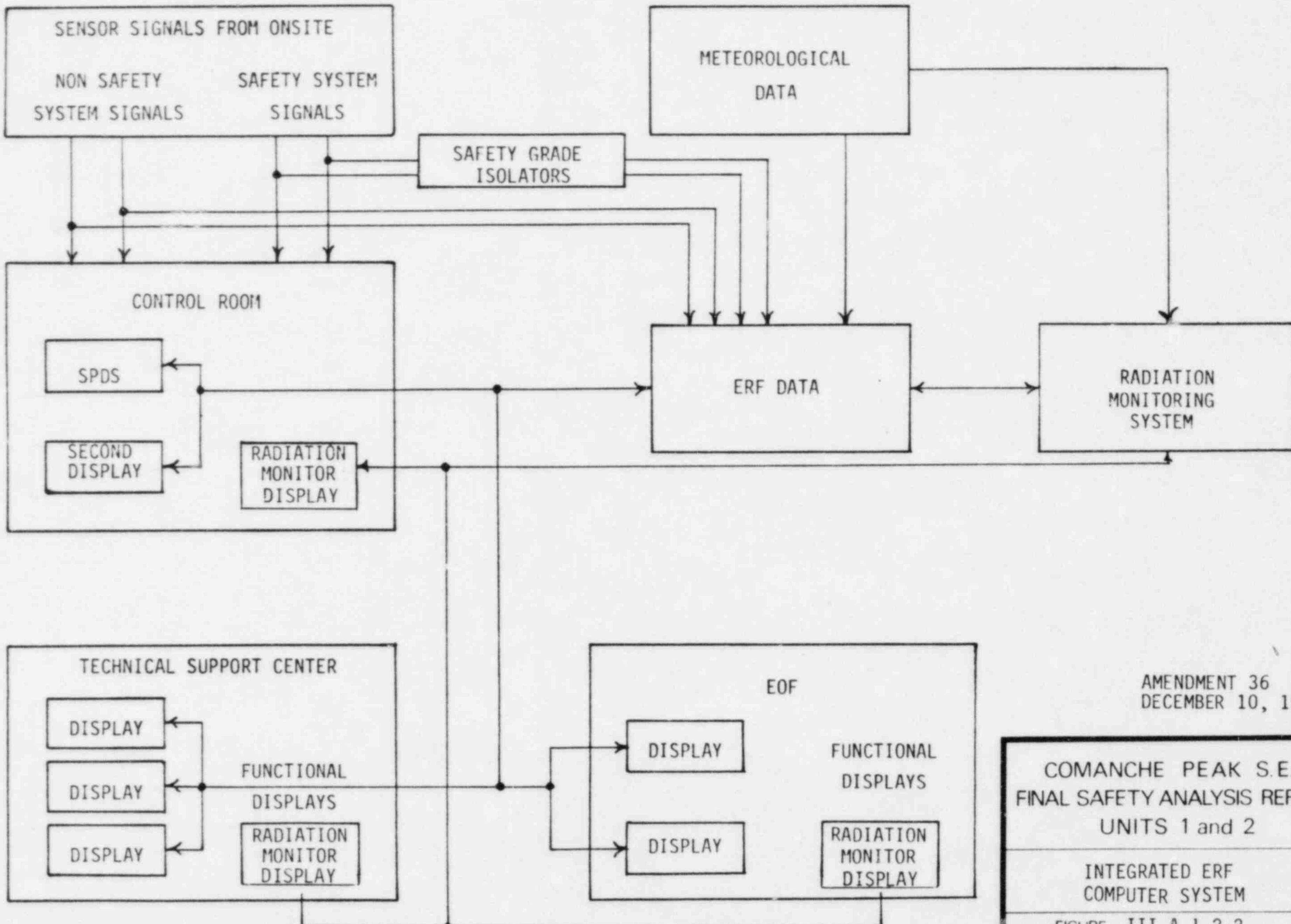
Install direct dedicated telephone lines between each plant and the NRC Operations Center.

- NUREG 0694, Pg. 27

CPSES Response

A direct dedicated telephone will be installed between CPSES and the NRC Operations Center prior to fuel loading.

INTEGRATED ERF COMPUTEP SYSTEM



AMENDMENT 36
DECEMBER 10, 1982

COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

INTEGRATED ERF
COMPUTER SYSTEM

FIGURE III.A.1.2-2

AMENDMENT 36
DECEMBER 10, 1982

COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

OPERATIONAL SUPPORT CENTER

FIGURE III.A.1.2-3

III.D RADIATION PROTECTION

III.D.1 RADIATION SOURCE CONTROL

OBJECTIVE:

"Perform evaluations to establish additional design features that should be included in the rulemaking proceeding of Item II.B.8. The purpose of these evaluations is to identify design features that will reduce the potential for exposure to workers at nuclear power plants and to offsite populations following an accident."

-NUREG 0660, Pg. III.D.1-1

III.D.1.1 Primary Coolant Sources Outside The Containment Structure

Action Plan Requirements:

"Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

"(1) Immediate leak reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates witn system in operation and report them to the NRC.

"(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle."

- NUREG 0737

CPSES Response

The residual heat removal (RHR) system; portions of the containment spray (CS) system; portions of the safety injection (SI) system; portions of the chemical and volume control (CVCS) system including letdown, makeup and high pressure ECCS; primary sampling (PS) system; and gaseous radioactive waste systems have been identified as systems which process primary coolant, and could contain high level radioactive materials. Programs will be implemented to reduce and maintain leakage as-low-as-practical. These programs will include but not be limited to the requirements of the ASME Boiler and Pressure Vessel Code, Section XI.

1. The RHR, CS, SI, and CVCS systems are ASME Code Class 2 and 3, and are subject to the in-service inspection requirements of the ASME Boiler and Pressure Vessel Code, Section XI, including pressure tests.

2. At intervals not exceeding refueling outages, operating pressure leakage tests will be performed on appropriate portions of RHR, SI, CS, CVCS (ECCS) and PS systems.

27

3. Portions of the charging system which are in service during normal operation (makeup and letdown) are monitored with the rest of the reactor coolant system for leakage during steady-state conditions by the reactor coolant system water inventory balance. Excessive leakage into controlled areas will also be indicated by

abnormally high airborne radioactivity levels. Moreover, this system will be visually inspected on a routine basis. Some portions of the CVCS are used for boron recyclying (feed and bleed to change RCS boron concentration) during normal operation. The boron recyclying system will not be used under post-accident circumstances. Therefore, the boron recycle system, including liquid holdup tanks will not be subject to intensive leakage surveillance.

4. Plant Initial Leakage Rates

The initial leakage rates of the systems to be tested will be determined from the system hydrostatic test or pneumatic test which is conducted prior to plant operations.

5. Test Methods

The primary testing method will be by leakage makeup to maintain test pressure. Systems or subsystems not readily testable in this manner will require system walkdown to quantify individual leakages. Individual leakage rates in gaseous systems will be 27
quantified most probably by helium leak detectors. Actual testing experience may suggest alternate methods which may be incorporated. Water inventory or water balance measurements may also be used on a periodic basis to determine liquid leakage rates. The flexibility of water balance measurements allows more frequent leakage determinations.

6. Limiting Leakage Value

The effort at CPSES will be to minimize the overall leakage of fluid from potentially radioactive systems to the environment. Therefore, the limiting leakage value will be based on a cumulative amount from all the systems tested. The value will be established initially as 10 gpm total with no more than 5 gpm allowable from any one system. The 10 gpm maximum is an average leakage value based on yearly average. A leakage rate from gaseous systems will be established from the initial leakage values determined from the pnuematic tests. Efforts to limit the actual leakage will be made during the initial tests. This will result in the establishment of more reasonable values.

7. <u>Maintaining Leakage Value</u>

The affected systems or subsystems will receive a periodic inspection for excessive leakages. This inspection will in general be conducted with the leak rate test of the system on a refueling interval. Inspections may be conducted more frequently as determined by maintenance policy. To minimize the actual leakage rate, packings or seals with provisions for adjustment will be adjusted whenever leakage is noted during the inspection. Replacements of seals, gaskets, o-rings, etc., will be accomplished on a selective basis to prevent exceeding the limiting leakage value. Through wall leaks will require repair
27 prior to plant restart.

Should the limiting leakage value be exceeded, a reinspection and leak test will be conducted on the system(s) with high leakage rates. Excessive leakage paths will be identified and evaluated. Areas of continued high leakage will be considered for modification or improvement in repair methods.

Modifications or repairs will be accomplished as quickly as possible. Consideration as to the type of leakage, fluid involved, leakage location and collectability will influence repair or modification schedules.

8. Incidence of Leakage at North Anna

A review and comparison of the designs of North Anna and Comanche Peak as pertains to the leakage incidence indicates that the design of the Comanche Peak volume control tank relief system is sufficiently different to preclude a similiar occurrence. In addition a further review of procedures and as-built configurations will be performed as they become available. Areas or activities which could result in release of radioactive material will be evaluated for modification. 27

III.D.3 WORKER RADIATION PROTECTION IMPROVEMENT

OBJECTIVES:

"Improve nuclear power plant worker radiation protection to allow workers to take effective action to control the course and consequences of an accident, as well as to keep exposures as low as reasonably achievable (ALARA) during normal operation and accidents, by improving radiation protection plans, health physics, inplant radiation monitoring, control room habitability, and radiation worker exposure data base."

-NUREG 0660, III.D.3-1

III.D.3.3 Improved Inplant Iodine Instrumentation Under Accident Conditions

Action Plan Requirements:

"(1) Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

"(2) Each applicant for a fuel-loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident."

- NUREG 0737

CPSES Response

22

The types of air sampling equipment to be used at CPSES for the
determination of radioiodine concentrations are portable (continuous
36 and grab-type).

Portable, continuous air samplers, which are cart-mounted will be
utilized for sampling specific areas where stationary
22 monitoring/sampling does not exist or is inadequate for specific job
needs. These are particulate, iodine and noble gas monitors/samplers,
which also use charcoal or silver zeolite cartridges for radioiodine
36 sampling. Again, the cart-mounted gamma scintillation detector can be
used to count the cartridge, or it can be removed and isotopically
analyzed in the count room. These monitors will be used for
22 sampling/monitoring for those tasks requiring it on a continuous basis
during or subsequent to an accident.

Grab-type high volume air samplers, utilizing charcoal cartridges, will be used for local sampling where stationary or continuous portable-type sampling is impossible or impractical. Cartridges will be removed and isotopically analyzed for radioiodines in the count room.

Generally, charcoal cartridges removed from air samplers will be taken to the "Hot" Chemistry Lab and purged with air to remove noble gases. Purging will be done under an appropriate ventilation hood. Cartridges will then be taken to the count room for isotopic analysis using a Ge (Li) detector, MCA and process computer to accurately determine the radioiodine concentrations. If, due to the accident conditions, the normal count room is unavailable or unaccessible, an alternative count room providing a background low enough to determine radioiodine concentrations in a reasonable period of time will be used. 22

Depending upon the conditions at the time of sampling, the sample media may be charcoal or silver zeolite cartridges, or charcoal or silver zeolite - impregnated filter paper. The appropriate sample media will be determined on the basis of the sampling environment, and the efficiency of the media for collection of radiohalogens particulates and noble gases.

Procedures will be developed based on specific equipment types/brands purchased by CPSES. A training program will be developed and include the review of procedures and demonstratin of methods for personnel responsible for air sample collection, preparation and counting/analysis to assure proficiency in determining radioiodine concentrations during accident and post-accident conditions.

III.D.3.4 Control Room Habitability Requirements

Action Plan Requirements:

"In accordance with Task Action Plan item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50)."

- NUREG 0737

CPSES Response

A reevaluation of control room habitability has been performed.

Section 2.2.1-2.2.2, 2.2.3 and 6.4 of the Standard Review Plan (SRP) were reviewed for compliance and are discussed as follows:

1) FSAR Section 2.2.1-2.2.2 is in compliance with SRP Section 2.2.1-2.2.2.

