

Public Service
Electric and Gas
Company

Corbin A. McNeill, Jr.
Vice President -
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Public Service Electric and Gas Company P.O. Box 236, Hancocks Bridge, NJ 08038 609-339-4800

March 21, 1986

Director of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20814

Attention: Ms. Elinor Adensam, Director
Project Directorate 3
Division of BWR Licensing

Dear Ms. Adensam:

FINAL SAFETY ANALYSIS REPORT REVISIONS
HOPE CREEK GENERATING STATION
DOCKET NO. 50-354

Public Service Electric and Gas Company (PSE&G) hereby submits various revisions to the Hope Creek Generating Station (HCGS) Final Safety Analysis Report (FSAR). The attached revisions to the HCGS FSAR contain: 1) revisions to maintain FSAR consistency with the Technical Specifications; 2) revisions to reconcile as-built plant discrepancies; and 3) general changes to the FSAR text, tables and figures.

Attachment 1 provides a brief summary and explanation for each change while Attachment 2 contains the actual marked-up sections of the FSAR. These revisions will be incorporated in FSAR Amendment 15 after fuel load but are being filed now in order to accurately reflect the design and operation of HCGS and support the issuance of an operating license. In addition, an affidavit is provided to affirm that the matters set forth in this transmittal are true and accurate.

This submittal supplements similar transmittals from C.A. McNeill to E. Adensam dated March 3, 1986 and March 17, 1986.

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Director of Nuclear
Reactor Regulation

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3-21-86

Should you have any questions on the subject filing, do not hesitate to contact us.

Sincerely,

A handwritten signature in black ink, appearing to read "C. H. Wagner", with a long horizontal stroke extending to the right.

Affidavit
Attachments (2)

C D.H. Wagner
USNRC Licensing Project Manager

R.W. Borchardt
USNRC Senior Resident Inspector

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
DOCKET NO. 50-354

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

FINAL SAFETY ANALYSIS REPORT
REVISIONS

Public Service Electric and Gas Company (PSE&G) hereby submits various revisions to the Hope Creek Generating Station (HCGS) Final Safety Analysis Report (FSAR). These HCGS FSAR revisions consist of text changes to maintain FSAR consistency with the Technical Specifications, revisions to reconcile as-built plant discrepancies, and general revisions to the FSAR text, tables and figures.

The matters set forth in these revisions are true and accurate to the best of my knowledge, information, and belief.

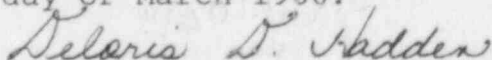
Respectfully submitted,
Public Service Electric
and Gas Company

By:



Corbin A. McNeill, Jr.
Vice President - Nuclear

Sworn to and subscribed
before me, a Notary Public
of New Jersey, this 21st
day of March 1986.



DELORIS D. HADDEN
A Notary Public of New Jersey
My Commission Expires March 14, 1990

ATTACHMENT 1

SUMMARY OF CHANGES, ADDITIONS AND/OR MODIFICATIONS

- 1.2.4.1.1 Revisions incorporate various references
1.2.5 to GESTAR.
1.3.1, 1.3.3
T1.3-1, sh. 1,2,3/6
T1.6-1, sh. 1-9/10
4.1.2.1.3
4.1.3.2
4.1.3.3
4.1.4.3
4.1.5, 4.2
4.3.2.4.2
4.3.2.5.1
4.3.5
4.4.2, 4.4.8
T4.4-1,10
6.3.6
15.0
T15.0-3, pg. 3/3
T15.0-4
F15.0-1
15.3.5, 15.4.2
15.4.10
- T3.11-5 Revisions to these two tables are necessary
pg. 6,10,16,21, to meet latest safety and hazard analysis
23,28,30,32-34, requirements for the Electrical Harsh
37,38,42-46, Environmental Program. The revisions
53,54,58,61-66, to T3.11-5, pg. 30/119 and T3.11-6, pg. 2/3
70,71,80,83, supercede those transmitted to the NRC
84,95,111, in a letter from C.A. McNeill to E. Adensam
118/119 dated March 17, 1986.
- T3.11-6
pg. 2/3
- 4.6.3.1.5 Revisions provide the commitments to
Q410.26 implement the requirements of NUREG-0803
regarding scram discharge volume piping
breaks. These revisions address License
Condition #6 contained in the Hope
Creek Draft Low Power License (NRC
letter to PSE&G dated March 17, 1986.)
- 7.3.1.1.1.2 Revision provides a description of
keylocked hand switches for local manual
control of certain ADS valves.