27 2) FSAR Section 2.2.3 is in compliance with SRP Section 2.2.3.

3) FSAR Section 6.4 shows that the present CPSES design for the control room is in compliance with SRP Section 6.4. An analysis is being performed to ensure that steam, from a postulated steam generator blowdown piping break in the Electrical and Control Building does not enter the control room. In the event modifications are necessary, these will also be installed prior to fuel load.

Information required by NUREG-0737, III.D.3.4, Attachment 1, is provided below:

(1) The control room ventilation mode of operation for radiological accident isolation and chlorine releases is described in FSAR Section 6.4.2.

(2) Control room characteristics

(a) The control room air volume is provided in FSAR Section 15.6.5.4 (item 4(k)).

(b) The control room envelope (i.e., emergency zone) is described in FSAR Section 6.4.

(c) The control room ventilation system schematic is provided by FSAR Figure 9.4-1.

(d) Control room infiltration leakage is discussed in FSAR Sections 6.4.2.3 and 9.4.1.2.

(e) HEPA filter and charcoal adsorber efficiencies are provided in FSAR Section 15.6.5.4 (item 4(b)).

(f) The distance between the containment and air intake is provided in FSAR Section 15.6.5.4 (item 4(g)).

(g) Layouts are provided in FSAR Figures 1.2-1, 1.2-33, 1.2-34 and 6.4-3.

(h) Control room shielding is discussed in FSAR Section 6.4.2.5.

(i) Automatic isolation capability is discussed in FSAR Sections 6.4.2.2 and 9.4.1.2.

(j) Chlorine detectors are discussed in FSAR Sections 6.4.1.5 and 6.4.4.2.

(k) Portable self-contained breathing apparatus (SCBA) are provided in the control room. There will be an adequate supply of air to sustain the five-man emergency team for a six-hour period. At least one SCBA will be provided for each member of the emergency team. There will be one additional SCBA to serve as a spare in case one unit fails.
36 See CPSES Emergency Plan Section 8.1.3.

(l) Bottled air supplys are discussed in the CPSES Emergency
36 Plan Section 8.1.3.

(m) Emergency food and water supplies are provided in FSAR Section 6.4.1.1.

(n) The shift supervisor controls the number of personnel allowed into the control room during both normal and emergency conditions. The personnel capacity is dependent on specific conditions at the time in question. The minimum capacity under design basis accident conditions is five men for a period of five days.

(o) Potassium iodide will be available to plant emergency personnel. The details of its use and the quantity to be stocked are being studied.

(3) Onsite Storage of Chlorine and Other Hazardous Chemicals

(a) The total amount and size of containers of chlorine are provided in FSAR Sections 2.2.3.1.3 and 2.2.3.2.2.

(b) The closest distance from the control room air intake is provided in FSAR Section 2.2.3.2.2 (item 2(c)).

(4) Offsite manufacturing, storage, or transportation facilities of hazardous chemicals is discussed in FSAR Section 2.2.

(5) Technical Specifications for CPSES will comply with the Standard Technical Specifications for Westinghouse Pressurized Water Reactors (NUREG-0452) for chlorine detection and the control room emergency filtration.

be applied to post-accident monitoring systems that will be used by plant operators.

R032.103 1. The following liquid level measuring systems inside containment are used to initiate safety actions.

A. Steam Generator Narrow Range Water Level

The steam generator narrow range water level detection system consists of four differential pressure measurement channels per steam generator. Each channel measures differential pressure between an upper and lower tap using an open column reference leg, a condensing pot to ensure that the reference leg maintains a constant level, and a differential pressure (level) transmitter. 36

The steam generator narrow range water level is used for the following safety functions:

- Turbine trip and feedwater isolation on high-high steam generator water level
- Reactor trip on low-low steam generator water level
- Auxiliary feedwater initiation on low-low steam generator water level 36
- Post accident monitoring

Steam Generator Wide Range Water Level

Each steam generator has one wide range differential pressure transmitter. Each transmitter employs an open column reference leg with a condensing pot to ensure that the reference leg maintains a constant level. When the steam generator level is high, the differential pressure between the vessel and the reference leg is smallest, as the steam generator level drops, the differential pressure increases.

The steam generator wide range water level has a post-accident monitoring function. 36

B. Pressurizer Water Level

The pressurizer has three differential pressure transmitters. Each transmitter employs an open column reference leg with a condensing pot to ensure that the reference leg maintains a constant level. The operation of the pressurizer water level measurement system is identical to that employed for measuring steam generator water level.

The pressurizer water level is used for the following safety functions: 36

- Reactor trip on high water level

- Post accident monitoring

2. Evaluation of the effect of post-accident conditions on indicated water level.

A. <u>Reference Leg Heatup</u>

High energy line breaks inside containment can result in the heating of level measurement reference legs. Increased reference leg water column temperature results in a decrease of the 36 water column density with a consequent apparent increase in the indicated water level (i.e., apparent level exceeding actual level).

The following formula can be used to calculate the magnitude of this bias:

$$E = \frac{H_L}{H} \quad \frac{(PL\ ,cal - PL)}{(P_f,\ cal - P_g,\ cal)}$$

where

E = level error due to reference leg heatup, as a fraction of level span

H = level span = vertical distance between pressure taps (on 36
CPSES steam generators,
Unit 1 = 233 in.,
Unit 2 = 128 in.)

H_L = height of reference leg (on CPSES steam generators, Unit 1 = 235 in., Unit 2 = 130 in.)

P_L, cal = water density at containment temperature and system pressure for which the level indication system was calibrated. At CPSES, the containment temperature is 122^{o}F and the steam generator pressure is 1107 psia. 36

P_L = water density in reference leg at the time of interest

(P_f,cal-P_g,cal) = difference between saturated water density and dry saturated steam density at the system pressure for which the level indication system was calibrated. (1107 psia for CPSES steam generators). 36

This procedure is based on the assumption that the tubing from the upper and lower taps, below the elevation of the lower tap, have the same temperature at all times.

Figure 032.103-1 shows the steam generator level bias as a function of reference leg temperature for CPSES conditions. 36

B. Reference Leg Boiling

In addition to the above reference leg density change under subcooled conditions, boiling could conceivably occur in the reference leg following depressurization of any steam generator with high containment temperature. This combination of conditions could only occur following a steamline or feedline rupture inside containment. If such boiling were to occur, it could cause a major bias in the indicated level for a short time period, in the extreme case indicating 100 percent level when the vessel is actually empty.

However, containment analyses performed by Westinghouse indicate that such boiling would not occur. 36

C. Coolant Density Changes

A bias in indicated water level may also be introduced by changes in pressurizer or steam generator pressure, due to changes in the density of the saturated water and steam within those vessels. While prediction of the effects of rapid depressurization requires complex calculations for each specific case, the bias which would exist at low power under quiescent conditions can be calculated directly, using the following formula:

$$E = \frac{H_L}{H} \frac{(PL, cal - PL - P_g, cal + P_g)}{(P_f, cal - P_g, cal)} + \frac{L}{H} \frac{(P_f - P_g)}{(P_f, cal - P_g, cal)} - \frac{L}{H}$$

where

E = level error due to density changes in both the vessel and the reference leg, as a fraction of level span

L = true water level in the vessel, above the lower level tap

P_f = saturated water density at the pressure of interest

P_g = dry saturated steam density at the pressure of interest

Other symbols have the same meaning as in the previous section.

For an example, Figure 032.103-2 shows the true water level as a function of steam generator pressure and indicated level using the following CPSES calibration conditions: containment temperature = 122^{O}F, steam generator pressure = 1107 psia, and the reference leg is at 122^{O}F. Figure 032.103-3 is similarly calculated for the pressurizer, with the assumptions noted on the Figure. 36

3. Safety Function Setpoints

A. Steam Generator Water Level Trip Setpoints

The only high-energy line rupture within the containment for which the steam generator water level provides the primary trip function is a secondary high energy line rupture from an initial high power condition. For such a case the steam generator low-low water level trip must be actuated when the pressure difference between the narrow range level taps corresponds to a minimum value of 3% of the distance 36
between the taps. Thus, the trip setpoints must be at or above the value that would be indicated at 3% true level.

Because large steam generator pressure changes are not expected before trip, only the reference leg heatup effects need be considered, and not the effects of system pressure changes.

The determination of low-low setpoints (minimum) for the steam generator low-low level trips at CPSES is as follows:

Normal errors (normal channel accuracy) (%)	$\pm$ 5	
Post-accident effects on transmitter (radiation & temperature) (%)	$\pm$10	36
Reference leg effects (post accident heatup) (%)	+5.2	
TOTAL ERROR, % of span	+20.2	
Lowest permissible Lo-Lo Trip Setpoint (%)	3	
	23.2%	

Minimum setpoint including inaccuracies	23.2% of span above lower top of narrow range level transmitter

At CPSES the low-low setpoints will be increased above the existing setpoints as shown below. Considered a feedline break and peak compartment temperature of 227°F the error due to reference leg heatup will be approximately 5.2%.

	Unit 1	Unit 2
Current Lo-Lo Level trip setpoint	40.8%	17%
Error due to heatup of ref leg to 227.3°F	5.2%	5.2%
Raising minimum permissible Lo-Lo trip setpoint from 2% to 3% on Unit 2		1%
New Lo-Lo Level trip setpoint	46%	23.2%

36

Note that although Unit 2 steam generator Lo-Lo level setpoint must be raised over its current setpoint to correct for reference leg heatup, the Unit 1 steam generator setpoint is already high enough (40.8%). Unit 1 SG Lo-Lo setpoint, however, will be increased from 40.8% to 46% to preclude Westinghouse reanalysis.

Raising the Lo-Lo level setpoints has the following effects on operating margin (margin between trip setpoint and lowest normal operating point):

Unit 1 - Operating margin existing 48.1 inches. Operating margin after setpoint change 36.1 inches.

Unit 2 - Operating margin existing 35.8 inches. Operating margin after setpoint change 27.9 inches. 36

Although the margin between the trip setpoints and operating water levels will be reduced, spurious trips are not expected to occur.

The high-high steam generator water level trip is not required for accident situations that could cause significant errors in level indication. The setpoint of this trip will remain unchanged.

Steam generator wide range level does not provide any automatic trip functions.

B. <u>Pressurizer Water Level</u>

Pressurizer water level provides no trip function following an accident which results in an adverse environment inside containment.

4. <u>OPERATOR ACTIONS</u>

A graph depicting the level measurement error due to steam generator reference leg heatup is shown in 36

Figure 032.103-1. Error due to system pressure changes are shown in Figures 032.103-2 and 032.103-3 for the steam generator and pressurizer indicated water level respectively. Plots similar to these will be made available to the operators to ensure they are aware of the potential level measurement errors. 36

Furthermore, a remote possibility exists that the fluid in the open reference legs may flash to steam in the depressurized steam generators following a secondary high energy line rupture. Therefore to alert the operator to the possibility of erroneous indications, Westinghouse has recommends that the following caution be inserted in all plant emergency instructions for indicated steam generator water level.

CAUTION

> The operator should not rely upon steam generator water level indications in any depressurized steam generators following a high energy line rupture inside containment. This is due to the possibility of reference leg boiling.

The Westinghouse reference Emergency Operating Instructions take the post accident indicated water level errors into account in the specification of the minimum levels required for safety injection terminations. 36

Similar cautions will be included in the CPSES emergency procedures.

stated that the performance of non-safety grade equipment subjected to an adverse environment could impact the protective functions performed by safety grade equipment. Of all the control systems considered, four specific control systems were identified as possible sources of problems.

Automatic Rod Control System
Pressurizer Power Operated Relief Valve Control System
Main Feedwater Control System
Steam Generator Power Operated Relief Valve Control System

We believe this response addresses the only CPSES control systems which might effect protective functions.

Our analysis shows that all four systems are vulnerable to unacceptable interactions due to high energy line breaks. CPSES eliminated the problem by qualifing or moving all susceptible components in those systems to locations not affected by the high energy line breaks. 36

1. Automatic Rod Control System

Safety Implications

A simplified block diagram of the automatic rod control system is shown in Figure 7.7-1 of the CPSES FSAR. The power mismatch compensation unit uses a power-range nuclear flux signal and turbine load signal to develop an input to the rod-speed unit. The power mismatch compensation unit causes the control rods to withdraw when the nuclear flux signal decreases to less than the turbine load signal. In the event of a main steam line break inside containment, the adverse environment could cause the nuclear flux detector to fail low causing the rods to withdraw. A feedline or steamline break outside the containment could cause the turbine load transmitters to fail high with the same consequences. This would cause the reactor power to increase to a level which could possibly damage the fuel cladding.

Westinghouse Postulated Sequence of Events

a) Reactor power is between 70 and 100 percent and the rod control system is in automatic.
b) A steamline break occurs inside reactor containment or feedline or steamline break outside the containment.
c) The break causes an adverse environment. This adverse environment causes the reactor excore neutron detectors to fail low or the turbine load transmitters to fail high.
d) One of these signals causes the rod control system to withdraw the control rods.
e) This control rod withdrawal increases the reactor power and the departure from nucleate boiling ratio (DNBR) drops to an undetermined value before the reactor trips.

Analysis of CPSES

a) Assumptions Made in the Analysis

   1) Only those components exposed to the adverse evironment fail.

2) Equipment which is environmentally qualified to IEEE 323-1974 will not fail in an adverse environment.

b) Relevant Design Features at CPSES

1) The power range excore detectors at CPSES are not presently environmentally qualified.

2) The turbine impulse pressure transmitters at CPSES are outside on the turbine deck. They are not exposed to the postulated environment.

c) CPSES Sequence of Events

The CPSES sequence of events is the same as the Westinghouse postulated sequence.

d) Termination of the Transient

This transient will be terminated by a reactor trip with DNBR falling to an undetermined value.

Resolution

The excore detectors have been Environmentally Qualified to the environments that may exist due to a steamline break inside containment prior to a reactor trip (see CPSES FSAR Section 3.11N and Appendix 3A). 36

Conclusion

The affected devices are qualified such that no unresolved safety question exists concerning the automatic rod control system. 36

2. Pressurizer PORV Control System

Safety Implications

A block diagram of the pressurizer pressure control system is given in Figure 7.7-4 of the CPSES Final Safety Analysis Report. The PORV's are controlled from the pressurizer pressure signal. A failure of the pressurizer pressure transmitter due to adverse environmental conditions could cause the PORV's to open.

Westinghouse Postulated Sequence of Events

a) The plant is operating at full load.
b) A rupture occurs in a main feedwater line inside containment.
c) Feedwater and the contents of the steam generator spill into the containment.
d) Steam generator water level drops and the reactor trips on low-low water level.
e) Auxiliary feedwater pumps start on low-low steam generator water level.
f) The turbine trips on reactor trip.
g) The feedwater inside containment flashes to steam at the lower pressure and affects the pressurizer pressure transmitter causing it to fail high.

h) The control system opens the PORV's in trying to lower pressurizer pressure to the pressure setpoint.

i) Pressure in the primary system decreases and hot leg boiling results.

j) Loss of cooling causes fuel cladding failures which allow fission products to escape from the fuel rods and enter the reactor coolant.

Analysis of CPSES

a) Assumptions Made in the Analysis

1) Only those components exposed to the adverse environment fail.
2) Equipment which is environmentally qualified to IEEE 323-1974 will not fail in an adverse environment. 36

b) Relevant Design Functions at CPSES

The components of the PORV Control System located inside containment are:

1) Pressurizer ' essure Transmitters
2) Transmitter Wiring
3) Solenoid Valves
4) Solenoid Valve Wiring

All the components are qualified to the postulated environmental conditions. 36

c) CPSES Sequence of Events

The CPSES sequence of events is the same as the Westinghouse postulated sequence.

d) Termination of the Transient

This transient is terminated by a reactor trip and auxiliary feedwater injection.

Resolution

The components of the PORV Control System at CPSES have been environmentally qualified for the postulated environment.

Conclusion

The postulated failure could have resulted in a reactor trip with a pressure boundary break and a loss of heat sink. This condition has not been analyzed in our Final Safety Analysis Report and is considered unacceptable. Therefore, all components of the control system have been environmentally qualified to the postulated environment. 36

3. Main Feedwater Control System

Safety Implications

A block diagram of the main feedwater control system is shown in Figure 7.7-6 of the CPSES Final Safety Analysis Report. The position of the main feedwater control valve is partially determined by several transmitter signals. A failure of these transmitters could cause the feedwater control valves to partially close. This would restrict the flow of feedwater to the steam generators. Since water is required in the steam generators to remove decay heat after a reactor trip, it is postulated that the steam generators might not contain enough water at the time of reactor trip to provide adequate core cooling.

Westinghouse Postulated Sequence of Events

a) The reactor is operating at full power.

b) A small feedwater line break occurs inside containment. 36

c) Feedwater spills out of the break and flashes to steam at the lower pressure.

d) Steam from the break causes the transmitters to fail. The control system then lowers the water level.

e) The affected steam generators all reach low-low water level.

f) It is undetermined at this time if the water inventory in the steam generators is sufficient to prevent hot leg boiling.

Analysis of CPSES

a) Assumptions Made in the Analysis

1) Only those components exposed to the adverse environment fail.

2) Equipment which is environmentally qualified to IEEE 323-1974 will not fail in an adverse environment.

b) Relevant Design Features at CPSES

The steam flow and steam generator level
transmitters of the feedwater control system
are inside the containment and are qualified to
the environmental conditions resulting from a
LOCA or MSL break. The other components of the 36
feedwater control system are located outside
containment and, therefore, will not be exposed
to an adverse environment.

c) CPSES Sequence of Events

The CPSES sequence of events is the same as the Westinghouse postulated sequence.

d) Termination of the Transient

This event is terminated with a reactor trip coincident with a lower than analyzed secondary coolant inventory.

Resolution

The steam flow and steam generator level components
of the main feedwater control system have been 36
upgraded to survive the postulated environment.

Conclusion

The postulated failure would have resulted in a 36
reactor trip with an unusually low secondary coolant
inventory. This condition has not been analyzed in
our Final Safety Analysis Report and is considered

unacceptable. Therefore, all components of the control system which can be exposed to the postulated environment have been qualified to that environment. | 36

4. Steam Generator PORV Control System

Safety Implications

A system logic diagram of steam generator PORV Control System (called atmospheric relief valves) is given in Figure 7.2-1 (sheet 10) of the CPSES Final Safety Analysis Report. The PORV's are controlled using a steam line pressure signal. A failure of the steam line pressure transmitters due to adverse environmental conditions could cause the PORV's to open and the result would be equivalent to a small line break. This would result in the steam generators blowing down to atmosphere with two undesirable consequences:

a) The uncontrolled loss of steam from the steam generators would cause the reactor coolant temperature to decrease reducing shutdown margin and a possible return to criticality.

b) The loss of steam to power the turbine-driven auxiliary feedwater pump.

Westinghouse Postulated Sequence of Events

a) The reactor is operating at 100 percent power.

b) A feedwater line ruptures in the safequards building.

c) Feedwater and steam generator water spills out of the break and flashes to steam at the lower pressure.

d) The reactor trips on low-low water level in affected steam generator.

e) Auxiliary feedwater pumps start on low-low steam generator water level.

f) The turbine trips on reactor trip.

g) The three unaffected steam generators begin repressurizing on steam line isolation.

h) The steam from the break affects the steam generator PORV control systems for vulernable steam generators causing them to fail high.

i) The steam generator control system opens the PORV's trying to reduce steam generator pressure to the pressure setpoint.

j) The PORV's open and blowdown to atmosphere. 36

k) As the steam generators depressurize, steam to the turbine driven auxiliary feedwater pump is lost and the pump fails to supply auxiliary feedwater to any steam generators.

36

Analysis of CPSES

a) Assumptions Made in the Analysis

1) Only those components of the control system exposed to a hostile environment fail.

2) Equipment which is environmentally qualified to IEEE 323-1974 will not fail in an adverse environment. 36

b) Relevant Design Features at CPSES

The components of the PORV control system in the postulated area are:

1) Steamline Pressure Transmitters
2) Wiring
3) PORV I/P's

The components are IE but were not specified for a break environment.

c) CPSES Sequence of Events

The CPSES sequence of events is the same as the Westinghouse postulated sequence.

d) Termination of the Transient

If this event were to occur as postulated, the operator would have to take manual action to close the blowdown path. This would also establish a source of steam to the steam driven auxiliary feedwater pump.

Resolution

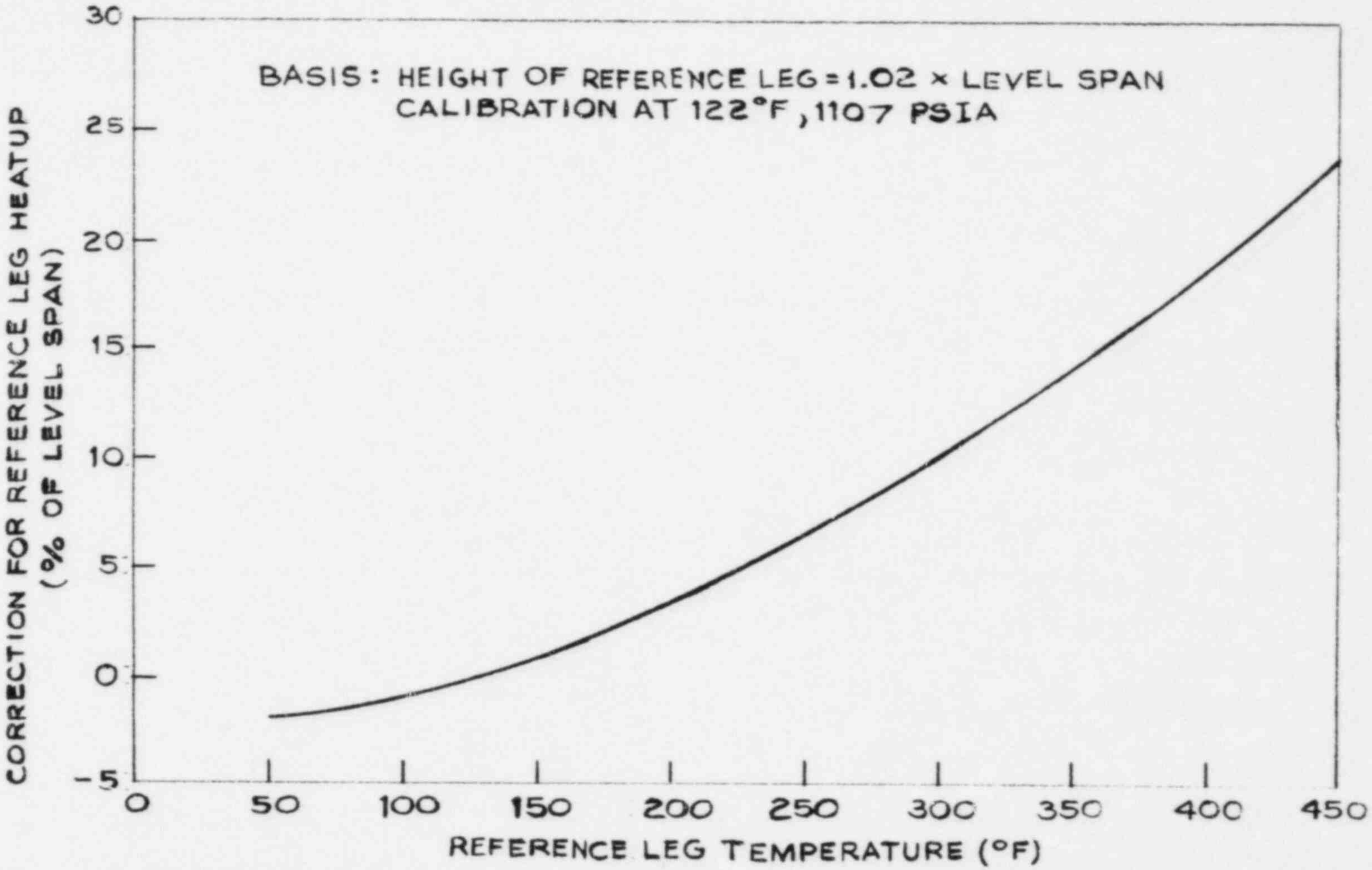
The sequence of events described requires operator action to terminate the transient. CPSES will therefore, upgrade or protect the transmitters if they are not qualified to the postulated environment. 36
CPSES has moved the I/P's outside of the compartment walls into an area unaffected by the break.