- 7.4.1.1.2 * Revision deletes the reactor vessel high water level automatic trip of the RCIC turbine and closure of the throttle valve. In actuality, the high water level trip is used to initiate closure of the RCIC steam supply valve to shut off the steam to the turbine and stop RCIC operation (pg. 7.4-3, Am. 14), Hence this revision is necessary to accurately reflect system design as summarized in the FSAR. This revision affects SER Section 7.4.1.1, pg. 7-37.
- 8.1.1 Revisions describe the addition of
8.2.1.4 the island substation and 13-kV feeders
8.2.1.5 to the substation.
F8.2-2
8.3.1.1.1
8.3.1.1.2.10
F8.3-1,2
- T8.3-1 Revisions reflect as-tested plant conditions.
pg. 2,4,10/10
- F8.3-16 Revision necessary to maintain consistency
sh. 7/8 with Technical Specification 4.8.2.1.d.
- T9.1-10 Revision deletes footnote (3) from Seismic Category I for Item 9 - Vacuum Breaker Valve Removal Hoist. This hoist is normally stored outside torus but brought in, installed and used during cold shutdown. Since shutdown cooling would not be lost if the hoist fell off the monorail during cold shutdown, the as-built hoist does not have to have seismic restraints.
- T9.3-5 Revision necessary to clarify surveillance
pg. 1/2 test prerequisites.
- 13.5 Revisions reflect the current station
13.5.1 review process for procedures as reflected
13.5.2.1.5 in the Technical Specifications and
17.2.2 plant administrative procedures. Revisions also reflect current procedure titles.
- 14.2.12.3.3 # Revisions reflect various power ascension
14.2.12.3.16 test program modifications approved
14.2.12.3.24 by the NRC in a letter to PSE&G dated
14.2.12.3.25 February 4, 1986. Revisions to Figure
14.2.12.3.28 14.2-5 supercede those provided to
14.2.12.3.32 the NRC in a letter from PSE&G dated
F14.2-5 March 3, 1986.

- 14.2.12.3.3.c Various revisions necessary to accurately
14.2.12.3.6.b reflect the startup test program.
14.2.12.3.8.b These revisions supplement similar
14.2.12.3.29.d changes previously submitted to the
14.2.12.3.33.b NRC from PSE&G on February 4, 1986
(FSAR Amendment 14).
- 14.2.12.3.12 Revision removes the reference to pump
oil temperature for the RCIC system
as such a temperature element is not,
nor need to be, provided in accordance
with General Electric requirements.

* These revisions impact the SER as noted

These revisions have already been accepted by the NRC
in a letter to PSE&G as noted.

ATTACHMENT 2

HCGS FSAR

The arrangement of structures on the site is shown on Figure 1.2-1. The general arrangement for the major power block structures is shown on Figures 1.2-2 through 1.2-11. The equipment arrangement for these structures is shown on Figures 1.2-12 through 1.2-43.

1.2.4 SYSTEM DESCRIPTION

A summary of the system description for Hope Creek Generating Station (HCGS) is provided below.

1.2.4.1 Nuclear System

The nuclear system includes a direct-cycle, forced-circulation, General Electric (GE) boiling water reactor (BWR) that produces steam for direct use in the steam turbine. A heat balance showing the major parameters of the nuclear system for the rated power conditions is shown on Figures 10.1-1 and 10.1-2.

1.2.4.1.1 Reactor Core and Control Rods

INSERT →

~~Fuel for the reactor core consists of slightly enriched uranium-dioxide pellets sealed in Zircaloy-2 tubes. These tubes (or fuel rods) are assembled into individual fuel assemblies.~~

~~Each fuel assembly has several fuel rods with gadolinia, Gd_2O_3 , mixed in solid solution with the UO_2 . The Gd_2O_3 is burnable-poison that diminishes the reactivity of the fresh fuel. It is considerably depleted as the fuel reaches the end of its first cycle.~~

~~Gross reactivity control of the core is achieved by movable, bottom-entry control rods. The control rods are cruciform in shape and are located throughout the lattice of fuel assemblies. The control rods are positioned by individual control rod drives (CRDs).~~

~~A conservative limit of plastic strain is the design criterion used for fuel rod cladding failure. The peak linear heat-generation for steady-state operation is well below the fuel damage limit, even late in plant life. Experience has shown that the control rods are not susceptible to distortion and have an~~

INSERT FOR PAGE 1.2-17

The reactor core and control rods are described in Section I and Appendix A, Subsection A.1.2.2.3.1 of Reference 1.2-1.

average life expectancy many times the residence time of a fuel loading.

1.2.4.1.2 Reactor Vessel and Internals

The reactor vessel contains the core and supporting structures, steam separators and dryers, jet pumps, control rod guide tubes, distribution lines for the feedwater, core sprays, core differential pressure and liquid control lines, in-core instrumentation, and other components. The main connections to the vessel include steam lines, coolant recirculation lines, feedwater lines, control rod drive (CRD) and in-core nuclear instrument housings, core spray lines, core differential pressure line, jet pump pressure sensing lines, water level instrumentation, and CRD system return lines (capped).

The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1250 psig. The nominal operating pressure in the steam space above the separators is 1020 psia. The vessel is fabricated of low alloy steel and is clad internally with stainless steel (except for the top head, which is not clad).