Conclusion

The postulated failure of the steam generator (PORV) control system would have resulted in the blowdown to atmosphere of the steam generators unless terminated by operator action. This condition has not been analyzed in our Final Safety Analysis Report and is 36
considered unacceptable from a safety standpoint. Therefore, all vulnerable components of the steam generator (PORV) control system will be environmentally qualified, protected or moved out of the break area. This action prevents the uncontrolled blowdown without operator action.

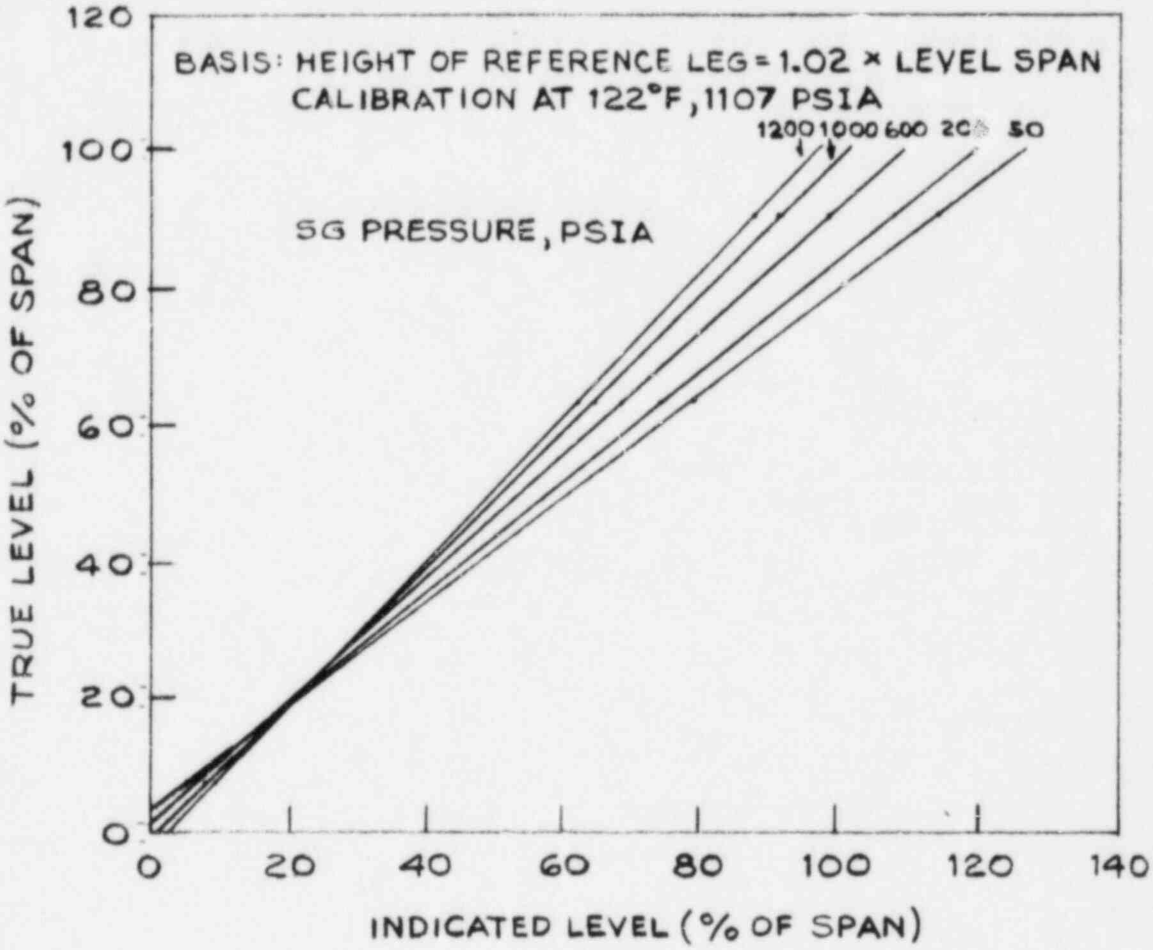
Summary

We have examined the four specific control systems identified by Westinghouse as possible sources of problems. We found that all four could lead to transients that were not analyzed in our Final Safety Analysis Report. The devices which are subject to a postulated adverse environment will be qualified or moved to locations not subject to the severe environment. We therefore, believe that unresolved safety questions do not exist for control systems at CPSES.



AMENDMENT 36
DECEMBER 10, 1982

COMANCHE PEAK S.E.S. FINAL SAFETY ANALYSIS REPORT UNITS 1 and 2
BIAS DUE TO STEAM GENERATOR REFERENCE LEG HEATUP
FIGURE 032.103-1

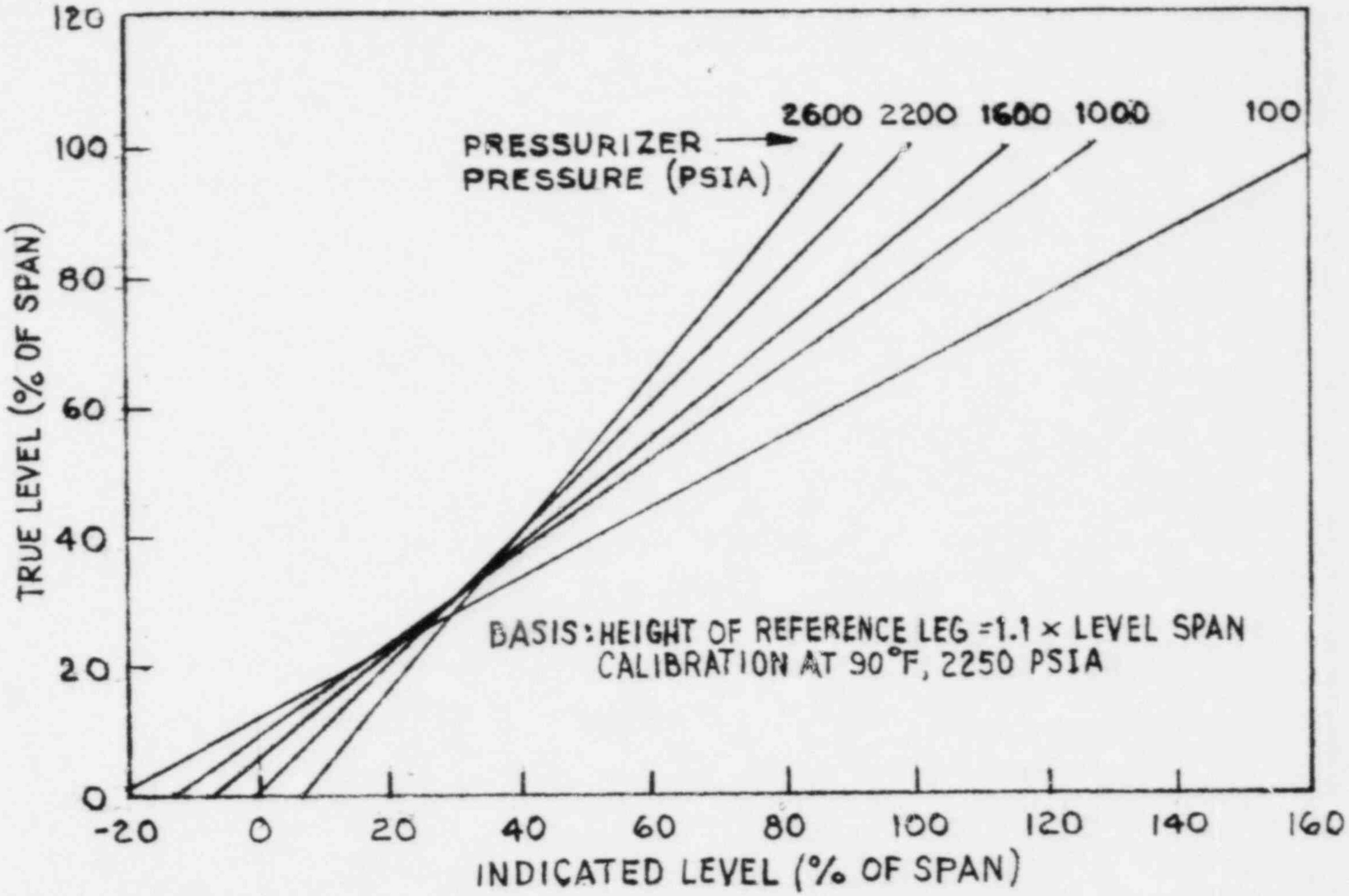


AMENDMENT 36
DECEMBER 10, 1982

COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

BIAS DUE TO STEAM GENERATOR PRESSURE CHANGE

FIGURE 032.103-2



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

BIAS DUE TO PRESSURIZER PRESSURE CHANGE

FIGURE 032.103-3

AMENDMENT 36
DECEMBER 10, 1982

Q212.145 The analyses of a reactor coolant pump locked rotor and a reactor coolant pump sheared shaft in the FSAR assumes the availability of offsite power throughout the event. In accordance with Standard Review Plan 15.3.3 and GDC 17, we require that this event be analyzed assuming turbine trip and coincident loss of offsite power to the undamaged pumps.

Appropriate delay times may be assumed for loss of offsite power if suitably justified.

Steam generator tube leakage should be assumed at the rates specified in the Technical Specifications.

The event should also be analyzed assuming the worst single failure of a safety system active component. Maximum technical specification primary system activity and steam generator tube leakage shoulo be assumed. The analyses should demonstrate that offsite doses are less than the 10CFR100 guidelines values.

R212.145 The postulated locked rotor accident, coincident with loss of offsite power, is considered extremely unlikely. A presentation given to the NRC in May 1976 showed the effects of loss of offsite power coincident with a locked rotor for 4-loop plants. The results showed a small effect on peak pressure, clad temperature and the number of rods in DNB. The number of rods in DNB increased by only a couple of percent. Locked rotor is a very short term accident and is over in a couple of seconds such that even with the coastdown in the other three loops, flow would still be high at the time of minimum DNBR. Grid stability analyses for Comanche Peak show that offsite power will not be lost afte a reactor trip at CPSES. Therefore, the results presented to the NRC in May 1976, where a two second time delay between reactor trip and loss of offsite power is assumed, is conservative.

The radiological consequences of the postulated locked rotor accident presented to the NRC in May 1976 are bounded by the analysis of the rod ejection accident. For the rod ejection accident, the fuel rods are assumed to be in DNB much longer than would be the case for the locked rotor event.

In performing the dose calculations, it is assumed 10% rods to be in DNB for rod ejection with loss of offsite power. The 10% assumed is much greater than would be expected from a locked rotor coincident with loss of offsite power for Comanche Peak.

The assumption that cladding fails when the DNBR reaches 1.3 is conservative. The time rods are expected to be in DNB is so short, only a small fraction would be conservatively predicted to actually experience DNB and potentially fail. Therefore, based on the above discussion, it is considered unnecessary to analyze the locked rotor event coincident with a loss of offsite power for Comanche Peak.

Q400.4 Provide a compilation which documents that the Comanche Peak Steam Electric Station, Units 1 and 2 will comply with all applicable NRC regulations. The compilation should list each of the subsection to 10 CFR Part 20, 50 and 100, with their respective appendixes, and state whether the applicable requirements of that regulation are statisfied. Where appropriate, the statement should include references to specific sections within the FSAR where additional details on the manner of compliance are described.

R400.4 A compilation of the CPSES compliance with subsections of 10 CFR Part 20, 50 and 100, with their appendixes is provided in Table Q400.4-1.

TABLE Q400.4-1 (Sheet 1 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
20.1(a)	This regulation states the general purpose for the Part 20 regulations and does not impose any independent obligations on licensees.
20.1(b)	This regulation describes the overall purpose of the Part 20 regulations. It does not impose any independent obligations on licensees.
20.1(c)	Conformance to the ALARA principle is ensured by Company policies, appropriate Technical Specifications and radiation protection procedures. Chapters 11 and 12 of the FSAR describe specific equipment and design features used in these efforts.
20.2	This regulation establishes the applicability of the Part 20 regulations and imposes no independent obligations on licensees.
20.3	This regulation defines words and phrases used in PART 20. It does not impose independent obligations on licensees.
20.4	The units of radiation "DOSE" defined in regulations 20.4(a), 20.4(b), 20.4(c) and 20.4(d) are accepted.
20.5(a)	The units of radioactivity specified in this regulation are accepted.
20.5(b)	[Deleted]

TABLE Q400.4-1 (Sheet 2 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
20.6	This regulation specifies the authorized interpretor of PART 20. It does not impose independent obligations on licensees.
20.7	This regulation gives the address of the NRC and does not impose independent obligations on licensees.
20.101(a)	The radiation dose limits specified in this regulation will be complied with by adherence to administrative policies and controls and appropriate radiation protection procedures.
20.101(b)	The radiation dose limits permitted in this regulation will be complied with by adherence to administrative policies and controls and appropriate radiation protection procedures.
20.102	The "Determination of Prior Dose" required by regulations 20.102(a), 20.102(b) and 20.102(c) are accepted. Records will be kept on form NRC-4 or on a clear and legible record containing all the information required in that form. These records will contain the required signatures and be retained and preserved. Administrative policies and radiation protection procedures will control all determinations.
20.103(a)	Compliance with this regulation is implementation of appropriate health physics procedures relating to air sampling for radioactive materials, and bioassay of individuals for internal contamination. Administrative policies and controls provide adequate margins of safety for the protection of individuals against intake of radioactive materials. The systems and equipment described in Chapters 11 and 12 of the FSAR provide the capability to minimize these hazards.
20.103(b)	Appropriate process and engineering controls and equipment, as described in Chapters 11 and 12 of the FSAR, are installed and operated to maintain levels of airborne radioactivity as

TABLE Q400.4-1 (Sheet 3 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
	low as practicable. When necessary, as determined by station administrative guidelines, additional precautionary procedures are utilized to limit the potential for intake of radioactive materials.
20.103(c)	The CPSES Respiratory Protection Program will comply with this regulation and follow the guidance of Regulatory Guide 8.15. See CPSES FSAR section 12.5.2.
20.103(d)	This regulation describes further restrictions which the Commission may impose on licensees. It does not impose any independent obligations on licensees.
20.103(e)	TUGCO will notify the appropriate NRC Regional office at least 30 days before the first use of respiratory protective equipment.
20.103(f)	The CPSES Respiratory Protection Program will comply with the requirements of Regulation 20.103(c).
20.104(a),(b)&(c)	TUGCo policies and the CPSES General Health Physics Plan shall prevent individuals under the age of 18 to access to restricted areas.
20.105(a)	The anticipated average radiation levels and anticipated occupancy times for unrestricted areas are discussed in FSAR Section 12.3.1.3.
20.105(b)	CPSES Administrative policies and technical specification which control the use and transfer of radioactive materials assure compliance with this regulation. Design limits for dose rates and occupancy times for unrestricted areas are provided in FSAR Section 12.3.1.3 and Table 12.3-6.
20.106(a)	CPSES facilities arrangements are such that for unrestricted areas radioactivity in effluents in those areas shall not exceed the limits of this regulation (i.e. Appendix B, Table II of Part 20). See FSAR section 12.3.1.3 discussion of Zone I.

TABLE Q400.4-1 (Sheet 4 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
20.106(b)&(c)	TUGCO does not currently intend application for limits in excess of those specified in 20.106(a) for unrestricted areas.
20.106(d)	Radioactivity concentrations at the boundary of Restricted Areas will be within the limits specified by this regulation. Physical structures (walls), shielding and containing radioactive fluids in vessels and pipes are assurances that the limits are not exceeded. Radioactivie effluents that may be discharged through conduits will pass through liquid or gaseous radwaste systems. The system descriptions and radiological calculation methods are described in detail in CPSES FSAR Section 11.
20.106(e)	This regulation provides criteria by which the Commission may impose further limitations on releases of radioactive materials made by a licensee. It imposes no independent obligations on licensees.
20.106(f)	This regulation states that the provisions of 20.106 do not apply to disposal of radioactive material into sanitary sewerage systems. It imposes no independent obligations on licensees.
20.107	This regulation clarifies that Part 20 regulations are not intended to apply to the intentional exposure of patients to radiation for the purpose of medical diagnosis or therapy. It does not impose any independent obligations on licensees.
20.108	This regulation describes criteria by which the Commission may require a licensee to provide an individual appropriate boi-assay services. The CPSES Radiation Protection Program will include a bio-assay program following the guidance of Regulatory Guide 8.9. See FSAR Section 12.5.2.
20.201(a)	This regulation defines "survey" as an evaluation of radiation hazards. It imposes no independent obligations on licensees.

TABLE Q400.4-1 (Sheet 5 of 34)

TABULATION OF CPSES COMPLIANCE WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
20.201(b)	The Radiation Protection Section will routinely survey selected areas at CPSES for the assessment of radiation-field, radioactive contamination and airborne radioactivity levels. Surveys are discussed in detail in CPSES FSAR Section 12.5.3.
20.202(a)	TUGCO will provide personnel monitoring equipment and require its use. Details of the monitoring equipment and the monitoring system are found in FSAR Section 12.5.2 and Table 12.5-2.
20.202(b)	This regulation provides some definitions of phrases used in this Part. It does not impose any indpendent obligations on licensees.
20.203(a)	The conventional radiation caution colors and "three-bladed design" radiation symbol prescribed by this regulation are accepted.
20.203(b)	The posting requirement for "Radiation Areas" prescribed by this regulation are accepted.
20.203(c)	The requirements of this regulation are accepted. Radiation protection design features (FSAR Section 12.3) and the General Health Physcis Plan (section 12.5) are the instruments of implementation.
20.203(d)	This regulation defines "Airborne Radioactivity Areas" and requires specific posting. It is accepted and complied with by use of Administrative Controls and the Health Physics Program.
20.203(e)	This regulation requires additional posting requirements. It is accepted and complied with by use of Administrative Controls and the General Health Physics Plan.
20.203(f)	The container labeling requirements of this regulation will be complied with by appropriate Administrative Controls and Radiation Protection procedures.

TABLE Q400.4-1 (Sheet 6 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
20.204(a),(b),(c)&(d)	The exceptions to part 20.203 specified by this regulation will be used where appropriate. Safe and proper application of these exception will be provided by Radiation Protection procedures.
20.205(a),(b),(c)&(d)	All of the requirements of this regulation pertaining to procedures for picking up, receiving, and opening packages of radioactive materials are implemented by the General Health Physics Plan and controlled by Administrative procedures.
20.206	This regulation requires Instruction of Personnel as specified in 10 CFR 19.12. Radiation protection training is included in the General Employee Training Program described in CPSES FSAR Section 13.2.1.1.4.
20.207(a)	Licensed material stored in unrestricted areas is secured from unauthorized removal as described in CPSES FSAR Section 12.3.1.2.9.
20.207(b)	Surveillance of licensed material is governed by Radiation Protection procedures.
20.301(a),(b)&(c)	The solid Waste Management System (CPSES FSAR Section 11.4) controls the packaging of licensed materials for ultimate disposal.
20.302(a),(b)&(c)	TUGCo does not currently plan application for Waste Disposal by any means other than authorized by this regulation.
20.303(a),(b),(c)&(d)	TUGCo has no current plans to dispose of licensed material by any other means than described in the response to Regulation 10 CFR 20.301.
20.304	[Removed]
20.305	TUGCO has no current plans to dispose of licensed material by any other means than described in the response to Regulation 10 CFR 20.301.

TABLE Q400.4-1 (Sheet 7 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
20.306(a),( ),(c)&(d)	TUGCO has no current plans to dispose of licensed material by any means other than the response to Regulation 10 CFR 20.301.
20.401(a),(b)&(c)	Records of surveys, radiation monitoring, and disposal of licensed materials are maintained as part of the Radiation Protection procedures.
20.402(a),(b),(c)&(d)	TUGCO has established an inventory and control program as part of the Radiation Protection procedures to account for all licensed material. Any loss or theft of licensed material in quantities large enough to be hazardous to persons in unrestricted areas shall be reported according with the requirements of this regulation.
20.403(a),(b),(c)&(d)	This regulation establishes notification guidelines for incidents involving licensed material and special nuclear material that has caused or threatens to cause excessive radiation exposures, releases, loss of operating facilities and excessive property damage. Compliance is assured by implementing Administrative Procedures and by the General Health Physics Plan.
20.404	[Sec. 20.404 was deleted effective September 17, 1973]
20.405(a)&(b)	Compliance with the reporting requirements of this regulation is assured by implementing Administrative Procedures and by the General Health Physics Plan.
20.406	[Sec. 20.406 was deleted August 17, 1973 effective September 17, 1973].
20.407(a)&(b)	Records that will supply the data needed to complete the personnel monitoring reports required by this regulation are maintained by implementing Radiation Protection procedures.
20.408(a)&(b)	The requirement to report an individuals exposure to radiation during employment, when his employment terminates, is accepted.

TABLE Q400.4-1 (Sheet 8 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
20.409(a)&(b)	The notification and reporting requirements of this regualtion, and those referred to by it, are satisfied by the provisions of the CPSES Health Physics procedures.
20.501	This regulation provides for the granting of exemptions for Part 20 regulations, provided such exemptions are authorized by law and will not result in undue hazard to life or property. It does not impose independent obligations on licensees.
20.502	This regulation notifies licensees that the Commission may impose upon any licensee requirements which are in addition to the regulations of Part 20. It does not impose independent obligations on licensees.
20.601	This regulation describes the remedies which the Commission may obtain in order to enforce its regulations and sets forth those penalties or punishments which may be imposed for violations of its rules. It does not impose any independent obligations on licensees.
50.1	This regulation merely states the purpose of the Part 50 regulations and does not impose any independent obligations on licensees.
50.2	This regulation defines terms used in Part 50 and does not impose independent obligations on licensees.
50.3	This regulation governs the interpretation of the regulations by the NRC and does not impose independent obligations on licensees.
50.4	This regulation gives the address on the NRC and does not impose independent obligations on licensees.
50.10(a),(b),(c),(d) &(e)	These regulations specify the types of activities that may not be undertaken without a license from the NRC. TUGCO is in compliance with this regulation and will not conduct any activities at the CPSES without an NRC license.

TABLE 0400.4-1 (Sheet 9 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
50.11(a),(b),&(c)	These regulations define exception and exemption from licensing requirements. They impose no independent obligations on licensees.
50.12(a)&(b)	These regulations allow an applicant to apply for exemption from Part 50.10 requirements. Such application was undertaken. The commission granted an LWA on Oct. 17, 1974 for certain activities within the scope of parts 50.10(e)(1) and 50.10(e)(3).
50.13(a)&(b)	This regulation says that a license applicant need not design against acts of war. It imposes no independent obligations on licensees.
50.20	This regulation states the commission shall issue only two classes of license. It imposes no independent obligation on licensees.
50.21(a),(b),&(c)	These regulations defines class 104 licenses. It does not impose any independent obligations on licensees.
50.22	This regulation defines class 103 licenses. It imposes no independent obligations on licensees.
50.23	This regulation allows the commission to issue a construction permit prior to the issuance of a license. It imposes no independent obligations on licensees.
50.24	[Deleted December 23, 1970, effective December 29, 1970]
50.30(a),(b),(c),(d) (e)&(f)	These regulations define the form and contents of applications for license. TUGCO has complied with these requirements when filing its application and its amendments (including an Environmental Report).
50.31	This regulation allows for combining several applications for different kinds of licenses. It does not impose independent obligations on licensees.