The reactor core is cooled by demineralized water that enters the lower portion of the core and boils as it flows upward around the fuel rods. The steam leaving the core is dried by steam separators and dryers located in the upper portion of the reactor vessel. The steam is then directed to the turbine through the main steam lines. Each steam line is provided with two automatic containment isolation valves in series; one on each side of the primary containment barrier.

1.2.4.1.3 Reactor Recirculation System

The reactor recirculation system consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each loop has one motor-driven recirculation pump powered and controlled by a dedicated motor-generator set located outside the primary containment. Recirculation pump speed can be varied, to allow some control of reactor power level through the effects of coolant flow rate on the moderator void content.

HCGS FSAR

effluents from the plant that are potentially radioactive are monitored.

1.2.4.8.2 Area Radiation Monitors

Area radiation monitoring systems alert plant and main control room personnel of excessive gamma radiation levels at various locations within the plant.

1.2.4.8.3 Site Environs Radiation Monitors

Radiation monitors are provided outside the plant structures to monitor radiation levels. The data obtained from these monitors are used to compute the onsite and offsite radiation levels due to the plant operations.

1.2.4.9 Shielding

Shielding is provided throughout the plant, as required, to reduce radiation levels to operating personnel and the general public within the applicable limits set forth in 10 CFR 20, 10 CFR 50, and 10 CFR 100. It is also designed to protect certain plant components from radiation exposures that could result in unacceptable alterations of material properties or activation.

1.2.5 REFERENCES

- 1.2-1 "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL, INCLUDING THE UNITED STATES SUPPLEMENT," NEDE-24011-P-A-7 AND NEDE-24011-P-A-7-US.

HCGS FSAR

1.3 COMPARISON TABLES

1.3.1 COMPARISONS WITH SIMILAR FACILITY DESIGNS

This section highlights the principal design features of the plant and compares its major features with those of other boiling water reactor (BWR) facilities. The design of this facility is based on proven technology obtained during the development, design, construction, and operation of BWRs of similar types. The data, performance characteristics, and other information presented here represent a current, firm design.

The following tables summarize the plant design characteristics of the Hope Creek Generating Station (HCGS), the Hatch Nuclear Plant, the Limerick Generating Station, and the Susquehanna Steam Electric Station:

<u>Table No.</u>	<u>System</u>
1.3-1	Comparison of Nuclear Steam Supply System Design Characteristics
1.3-2	Comparison of Power Conversion System Design Characteristics
1.3-3	Comparison of Engineered Safety Features and Auxiliary Systems Design Characteristics
1.3-4	Comparison of Containment Design Characteristics
1.3-5	Radioactive Waste Management Systems Design Characteristics
1.3-6	Comparison of Structural Design Characteristics
1.3-7	Comparison of Instrumentation and Electrical Systems Design Classifications

1.3.2 COMPARISON OF FINAL AND PRELIMINARY INFORMATION (FSAR)

All of the significant changes that have been made in the facility design since submission of the PSAR are listed in Table 1.3-8. Each item in Table 1.3-8 is cross-referenced to the appropriate portion of the FSAR that describes the changes and the bases for them.

1.3.3 REFERENCES
1.3-1 "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL," INCLUDING THE "UNITED STATES SUPPLEMENT," NEDE-24011-P-A-7 AND NEDE-24011-P-A-7-US. 1.3-1

THE FUEL DATA IN TABLE 1.3-1 IS PROVIDED IN REFERENCE 1.3-1.

TABLE 1.3-1

COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS(1)

	Hope Creek BWR 4/5 251-764	Hatch 1 BWR 4 218-560	Limerick BWR 4/5 251-764	Susquehanna BWR 4 251-764
<u>Thermal and Hydraulic Design</u> (Section 4.4)				
Rated power, Mwt	3293	2436	3293	3293
Design power, Mwt (ECCS design basis)	3430	2550	3435	3439
Steam flow rate, lb/h	14.159 E6	10.03 E6	14.156 E6	13.48 E6
Core coolant flow rate, lb/h	100.0 E6	78.5 E6	100.0 E6	100.0 E6
Feedwater flow rate, lb/h	14.127 E6	10.445 E6	14.117 E6	13.574 E6
System pressure, nominal in steam dome, psia	1020	1020	1020	1020
Average power density, kW/liter	48.7	51.2	48.7	48.7
Maximum linear heat generation rate, kW/ft	13.4	13.4	13.4	13.4
Average linear heat generation rate, kW/ft	5.34	7.11	5.3	5.34
Maximum heat flux, Btu/h-ft²	361,600	428,300	361,600	361,000
Average heat flux, Btu/h-ft²	144,100	164,700	143,700	144,100
Maximum UO₂ temperature, °F	3412	4380	3435	3330
Average volumetric fuel temperature, °F	2149	2781	2130	2130
Average cladding surface temperature, °F	566	558	566	558
Minimum critical power ratio	1.20	(*)	1.24	1.23
Coolant enthalpy at core inlet, Btu/lb	526.1	526.2	526.1	521.8
Core maximum exit voids within assemblies	77.1	79	77.1	76.00
Core average exit quality, % steam	14.1	12.7	14.1	13.2
Feedwater temperature, °F	419.9	387.4	420	383