TABLE Q400.4-1 (Sheet 10 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
50.32	This regulation allows incorporation, of clear and specifiec reference information contained in previous applications, in his application. It does not impose independent obligation on licensees.
50.33(a) thru (k)	This regulation requires the license application to contain certain general information. This information is provided in TUGCO's operating license application for CPSES.
50.33a(a) thru (e)	This regulation requires applicants for construction permits to submit information required for antitrust review. The required information was provided and accepted when TUGCO applied for construction permit.
50.34(a)	This regulation governs the contents of the Preliminary Safety Analysis Report and is relevant to the construction permit stage rather than the operating license stage.
50.34(b)	A Final Safety Analysis Report (FSAR) has been prepared and submitted, which addresses in the chapters indicated the information required: (1) site evaluation factors - Chapter 2 (2) structures, systems and components - Chapters 3, 4, 5, 6, 7, 8, 9, 10, 11, 12 (3) radioactive effluents and radiation protection - Chapters 11, 12 (4) design and performance evaluation - ECCS performance is discussed and shown to meet the requirements of 10CFR50.46 in Chapters 6 and 15. (5) results of research programs - Chapter 1 (6) (i) organizational structure - Chapter 13 (ii) managerial and adminstrative controls - Chapters 13 and 17.

TABLE Q400.4-1 (Sheet 11 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
	Chapter 17 discusses compliance with the quality assurance requirements of Appendix B.
	(iii) plans for preoperational testing and initial operations - Chapter 14
	(iv) plans for conduct of normal operations - Chapters 13 and 17. Surveillance and periodic testing is specified in the Technical Specification.
	(v) plans for coping with emergencies - Emergency Plan (Chapter 13)
	(vi) Technical Specifications - prepared in conjuction with the Staff (Chapter 16)
	(vii) not applicable, since the operating license application was filed prior to February 5, 1979
	(7) technical qualifications - Chapter 13
	(8) operator requalifications program - Chapter 13.
50.34(c)&(d)	Industrial security program planning for CPSES, in accordance with 10CFR 73, is discussed in FSAR Section 13.6.
50.34a(a)	The design objectives and system description for the Liquid Waste Management System is provided in CPSES FSAR Section 11.2.1. The design objectives and system description for the Gaseous Waste Management Systems are provided in Section 11.3.1.
50.34a(b)	Same as for 50.34a(a). Expected releases of radio-nuclides in liquids is provided in FSAR Section 11.2.3. The estimated gaseous releases are provide in Section 11.3.3.
50.34a(c)	See compliance to 50.34a(a) and 50.34a(b).

TABLE Q400.4-1 (Sheet 12 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
50.35(a),(b)&(c)	These regulations are relevant to the construction permit stage which is completed for CPSES.
50.36(a)	Technical specifications for CPSES shall comply with STANDARD TECHNICAL SPECIFICATIONS FOR WESTINGHOUSE PRESSURIZED WATER REACTORS, NUREG-0452, Revision 3. See CPSES FSAR Section 16.2.
50.36(b)	The CPSES technical specifications shall be derived from analyses and evaluations included in the FSAR.
50.36(c)	The CPSES technical specifications shall include those pertinent items of NUREG-0452, Revision 3 and additional technical specifications that may be appropriate. (per 10CFR 50.36(b))
50.36(d)	The CPSES Technical specification will be issued by the Commission as Appendix "A" of the operating license.
50.36a(a)&(b)	Technical specifications for CPSES shall comply with NUREG-0452, Rev. 3. They shall be issued as Appendix "A" of the Operating License.
50.37	TUGCO agrees to limit access to restricted data as required in this regulation. See the CPSES Application for Operating License, Part 10 (page 23).
50.38	This regulation prohibits the NRC from issuing a license to foreign-controlled entities. TUGCO's statement that it is not owned, controlled, or dominated by an alien, foreign corporation, or foreign government is in Part 1 of the operating license application for Comanche Peak Steam Electric Station at page 4.
50.39	This regulation provides that applications and related documents may be made available for public inspection. This imposes no direct obligations on applicants and licensees.

TABLE Q400.4-1 (Sheet 13 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
50.40(a)	The design and operation of the facility is to provide reasonable assurance that the applicant will comply with NRC regulations, including those in 10 CFR Part 20, and that the health and safety of the public will not be endangered. The basis for TUGCO's assurance that the regulations will be met and the public protected is contained in this Attachment and in the license application and the related correspondence over the years.
50.40(b)	TUGCO's technical qualifications and its financial qualifications were examined in detail by the NRC staff.
50.40(c)	The CPSES construction permits CPPR-126 and CPPR-127, parts 1H, state that the issuance of the permit "will not be inimical to the common defense and security or to the health and safety of the public." To TUGCO's knowledge this conclusion has not been challenged, and no evidence contrary to it has been presented.
50.40(d)	The applicable requirements of PART 51 have been satisfied in the CPSES Environmental Report.
50.41(a),(b),&(c)	This regulation applies to class 104 licenses. TUGCO has applied for a class 103 license. PART 50.41 is not applicable.
50.42(a)	CPSES and its facilities will be used for the generation of commercial electric energy for transmission and sale within the State of Texas. See the CPSES Application for operating license, part 5, page 18.
50.42(b)	The advice of the Attorney General in regards to antitrust is presented in CPSES Construction permits CPPR-126 and CPPR-127 parts 3D(2).
50.43(a),(b),(c)&(d)	This regulation imposes certain duties on the NRC and addresses the applicability of the Federal Power Act and the right of government agencies to obtain NRC licenses. It imposes no direct obligations on licensees.

TABLE 0400.4-1 (Sheet 14 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
50.44(a)	CPSES FSAR Section 6.2.5 describes the Combustible Gas Control System.
50.44(b)	The Containment Hydrogen Monitoring System described in CPSES FSAR Section 6.2.5.2.3 is provided for measuring hydrogen concentration in the containment. POST-LOCA Hydrogen mixing is discussed in FSAR Sections 6.2.2.2 and 6.2.5.3.2. Controlling combustible gases in the containment following a Loss of Coolant Accident is accomplished by electric Hydrogen Recombiners (FSAR Section 6.2.5.2.1) and the Hydrogen Purge System (FSAR Section 6.2.5.2.2).
50.45	This regulation provides standards for construction permits and is not material to this operating license proceeding.
50.46 (a),(b),(c)(d)	As discussed in FSAR Section 6.3 the ECCS will meet the Acceptance Criteria as presented in this regulation. That is: 1. The calculated peak fuel element clad temperature provider margin to the requirement of 2200oF. 2. The amount of fuel element cladding that reacts chemically with water does not exceed 1 percent of the total amount of Zircaloy in the reactor. 3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The cladding oxidation limit of 17 percent are not exceeded during or after quenching. 4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long lived radioactivity remaining in the core.
50.47(a)	This regulation sets forth the standards for the Emergency Plan.

TABLE Q400.4-1 (Sheet 15 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
50.47(b),(c)	Compliance with these regulations is being achieved by the submission of the CPSES Emergency Plan dated October 8, 1980 and revised March 31, 1981.
50.48(a),(b),(c) (d)&(e)	The CPSES Fire Protection Program is described in FSAR Section 9.5.1.
50.50	This regulation provides that the NRC will issue a license upon determining that the application meets the standards and requirements of the Atomic Energy Act and the reguations and that the necessary notifications to other agencies or bodies have been duly made. It imposes no direct obligations on licensees.
50.51	This regulation specifies the maximum duration of licenses. Compliance will be affected simply by the Commission's writing the license so as to comply.
50.52	This regulation provides for the combining in a single license of a number of activities. It imposes no independent obligation on the licensee.
50.53	This regulation provides that licenses are not to be issued for activities that are not under or within the jurisdiction of the United States. The CPSES is within the United States and subject to the jurisdiction of the United States.
50.54(a)	[Deleted, Mary 9, 1967]
50.54(b) thru (u)	These regulations specify conditions that are incorporated in every license issued. Compliance is effected by including these conditions in a license when issued.
50.55(a),(b),(c) (d)&(e)	These regulations provide conditions for construction permits and are not material to the CPSES operating license procedings.

TABLE Q400.4-1 (Sheet 16 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
50.55a(a)	Sturctures, systems and components for the CPSES are designed, fabricated, constructed, tested and inspected to quality standards commensurate with their importantance. Specifics are provided in CPSES FAR Table 17A-1.
50.55a(b)	This regulation identifies pertinent code sections, editions and addendas.
50.55a(c)	The design and fabrication of the Reactor Coolant System components was completed in accordance with ASME B&PV Code Section III. The applicable addenda for each components are provided in CPSES FAR Table 5.2-1.
50.55a(d)	Reactor coolant system piping is designed and fabricated in accordance with ASME III, 1974 edition thru summer 1974 addenda.
50.55a(e)	Reactor Coolant pumps are designed and fabricated in accordance with ASME III, 1971 edition thru summer 1973 addenda.
50.55a(f)	Valves within the Reactor Coolant System Pressure boundary are designed and fabricated in accordance with ASME III. For the applicable editions and addenda of these valves see CPSES FSAR Table 5.2-1.
50.55a(g)	A response to NRC question 121.9 has been included in the CPSES FSAR in amendment 19. The response references changes to FSAR sections 5.2.4, 6.6.2 and 6.6.5, and Letter TXX-3277. These changes and the letter are about the CPSES Preservice Inspection Plan.
50.55a(h)	The CPSES Protection System is provided in Chapter 7 of the FSAR.
50.55a(i)	Assurance of adequate fracture toughness of ferritic materials in the reactor coolant pressure boundary is provided by compliance with the requirements for fracture toughness testing included in NB-2300 of ASME Section III and 10 CFR 50 Appendix G (see FSAR Section 5.3.1.5).

TABLE Q400.4-1 (Sheet 17 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
	The Material Surveillance program will conform with ASTM-E-185, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," and 10 CFR 50, Appendix H (see FSAR section 5.3.1.6).
50.55a(j)	This regulation applies to power reactors for which a notice of hearing on an application for a provisional construction permit or a construction permit has been published on or before December 31, 1970. It is not applicable to CPSES.
50.55b	[Revoked effective October 25, 1978]
50.55e	This reglation applies to fuel reprocessing plants. It is not applicable to CPSES.
50.56	This regulation provides that the Commission will, in the absence of good cause shown to the contrary, issue an operating license upon completion of the construction of a facility in compliance with the terms and conditions of the construction permit. This imposes no independent obligations on the applicant.
50.57(a)	This regulation allows the commission to issue an operating license upon satisfactory findings which are defined in the regulation.
50.57(b)	This regulation provides the commission the ability to include appropriate provisions in each operating license.
50.57(c)	This regulation allows an applicant to make a motion in writing for a license authorizing low-power testing, and further operations short of full power. It allows the Director of Nuclear Reactor Regulation, under defined conditions, to issue a license for the requested operation.
50.57(d)	[Deleted]

TABLE Q400.4-1 (Sheet 18 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
50.58(a)&(b)	This regulation provides for the review and report of Advisory Committee on Reactor Safeguards. CPSES is presently in this stage of licensing.
50.59(a),(b)&(c)	These regulations define changes, tests and experiments for production and utilization facilities, and provides guidance for record keeping of such and licensing when required.
50.60	[Deleted effective March 3, 1975]
50.65	[Deleted February 17, 1978, effective May 3, 1978]
50.70(a)&(b)	These regulations require licensees to permit inspections of his records, premises, activities and licensed materials.
50.71(a) thru (e)	This regulation address the maintenance of records and making of reports. CPSES will comply.
50.72(a)&(b)	Notification of significant events to the NRC will be made in accordance with this regulation.
50.78	This regulation requires holders of construction permits to submit installation information and permit verification by the International Atomic Energy Agency, if requested by the commission.
50.80(a),(b)&(c)	This regulation allows for transferring, assigning or disposing of a license with the permission of the Commission in writing. CPSES has not applied for such transfer.
50.81(a) thru (d)	This regulation permits the creation of mortgages, pledges, and liens on licensed facilities, subject to certain provisions. It prohibits secured creditors from violating the Atomic Energy Act and the Commission's regulations.