TABLE 1.3-1 (cont)

	Hope Creek BWR 4/5 251-764	Hatch 1 BWR 4 218-560	Limerick BWR 4/5 251-764	Susquehanna BWR 4 251-764
<u>Design Power Peaking Factor</u> (Section 4.4)				
Maximum relative assembly power	1.40	1.40	1.40	1.40
Local peaking factor	1.15	1.24	(*)	1.15
Axial peaking factor	1.4	1.5	1.4	1.40
Total peaking factor	2.51	2.6	(*)	2.51
<u>Nuclear Design (First Core)</u> (Section 4.3)				
<u>Water/DO₂ volume ratio (cold)</u>	2.68	2.53	2.74	2.86

Reactivity with strongest control rod cut, k_{eff}

<0.99

<0.99

<0.99

Dynamic void coefficient (cc-c)

At core average voids, (%)

At rated output (%)

38.0

39.7

39.7

Fuel temperature doppler coefficient at rated output, (1/k) (dk/dt) (c/s)

-1.98 E-5

-1.76 E-5

-1.85 E-5

-1.85 E-5

-1.85 E-5

Initial average U-235 enrichment (wt %)

1.85

2.23

1.88

1.88

Initial cycle exposure, MWD/short ton

8100

9413

9600

9600

Core Mechanical Design
(Sections 4.2 and 4.6)

Fuel Assembly

<u>Number of fuel assemblies</u>	764	560	764	764
<u>Fuel rod array</u>	8x8	7x7	8x8	8x8
<u>Overall length, in.</u>	176	176	176	176
<u>Weight of UO₂ per assembly lb (pellet type)</u>	457	483.4 (dished)	456	458 (chamfered)
<u>Weight of fuel assembly, lb</u>	666	675	689	665

TABLE 1.3-1 (cont)

	Hole Creek DWR-475- 251-764	Hatch 1- DWR-4- 210-560	Simons DWR-475- 251-764	Susquehanna DWR-4- 251-764
Fuel Rods				
Number per fuel assembly	62	49	62	62
Outside diameter, in.	0.483	0.563	0.403	0.403
Cladding thickness, in.	0.012	0.032	0.032	0.032
Diametral gap, pellet to cladding, in.	0.009	0.012	0.009	0.009
Length of gas plenum, in.	10	16	9.48	10
Cladding material(2)	Zircaloy-2	Zircaloy-2	Zircaloy-2	Zircaloy-2
Fuel Pellets				
Material	UO ₂	UO ₂	UO ₂	UO ₂
Density, % of theoretical	95	95	94	95
Diameter, in.	0.410	0.487	0.410	0.410
Length, in.	0.410	0.5	0.410	0.410
Fuel Channel				
Overall length, in.	166.9	166.9	166.9	166.9
Thickness, in.	0.080	0.080	0.100	0.080
Gross section dimension, in.	5.44 x 5.44	5.44 x 5.44	5.48 x 5.48	5.48 x 5.48
Material	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4
Core Assembly				
Fuel weight as UO ₂ , lb	348,997	272,850	348,939	349,000
Core diameter (equivalent), in.	187.1	160.2	187.1	187.1
Core height (active fuel), in.	150	144	150	150

TABLE 1.6-1

Page 1 of 10

REFERENCED REPORTS

<u>Report Number</u>	<u>Title</u>	<u>Referenced in FSAR Section</u>
GENERAL ELECTRIC COMPANY REPORTS:		
APED-4827	Maximum Two-Phase Blowdown from Pipes, April 1965.	6.2
APED-5286	Design Basis for Critical Heat Flux Condition in BWRs, September 1966.	1.5
APED-5458	Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors, March 1968.	5.4
APED-5460	Design and Performance of General Electric BWR Jet Pumps, July 1968.	3.9
APED-5555	Impact Testing on Collet Assembly for Control Rod Drive Mechanism 7RDB144A, November 1967.	4.6
APED-5640	Xenon Considerations in Design of Large Boiling Water Reactors, June 1968.	4.1
APED-5750	Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves, March 1969.	5.4
APED-5756	Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor, March 1969.	15.4, 15.7
GEAP-4616	Two-Phase Pressure Drop in Straight Pipes and Channels: Water-Steam Mixtures at 600 to 1400 psia, May 1964.	4.4

TABLE 1.6-1 (cont)

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<u>Report Number</u>	<u>Title</u>	<u>Referenced in FSAR Section</u>
GEAP-5620	Failure Behavior in ASTM A1063 Pipes Containing Axial Through-Wall Flows, April 1968	5.2
GEAP-10546	Theory Report for Creep-Plast Computer Program, January 1972.	4.4
KAPL-2170	Hydrodynamic Stability of a Boiling Channel, October 1961.	4.4
KAPL-2208	Hydrodynamic Stability of a Boiling Channel, Part 2, April 1962.	4.4
KAPL-2290	Hydrodynamic Stability of a Boiling Channel, Part 3, June 1963.	4.4
KAPL-3070	Hydrodynamic Stability of a Boiling Channel, Part 4, August 1964.	4.4
KAPL-3072	Reactivity Stability of a Boiling Water Reactor, Part 1, September 1964.	4.4
KAPL-3093	Reactivity Stability of a Boiling Water Reactor, Part 2, March 1965.	4.4
NEDE-10313	PDA - Pipe Dynamic Analysis Program for Pipe Rupture Movement (Proprietary Filing).	3.6
NEDE-10958	General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, November 1973.	15.0
NEDE-10958A	General Electric Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application, January 1977	4.4
NEDE-20566-P-A	Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, November 1975.	4.2
NEDE-20944-P	BWR/4 and BWR/5 Fuel Design, Proprietary Versions, October 1976.	
NEDE-20944-P-1	BWR/4 and BWR/5 Fuel Design, Amendment 1, (only BWR/4&5,)	4.3 4.4

TABLE 1.6-1 (cont)

Report Number	Title	Referenced in FSAR Section
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January 1977.

NEDE-21156	Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibration, January 1976.	4.4
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NEDE-21354-P	BWR Fuel Channel Mechanical Design and Deflection, September 1976.	3.9
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NEDE-23014	HEX 01 User's Manual, July 1976.	15.2
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NEDE-23786-P	Fuel Rod Prepressurization, March 1978.	4.2
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REVISION 7

NEDE-24011-P-A	General Electric Standard Application for Reactor Fuel latest approved revision	4.4,
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1.2, 1.3, 4.1
4.2, 4.3,
6.3, 15.0,
15.3, 15.4

NEDE-24011-P-A-US	General Electric Standard Application for Reactor Fuel, United States Supplement latest approved revision	4.4,
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NEDE-24222	Assessment of BWR Mitigation of ATWS (NUREG-0460 Alternate No. 3), Volume 1, May 1979; Volume 2, December 1979.	15.8
------------	---	------

NEDE-24226-P	Evaluation of Control Blade Life with Potential Loss of B₄C, December 1979.	4.2
-------------------------	---	-----

NEDE-24834	Hanford 2 Crimped CRD Hydraulic Withdrawal Lines, (Proprietary).	3.6
------------	--	-----

NEDO- 10029	An Analytical Study on Brittle Fracture of GE-BWR Vessel Subjected to the Design Basis Accident, July 1969.	5.3
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NEDO-10173	Current State of Knowledge, High Performance BWR Zircaloy-clad UO ₂ Fuel, May 1970.	11.1
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NEDO-10299-A	Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello, October 1976.	4.4
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TABLE 1.6-1 (cont)

Page 4 of 10


<u>Report Number</u>	<u>Title</u>	<u>Referenced in FSAR Section</u>
NEDO-10320	The General Electric Pressure Suppression Containment Analytical Model, April 1971; Supplement 1, May 1971.	6.2
NEDO-10349	Analysis of Anticipated Transients Without Scram, March 1971.	15.8
NEDO-10505	Experience with BWR Fuel Through September 1971, May 1972.	11.1
NEDO-10527	Rod Drop Accident Analysis for Large Boiling Water Reactors, March 1972; Supplement 1, July 1972; Supplement 2, January 1973.	15.4
NEDO-10585	Behavior of Iodine in Reactor Water During Plant Shutdown and Startup, August 1972.	15.6
NEDO-10602	Testing of Improved Jet Pumps for the BWR/6 Nuclear System, June 1972.	3.9
NEDO-10739	Methods for Calculating Safe Test Intervals And Allowable Repair Times for Engineered Safeguard Systems, January 1973.	6.3, 15.9
NEDO-10801	Modeling the BWR/6 Loss-of-Coolant Accident: Core Spray and Bottom Flooding Heat Transfer Effectiveness, March 1973.	1.5
NEDO-10802	Analytical Methods of Plant Transient Evaluations for General Electric Boiling Water Reactor, February 1973.	4.1, 15.1 
NEDO-10846	BWR Core Spray Distribution, April 1973.	1.5
NEDO-10871	Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms, March 1975.	11.1
NEDO-10899	Chloride Control in BWR Coolants, June 1973.	5.2

TABLE 1.6-1 (cont)