TABLE Q400.4-1 (Sheet 19 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
50.82(a)&(b)	This regulation provides for the termination of licenses. It is not applicable to the application for license of CPSES.
50.90	This regulation allows a holder of a construction permit or license to apply for amendment of the permit or license.
50.91	This regulation provides the Commission guidance in issuing license amendments.
50.100 50.101 50.102 50.103	These regulations govern the revocation, suspension, and modification of licenses by the commission under defined unusual circumstances.
50.109(a),(b)&(c)	This regulation specifies the conditions under which NRC may require the backfitting of a facility. It imposes no independent obligations on licensees.
50.110	This regulation governs enforcement of the Atomic Energy Act, the Energy Reorganization Act of 1974, and the NRC's regulations and orders.
Appendix A	The Seismic and Geologic siting criteria for CPSES are extensively provided in FSAR Chapter 2.
GDC 1	Codes and standards utilized for the unit are specified throughout the FSAR. Chapter 17 describes the quality assurance program and the provisions for maintenance of records.
GDC 2	FSAR Section 3.1.1.2 addresses the design considerations for natural phenomena, which are described in detail in Chapters 2 and 3. Appropriate considerations have been made in the design basis for historical data, combined effects of normal and accident conditions with the effects of natural phenomena, and the importance of the safety functions to be performed.

TABLE Q400.4-1 (Sheet 20 of 34)

TABULATION OF CPSES COMPLIANCE WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
GDC 3	FSAR Section 3.1.1.3 describes in general the measures which have been taken to minimize the probability and effects of fires and explosions. Section 9.5.1 describes the fire detection and protection systems.
GDC 4	FSAR Section 3.1.1.4 describes the design features used to accommodate the effects of and be compatible with the environmental conditions associated with all modes of operation and postulated accidents. Chapter 3 provides information concerning the specific design features for protection against missiles, jet impingement and pipe rupture.
GDC 5	As described in FSAR Section 3.1.1.5, those structures, systems and components which are shared with Unit 1 are tabulated in FSAR Section 1.2.2.11. It is concluded that safety functions are not significantly impaired by such sharing.
GDC 10	FSAR Section 3.1.2.1 indicates that the reactor core and associated systems are designed to function throughout the design lifetime without exceeding fuel damage limits.
GDC 11	FSAR Section 3.1.2.2 indicates that prompt compensatory reactivity feedback effects are assured by unit design and operational limit considerations. The core inherent reactivity feedback characteristics and reactivity control methods are described in FSAR Sections 4.3.
GDC 12	FSAR Section 3.1.2.3 describes the inherent and design features which eliminate or limit the various types of oscillations. Core stability is further described in Section 4.3.
GDC 13	As indicated in FSAR Section 3.1.2.4, and described in more detail in Chapter 7, instrumentation and control systems have been provided to monitor and maintain plant variables including those variables which affect the fission process, integrity of the reactor core, the reactor coolant pressure boundary, and the containment, over their

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
	prescribed ranges for normal operation, anticipated occurrences, and under accident conditions.
GDC 14	FSAR Section 3.1.2.5 indicates that the reactor coolant pressure boundary has been designed to accommodate the system temperatures and pressures attained under all expected operational modes and anticipated transients, and to maintain stresses within applicable limits.
GDC 15	As indicated in FSAR Section 3.1.2.6, the reactor coolant system and associated auxiliary, control and protection systems are designed to ensure the integrity of the reactor coolant pressure boundary with adequate margins during normal operations and anticipated transients. The design codes used for the Reactor Coolant System are described in Chapter 5.
GDC 16	As described in FSAR Section 3.1.2.7 and Chapter 6, a reinforced concrete, steel-lined containment structure, is provided. It is designed to sustain, without loss of required integrity, all effects of gross equipment failures, up to and including the rupture of the largest pipe in the Reactor Coolant System. The containment and its associated Engineered Safety Features thus meet the required functional capability of this criteria.
GDC 17	As described in FSAR Section 3.1.2.8, onsite and offsite power systems are provided which can independently supply the electric power required for the operation of safety related systems. This capability is maintained even with the failure of any single active component in either system. Chapter 8 provides the design details of the power systems and their compliance with this criterion.
GDC 18	As described in FSAR Section 3.1.2.9 the redundant electric power systems important to safety are continuously monitored and energized during normal plant operation from redundant

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
	offsite power sources. Redundant onsite diesel generators provide automatic backup power sources. Periodic tests of the diesel generators, the transfer system and the station batteries will be made.
GDC 19	FSAR Section 3.1.2.10 describes the Main Control Room, which contains the controls and instrumentation necessary for safe operation of the unit during normal and accident conditions. Sufficient shielding, distance, structural integrity, and ventilation systems are provided to ensure that control room personnel will not receive radiation exposures in excess of the criterion for the duration of the accident. In the event that access to the main control room is restricted, auxiliary shutdown panels have been provided, within the protected envelope, which may be used to maintain the reactor in a hot shutdown condition. Subsequent cold shutdown may be achieved using suitable procedures.
GDC 20	FSAR Section 3.1.3.1 discusses the design criteria for the protection system and engineered safety features actuation, to ensure that the requirements of this criterion are met. Further details are supplied in FSAR Chapter 7.
GDC 21	As indicated in FSAR Section 3.1.3.2, the protection system is designed for the high functional reliability and inservice testability commensurate with the safety functions to be performed. This section, as well as Sections 7.2.2.2.3 and 7.3.2.2.5, describe in detail the design features provided to ensure redundancy and testability.
GDC 22	FSAR Section 3.1.3.3 indicates that the protection system has been designed to provide sufficient resistance to a broad class of accident conditions or postulated events. Section 7.3 provides further design details concerning this resistance such that independence is maintained.

TABLE Q400.4-1 (Sheet 23 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
GDC 23	As indicated in FSAR Section 3.1.3.4, the protection system is designed with due consideration of the most probable failure modes of the components under various perturbations of energy sources and the environment. Further details are supplied in Sections 7.2 and 7.3.
GDC 24	FSAR Section 3.1.3.5 discusses separation of the protection and control systems, such that the failure of any single control system component or channel or the failure or removal from service of any protection system component or channel which is common to the protection and control systems, leaves intact a system satisfying all redundancy, reliability, and independence requirements of the protection system. Details concerning separation of protection and control systems are provided in FSAR Chapter 7.
GDC 25	FSAR Section 3.1.3.6 indicates that the protection system has been designed to assure that specified acceptable fuel design limits are not exceeded in the event of any single reactivity control system malfunction, including an accidental withdrawal of control cluster groups. Further details are provided in FSAR Chapter 15.
GDC 26	As indicated in FSAR Section 3.1.3.7, two independent reactivity control systems of different design principles are provided. One of the systems uses control rods; the second system employs dissolved boron as a chemical shim.
GDC 27	As described in FSAR Section 3.1.3.8, means are provided for shutdown reactivity for cooling the core under any anticipated condition and with appropriate margin for contingencies. Combined use of rod cluster control and chemical shim control permit the necessary shutdown margin to be maintained during the long term xenon decay and plant cooldown. These means are discussed in detail in FSAR Chapters 4 and 9.

TABLE Q400.4-1 (Sheet 24 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
GDC 28	FSAR Section 3.1.3.9 indicates that core reactivity is controlled by a chemical poison dissolved in the coolant, rod cluster assemblies and burnable poison rods. The maximum reactivity insertion rates due to withdrawal of a bank of rod cluster control assemblies or by boron dilution are limited. The maximum worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to values which prevent rupture of the coolant pressure boundary or disruption of the core internals to a degree which would impair core cooling capacity. Further details are provided in Section 4.3.
GDC 29	As indicated in FSAR Section 3.1.3.10, the protection and reactivity control systems are designed to assure extremely high probability of performing their required safety functions in the event of anticipated operational occurrences. The protection system is further discussed in Chapter 7.
GDC 30	As described in FSAR Section 3.1.4.1, reactor coolant pressure boundary components are designed, fabricated, inspected and tested in conformance with ASME Nuclear Power Plant Components Code Section III. Major components are classified as seismic Class 1 and are accorded the quality measures appropriate to this classification. The evaluations of reactor coolant pressure boundary components are discussed in Chapter 5.
GDC 31	As indicated in FSAR Section 3.1.4.2, close control is maintained over material selection and fabrication for the Reactor Coolant System to assure that the boundary behaves in a nonbrittle manner. The materials testing is consistent with 10CFR50, Appendices G and H. These tests ensure the selection of materials with proper toughness properties and margins as well as verify the integrity of the reactor coolant pressure boundary. Operating procedures and Technical Specifications ensure operation within the pressure-temperature limit relative to this criterion.

TABLE Q400.4-1 (Sheet 25 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
GDC 32	FSAR Section 3.1.4.3 describes how the design of the reactor vessel and its arrangement in the system provides the capability for accessibility during service life to the entire internal surface of the vessel and certain external zones of the vessel. The reactor arrangement within the containment provides sufficient space for inspection of the external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete.
GDC 33	As indicated in FSAR Section 3.1.4.4, the Chemical and Volume Control System provides a means of reactor coolant makeup and adjustment of the boric acid concentration. A high degree of functional reliability and safe response to probable modes of failure is assured by provision of standby components. Details of system design are included in Sections 6.3 and 9.3 and details of the electrical power systems are given in Chapter 8.
GDC 34	FSAR Section 3.1.4.5 indicates that the Residual Heat Removal System, in conjunction with the steam and power conversion system, is designed to transfer the fission product decay heat and other residual heat from the reactor core within acceptable limits. Suitable redundancy is accomplished below 350°F with the two residual heat removal pumps with means available for draining and monitoring of leakage, two heat exchangers, and the associated piping and cabling. The Residual Heat Removal System is able to operate on either onsite or offsite electrical power. Suitable redundancy above 350oF is provided by the steam generators, auxiliary feed pumps, and attendant piping. Details of the Residual Heat Removal System design are in FSAR Section 5.4.7.
GDC 35	FSAR Section 3.1.4.6 describes the use of passive accumulators with two centrifugal charging pumps and two low head safety injection pumps to provide redundancy for

TABLE Q400.4-1 (Sheet 26 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
	failure of any component in any system. The primary function of the Emergency Core Cooling System is to deliver borated cooling water to the reactor core in the event of a loss-of-coolant-accident. This limits the fuel clad temperature and thereby ensures that the core will remain substantially intact and in place, with its essential heat transfer geometry preserved. Further details are provided in Section 6.3.
GDC 36	As described in FSAR Section 3.1.4.7, design provisions are made for inspection to the extent practical of all components of the Emergency Core cooling System. An inspection is performed periodically to demonstrate system readiness. To the extent possible, the critical parts of the reactor vessel internals, injection nozzles, pipes, valves, and pumps are inspected visually or by boroscopic examination for erosion, corrosion, and vibration wear evidence. Non-destructive inspection is performed where such techniques are desirable and appropriate. Technical Specifications require inservice inspection in accordance with applicable ASME Codes. Details of the inspection programs are provided in Section 6.3.
GDC 37	FSAR Section 3.1.4.8 indicates that the components of the Emergency Core Cooling System located outside the containment will be accessible for leak-tightness inspection during appropriate periodic tests. Each active component of the system may be individually actuated on the normal power source at any time during plant operation to demonstrate operability. The centrifugal charging pumps are part of the charging system, and this system is in continuous operation during plant operation. Actuation circuits are tested and remote operated valves are exercised periodically. The testing is described in detail in FSAR Section 6.3.