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Report Number	Title	Referenced in FSAR Section
NEDO-10958	General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application, November 1973.	15.0
NEDO-10958-A	General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application, January 1977.	1.5, 4.4
NEDO-11209-0A	Nuclear Energy Business Operations Boiling Water Reactor Quality Assurance Program Description, Latest NRC-Accepted Revision	1.8
NEDO-12037	Summary of X-Ray and Gamma-Ray Energy and Intensity Data, January 1970.	12.3
NEDO-20231	Emergency Core Cooling Tests of an Internally Pressurized, Zircaloy-Clad, 8X8 Simulated BWR Fuel Bundle, December 1973.	1.5
NEDO-20360	General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel, May 1975.	15.4
NEDO-20566	General Electric Company Model for Loss-of-Coolant Accident Analysis in Accordance with 10 CFR 50, Appendix K, April 1977	1.5, 3.9, 6.3
NEDO-20626	Studies of BWR Designs for Mitigation of Anticipated Transients Without Scrams, October 1974.	15.8
NEDO-20651	BWR Radiation Effects Design Curve, March 1975.	5.3
NEDO-20922	Experience With BWR Fuel Through September 1974, June 1975.	11.1
NEDO-20944	BWR/4 and BWR/5 Fuel Design, October 1976.	4.1, 4.2, 4.3, 4.4
NEDO-20953	Three-Dimensional Boiling Water Reactor Core Simulator, May 1976.	15.4

TABLE 1.6-1 (cont)

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<u>Report Number</u>	<u>Title</u>	<u>Referenced in FSAR Section</u>
NEDO-20953-A	Three-Dimensional BWR Core Simulator, January 1977.	4.4
NEDO-20994	Peach Bottom Atomic Power Station Units 2 and 3, Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations, October 1975.	4.4
NEDO-21142	Realistic Accident Analysis for General Electric Boiling Water Reactor-The RELAC Code and User's Guide, January 1978.	15.4, 15.6, 15.7
NEDO-21143	Conservative Radiological Accident Evaluation The CONAC01 Code, March 1976.	15.4, 15.6, 15.7
NEDO-21159	Airborne Release from BWRs for Environmental Impact Evaluations, March 1976.	11.1, 12.3
NEDO-21159-2	Airborne Releases from BWRs for Environmental Impact Evaluations, 1977.	12.3
NEDO-21506	Stability and Dynamic Performance of the General Electric Boiling Water Reactor, January 1977.	4.1, 4.4
NEDO-21660	Experience with BWR Fuel through December 1976, July 1977.	11.1
NEDO-21778-A	Transient Pressure Rises Affecting Fracture Toughness Requirements for BWRs, December 1978.	5.3
NEDO-21821-2	Boiling Water Reactor Feedwater Nozzle/Sparger Final Report (Nonproprietary), August 1979.	5.3
NEDO-23786-1	Fuel Rod Prepressurization, Amendment 1, May 1978.	4.2
NEDO-24057-A	Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants, November 1977.	1.5, 3.9

TABLE 1.6-1 (cont)

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<u>Report Number</u>	<u>Title</u>	<u>Referenced in FSAR Section</u>
NEDO-24154	Qualification of the One-Dimensional Core Transient Model For BWR, October 1978.	4.1, 5.2 5.0 15.1
NEDO-24988	Analysis of Generic BWR Safety/ Relief Valve Operability Test Results, October 1981.	5.2

OTHER REFERENCED REPORTS

AEW-R-705	An Investigation Into the Effects of Crud Deposits on Surface Temperature, Dry-Out, and Pressure Drop, with Forced Convection Boiling of Water at 69 Bar in an Annular Test Section, 1971.	4.4
AE-RTL-788	Void Measurements in the Region of Subcooled and Low Quality Boiling, Part II, April 1966.	4.4
AI-75-2	Thermal Hydrogen Recombiner System for Water-Cooled Reactors, Revision 2, July 1975.	6.2
AI-77-55	Thermal Hydrogen Recombiner System for Mark I and II Boiling Water Reactors, September 1977.	6.2
ANL-5522	The Effect of Pressure on Boiling Density in Multiple Rectangular Channel, February 1956.	4.4
ANL-5621	Boiling Density in Vertical Rectangular Multichannel Sections with Natural Circulation, November 1956.	4.4
ANL-6385	Power-to-Void Transfer Functions, July 1961.	4.4
ANL-6948	Condensation of Metal Vapors: Mercury and the Kinetic Theory of Condensation, October 1964.	6.2

TABLE 1.6-1 (cont)

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<u>Report Number</u>	<u>Title</u>	<u>Referenced in FSAR Section</u>
MBI-1163	Vapor Formation and Behavior in Boiling Heat Transfer, February 1957.	4.4
BHR/DER 70-1	Radiological Surveillance Studies at a Boiling Water Nuclear Power Reactor, March 1970.	11.1
BMI-1163	Vapor Formation and Behavior in Boiling Heat Transfer, February 1957.	4.4
CF 59-6-47 (ORNL)	Removal of Fission Product Gases from Reactor Offgas Streams by Adsorption, 1959.	11.3
EPRI NP-495	Sources of Radioiodine at Boiling Water Reactors, February 1978.	12.2
ORNL-3041	SDC, A Shielding-Design Calculation for Fuel-Handling Facilities, March 1966.	12.3
ORNL-4585	Morse - A Multigroup Neutron and Gamma-Ray Monte Carlo Transport Code, September 1973.	12.3
ORNL-4628	Origen - The ORNL Generation and Depletion Code, May 1973.	12.3
ORNL-4932	Radioactive Atoms Supplement 1, August 1973.	12.3
ORNC-NSIC-23	Potential Metal-Water Reaction in Light-Water-Cooled Power Reactors, August 1968.	6.2
ORNL-RSIC-10	A Survey of Empirical Functions Used to Fit Gamma-Ray Buildup Factor, February 1966.	12.3
ORNL-RSIC-21	Neutron and Gamma-Ray Albedos, February 1968.	12.3
ORNL-TM-4280	The DOT 3 Two-Dimensional Discrete Ordinates Transport Code, September 1973.	12.3

TABLE 1.6-1 (cont)

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<u>Report Number</u>	<u>Title</u>	<u>Referenced in FSAR Section</u>
FC-4290	Hydrogen Evolution from Zinc Corrosion under Simulated Loss-of-Coolant Accident Conditions, August 1976.	6.2
STL-372-38	Kinetic Studies of Heterogeneous Water Reactors, April 1966.	4.4
WASH-1258	Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion as Low as Practicable for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents.	11.3
WCAP-8776	Corrosion Study for Determining Hydrogen Generation from Aluminum and Zinc During Post-Accident Conditions, 1976.	6.2
WAPD-TM-918	Thermal and Hydraulic Effects of Crud Deposited on Electrically Heated Rod Bundles, September 1970.	4.4
WAPD-BT-19	A Method of Predicting Steady-State Boiling Vapor Fractions in Reactor Coolant Channels, June 1960.	4.4
BECHTEL POWER CORPORATION REPORTS		
BC-TOP-4A	Seismic Analyses of Structures and Equipment for Nuclear Power Plants, Revision 3, November 1974.	3.7
BC-TOP-3A	Tornado and Extreme Wind Design Criteria for Nuclear Power Plants, Revision 3, August 1974.	3.3
BC-TOP-9A	Design of Structures for Missile Impact, Revision 2, September 1974.	3.5
BN-TOP-1	Testing Criteria for Integrated Leakage Rate Testing of Primary Containment Structures for Nuclear Power Plants, November 1972.	6.2

See Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

1.8.1.107 Conformance to Regulatory Guide 1.107, Revision 1, February 1977: Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures

Regulatory Guide 1.107 is not applicable to HCGS.

1.8.1.108 Conformance to Regulatory Guide 1.108, Revision 1, August 1977: Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants

HCGS complies with Regulatory Guide 1.108, with the following exception:

- a. During the preoperational test phase, following the diesel 24-hour full load test, the proper design accident loading sequence will be demonstrated by the test described in Section 14.2.12.1.47. This test will verify the ability of the SDG to start and accept the sequenced design loads as specified in Table 8.3-1. This test will provide ECCS flows to the reactor vessel.

INSERT →

- c.k. For periodic testing required by the Hope Creek Technical Specifications, the test per this regulatory position will be performed during shutdown. This test will simulate, separately, a loss of offsite power, and a loss of offsite power plus a LOCA condition, to verify the SDGs' ability to start and accept the sequenced design loads.

1.8.1.109 Conformance to Regulatory Guide 1.109, Revision 1, October 1977: Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I

HCGS complies with Regulatory Guide 1.109.

INSERT FOR PAGE 1.8-97

- b. The criteria regarding sustained load levels of 100% and 110% can be demonstrated when those significant parameters being measured have stabilized to acceptable values. Although the 110% load level should be maintained for 2 hours, reduced runtimes at 110% load levels are not regarded as an inadequate demonstration as defined by Position C.2.c provided: Runtime is sufficient to stabilize significant parameters being measured at the 110% load level and additional runtime (continuous beyond 22 hours) at the 100% load level is available to adequately demonstrate the diesel generator's load - carrying capability on an extended basis to compensate for the reduced runtime at the 110% load level.