TABLE Q400.4-1 (Sheet 27 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
GDC 38	As indicated in FSAR Section 3.1.4.9, two Quench Spray Subsystems and four separate Recirculation Spray Subsystems are provided to remove heat from the containment following a loss-of-coolant-accident. Each subsystem contains a separate pump and spray header, and each Recirculation Spray Subsystem contains a separate cooler. Two electrical buses, each connected to both onsite and offsite power, feed the pump motors and the necessary valves. Further details are provided in Sections 6.2.2 and 8.3.
GDC 45	As indicated in FSAR Section 3.1.4.16, the Service Water System and the Component Cooling System transfer heat from structures, systems, and components important to safety to an ultimate heat sink. Inspection of the majority of header piping in the Service Water System is not anticipated since it is buried and encased in concrete. All remaining piping, valves, equipment, and associated electrical gear from the Service Water System can be readily inspected. All piping, valves, equipment, and associated electrical gear for the Component Cooling System can be readily inspected. Further discussion of these systems is provided in FSAR Section 9.2.
GDC 46	As described in FSAR Section 3.1.4.17, the cooling water system referred to in this criterion encompasses the Service Water System and the Component Cooling System. The service water supply to the recirculation spray heat exchangers is tested periodically to ensure that the automatic valves function as required and the leak tight and structural integrity of the pressure containing components is retained. The Component Cooling system is in continuous use, thereby ensuring that the structural integrity, operability of active components, and operability of the system in its entirety is monitored continuously.

TABLE Q400.4-1 (Sheet 28 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
GDC 50	FSAR Section 3.1.5.1 indicates that the containment structure is designed to leak less than 0.1 percent by volume of its contents per day under post design basis accident conditions and to withstand pressures and temperatures above those that are conservatively calculated to result from a design basis accident by a sufficient margin to ensure that design conditions are not exceeded. Containment design basis is discussed further in Sections 3.8 and 6.2.1.
GDC 51	As discussed in FSAR Section 3.1.5.2, the design condition of the containment pressure boundary is based on the parameters derived after the design basis accident. For this design condition, the steel liner material behaves in a nonbrittle manner and has the capability to minimize the propagation of any undetected flaw. Detailed information on the steel liner material is found in Section 3.8.
GDC 52	As indicated in FSAR Section 3.1.5.3, the containment has been designed for periodic integrated leakage rate tests as required by 10CFR Appendix J.
GDC 53	FSAR Section 3.1.5.4 indicates that the containment design includes provisions for testing the leak tightness of all penetrations. Penetrations with resilient seals will be visually inspected and pressure tested. Penetrations with expansion bellows will be pressure tested. Test channels for checking the weld between penetrations and the containment liner have been provided, thereby allowing surveillance of the conditions inside the containment.
GDC 54	As described in FSAR Section 3.1.5.5, the Containment Isolation System provides at least two barriers between the atmosphere outside the containment structure and either the atmosphere inside the containment structure or the fluid inside the reactor coolant pressure boundary during accident conditions. Operation of the

TABLE Q400.4-1 (Sheet 29 of 34)

TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
	Containment Isolation System is automatic. The failure of one barrier or valve does not prevent isolation. Means are provided to periodically test the sensor set-points, the speed of response, the operability of fail-safe features, and the leakage rates of all valves used for containment isolation. Further discussion is provided in Section 6.2.4.
GDC 55	As indicated in FSAR Section 3.1.5.6, all pipe penetrations through the containment structure have at least two barriers between the atmosphere outside the containment and either the atmosphere inside the containment structure or the fluid inside the reactor coolant pressure boundary during accident conditions. The isolation valves outside the containment are located as close as practical to the penetration. All valves used to effect containment isolation may be readily inspected at any time. Additional details of the Containment Isolation System are provided in Section 6.2.4.
GDC 56	As indicated in FSAR Section 3.1.5.7, all pipe penetrations through the containment structure have at least two barriers between the atmosphere outside the containment and either the atmosphere inside the containment structure or the fluid inside the reactor coolant pressure boundary during structural accident conditions. The isolation valves outside the containment are located as close as practical to the penetration. All valves used to effect containment isolation may be readily inspected at any time. Additional details of the Containment Isolation System are provided in Section 6.2.4.
GDC 57	FSAR Section 3.1.5.8 indicates that each pipe penetration through the containment structure has at least two barriers between the atmosphere outside the containment and either the atmosphere inside the containment structure or the fluid inside the reactor coolant pressure boundary during accident conditions.

TABULATION OF CPSES COMPLIANCE
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Regulation (10CFR)	Compliance
	The isolation valves outside the containment are located as close as practical to the penetration. Further details regarding the Containment Isolation System are provided in FSAR Section 6.2.4.
GDC 60	As described in FSAR Section 3.1.6.1, the control of waste gas effluents is accomplished by holdup of waste gases in decay tanks. Sufficient holdup of waste gases is provided to cope with all anticipated operational occurrences and all site environmental conditions. Control of liquid waste effluents is accomplished by holdup of waste liquids in storage tanks. Liquid effluents are monitored for radioactivity and rate of flow. Sufficient holdup of liquid wastes is provided to cope with all anticipated operation occurrences and all site environmental conditions. Sufficient handling capacity is provided for solid wastes, which will be prepared batch-wise for offsite disposal, to cope with all anticipated operational occurrences. Further details are provided in FSAR Chapter 11.
GDC 61	FSAR Section 3.1.6.2 indicates that systems which may contain radioactivity are designed to ensure adequate safety under normal and postulated accident conditions. The systems are designed to permit inspection and testing as described in FSAR Chapters 5, 6, 9, and 11. The systems and components are provided with suitable shielding for radiation protection. To preclude gross mechanical failures which could lead to significant radioactivity release, appropriate containment, confinement, and treatment facilities and procedures are provided. Residual heat removal capability which has reliability and testability is provided. The fuel pit storage, fuel pit cooling, and fuel pit water makeup systems are designed to prevent significant reduction in fuel pit water inventory under accident conditions.

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TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
GDC 62	This criterion requires that criticality in the fuel storage and handling system be prevented by physical systems and processes, preferably by the use of geometrically safe configurations. As noted in FSAR Sections 3.1.6.3 and 9.1, the new and spent fuel handling, transfer and storage equipment and facilities are designed and arranged to provide sufficient center-to-center spacing to preclude criticality. The fuel transfer equipment is designed to handle only one fuel assembly at a time. The water used in the reactor cavity when the reactor vessel head is removed is maintained with a sufficient boron concentration to ensure a k effective less than 0.99 even if all control rods are withdrawn.
GDC 63	This criterion specifies that fuel storage and radioactive waste areas be monitored for loss of residual heat removal capability and for excessive radiation levels, and that appropriate safety actions be taken if such conditions arise. FSAR Section 3.1.6.4 and Chapters 9, 11 and 12 describes the monitoring capability in the fuel storage and waste handling areas.
GDC 64	This criterion requires means be provided for monitoring radioactive releases. FSAR Section 3.1.6.5 indicates that monitoring is provided for the containment atmosphere, effluent discharge paths, and the safeguards areas. Further details are provided in Chapters 11 and 12.
Appendix B	Chapter 17 of the FSAR describes in detail the provisions of the quality assurance program which has been implemented to meet all applicable requirements of Appendix B.
Appendix C	This Appendix provides a guide for establishing the applicant's financial qualifications. TUGCO's financial qualifications have been reviewed by the commission and reported in Section 20 of the Safety Evaluation Report, NUREG-0797.

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TABULATION OF CPSES COMPLIANCE
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Regulation (10CFR)	Compliance
Appendix D	This Appendix has been superseded by 10CFR Part 51. As noted in the discussion for 10CFR50.40(d), the requirements of Part 51 have been satisfied.
Appendix E	This Appendix specifies requirements for emergency plans.
Appendix F	This Appendix applies to fuel reprocessing plants and related waste management facilities, not to power reactors and is not applicable.
Appendix G	Fracture toughness requirements of this Appendix and program requirements given in Appendix H form the basis for Technical Specification surveillance requirements dealing with the use of surveillance specimens. Additional information to demonstrate compliance can be found in FSAR Section 5.3.1.5, concerning the irradiation surveillance program.
Appendix H	Reactor vessel material surveillance program requirements are delineated in this part. Further information is provided in FSAR Section 5.3.1.6.
Appendix I	This Appendix provides numerical guides for design objectives and limiting conditions for operation to meet the criteria "as low as is reasonably achievable" for radioactive material in light-water-cooled nuclear power reactor effluents. The design objective for the operation of CPSES within the guidelines of this appendix are provided in FSAR Chapter 11.
Appendix J	Reactor containment leakage testing for water cooled power reactors is delineated in this Appendix. Information concerning compliance can be found in FSAR Chapter 3 and FSAR Chapter 6.
Appendix K	This Appendix specifies features of acceptable ECCS evaluation models. Details are discussed in the response for regulation 50.46.

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TABULATION OF CPSES COMPLIANCE WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
Appendix L	This Appendix covers information requested by the Attorney General for anti-trust review of license applications. As noted above, the anti-trust review for CPSES 1 and 2 took place at the construction permit stage.
Appendix M	This Appendix covers standardization of design and is not applicable to the CPSES 1 and 2 proceeding.
Appendix N	This Appendix covers standardization of nuclear power plant designs and is not applicable to CPSES 1 and 2.
Appendix O	This Appendix covers standardization of design and is not applicable to CPSES 1 and 2.
Appendix P	This Appendix is proposed, 39 Fed. Reg. 26293, and it applies to fuel reprocessing plants. It is not applicable to CPSES.
Appendix Q	This Appendix governs preapplication early review of site suitability issues and is not applicable to CPSES 1 and 2.
Appendix Q (Proposed)	This Appendix is only proposed, 39 Fed. Reg. 26297, and it would apply to fuel reprocessing plants.
Appendix R	The CPSES Fire Protection System is described in FSAR Section 9.5.1.
100.1(a)	This regulation states the purpose for the part 100 regulations. It imposes no obligations on licensees.
100.1(b)	This regulation identifies part 100 as an interim guide to identify factors to be considered in the evaluation of reactor sites. Any other factors to be considered for CPSES will be submitted to the commission to demonstrate applicability and significance.
100.2(a)&(b)	This regulation states the applicability of Part 100 to stationary power and test reactors of general type and design, for which

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TABULATION OF CPSES COMPLIANCE
WITH 10 CFR 20, 50 & 100

Regulation (10CFR)	Compliance
	experience has been developed. CPSES is such a reactor. This regulation imposes no obligations on licensees.
100.3(a),(b),(c) (d)&(e)	This regulation defines words and phrases used in Part 100. It does not impose obligations on licensees.
100.10	(a) The characteristics of the CPSES reactor design and proposed operation provided throughout the CPSES FSAR.
	(b) The CPSES site geography and demography are provided in FSAR section 2.1.
	(c) Physical characteristics for the CPSES site are found in the FSAR: 1. Meteorology - FSAR section 2.3 2. Hydrology - FSAR section 2.4 3. Geology and Seismology FSAR section 2.5
	(d) The physical characteristics of the site and the design of the reactor and engineered safeguards are provided in the CPSES FSAR. This regulation allows for accetability based on the combination of site characteristics and design. It does not impose obligation on the licensee.
100.11	(a) The determination of exclusion area, low population zone and population center distance are provided in CPSES FSAR section 2.1.1.2, 2.1.3.4 and 2.1.3.5 respectively.
	(b) The CPSES site will accomdate two units which are independent to the extent that an accident in one reactor will not initiate an accident in the other.
Appendix A	Appendix A to 10 CFR Part 100 provides seismic and geologic siting criteria for nuclear power plants. The compliance of CPSES site with this Appendix is thoroughly documented in the FSAR.