HCGS PSAR
TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-11-1

SYSTEM: SAFETY AUX COOL SYSTEM
EG

P.O.	ID NO. Note (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	EES REF. NO.
				BLDG.	ELEV.			
J2010	1A-C-201		SACS Control Panel A	Reactor	102	No	No	26,27,68,69,70
J2010	1B-C-201		SACS Control Panel B	Reactor	102	No	No	26,27,68,69,70
J2010	1C-C-201		SACS Control Panel C	Reactor	102	No	No	26,27,68,69,70
J2010	1D-C-201		SACS Control Panel D	Reactor	102	No	No	26,27,68,69,70
E112AQ	1A-P-210		SACS Pump Motor	Reactor	102	No	No	1
E112AQ	1B-P-210		SACS Pump Motor	Reactor	102	No	No	1
E112AQ	1C-P-210		SACS Pump Motor	Reactor	102	No	No	1
E112AQ	1D-P-210		SACS Pump Motor	Reactor	102	No	No	1
P301Q	1-EG-SV-2290A		Solenoid Valve	Reactor	54	No	No	124
P301Q	1-EG-ZS-2290A		Position Switch	Reactor	54	No	No	125
P301Q	1-EG-SV-2290B		Solenoid Valve	Reactor	54	No	No	124
P301Q	1-EG-ZS-2290B		Position Switch	Reactor	54	No	No	125
P301Q	1-EG-SV-2290C		Solenoid Valve	Reactor	54	No	No	124
P301Q	1-EG-ZS-2290C		Position Switch	Reactor	54	No	No	125
P301Q	1-EG-SV-2290D		Solenoid Valve	Reactor	54	No	No	124
P301Q	1-EG-ZS-2290D		Position Switch	Reactor	54	No	No	125
P301Q	1-EG-SV-2290E		Solenoid Valve	Reactor	77	No	No	124
P301Q	1-EG-ZS-2290E		Position Switch	Reactor	77	No	No	125
P301Q	1-EG-SV-2290F		Position Switch	Reactor	77	No	No	124
P301Q	1-EG-ZS-2290F		Position Switch	Reactor	77	No	No	125
P301Q	1-EG-SV-2290G		Solenoid Valve	Reactor	54	No	No	124
P301Q	1-EG-ZS-2290G		Position Switch	Reactor	54	No	No	125

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TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

PK1D
M-11-1

SYSTEM: SAFETY AUX COOL SYSTEM
EG

P.O.	ID NO. Note (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	FFSS REF. NO.
				BLDG.	FLEV.			
M001	1-EG-PDT-2485A		Diff. Press Trans.	Reactor	102	No	No	155
M001	1-EG-PDT-2485B		Diff. Press Trans.	Reactor	102	No	No	155
M001	1-EG-PDT-2485C		Diff. Press Trans.	Reactor	102	No	No	155
M001	1-EG-PDT-2485D		Diff. Press Trans.	Reactor	102	No	No	155
J2010	1-EG-HS-2485A2		Hand Switch	Reactor	102	No	No	25
J2010	1-EG-HS-2485B2		Hand Switch	Reactor	102	No	No	25
J2010	1-EG-HS-2485C2		Hand Switch	Reactor	102	No	No	25
J2010	1-EG-HS-2485D2		Hand Switch	Reactor	102	No	No	25
P3050	1-EG-HV-2491A		Contr. Valve	Reactor	102	No	No	144
P3050	1-EG-ZS-2491A		Limit Switch	Reactor	102	No	No	144
P3050	1-EG-HV-2491B		Contr. Valve	Reactor	102	No	No	144
P3050	1-EG-ZS-2491B		Limit Switch	Reactor	102	No	No	144
P3050	1-EG-HV-2494A		Contr. Valve	Reactor	102	No	No	144
P3050	1-EG-ZS-2494A		Limit Switch	Reactor	102	No	No	144
P3050	1-EG-HV-2494B		Contr. Valve	Reactor	102	No	No	144
P3050	1-EG-ZS-2494B		Limit Switch	Reactor	102	No	No	144
P3050	1-EG-HV-2496A		Contr. Valve	Reactor	102	No	No	144
P3050	1-EG-ZS-2496A		Limit Switch	Reactor	102	No	No	144
P3050	1-EG-HV-2496B		Contr. Valve	Reactor	102	No	No	144
P3050	1-EG-ZS-2496B		Limit Switch	Reactor	102	No	No	144
P3050	1-EG-HV-2496C		Contr. Valve	Reactor	102	No	No	144
P3050	1-EG-ZS-2496C		Limit Switch	Reactor	102	No	No	144
P3050	1-EG-HV-2496D		Contr. Valve	Reactor	102	No	No	144
P3050	1-EG-ZS-2496D		Limit Switch	Reactor	102	No	No	144
J3010	1-EG-LT-2508A		Level Trans.	Reactor	201	No	No	29
J3010	1-EG-LT-2508H		Level Trans.	Reactor	201	No	No	29
J3010	1-EG-LT-2508C		Level Trans.	Reactor	201	No	No	29
J3010	1-EG-LT-2508D		Level Trans.	Reactor	201	No	No	29

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TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
~~XXXXXXXXXX~~
M-22-0

SYSTEM: DEMINERALIZED WATER
AN

P.O.	ID NO. Note (5)	MPL NO.	COMPONENT	BLDG.	LOCATION ELEV.	PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	FESS REF. NO.
						No	No	
P101Q	1-KC-HV-3408M		Control Valve	Reactor	77			127

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TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-25-0

SYSTEM: PLANT LEAK DETECTION
SK

P.O.	ID NO. Note (1)	MPL NO.	COMPONENT	LOCATION		P&M EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	FESS REF. NO.
				BLDG.	ELEV.			
J4810	1-EG-LSH-2359A		Level Sw. High	Reactor	102	No	No	42
J4810	1-EG-LSH-2359B		Level Sw. High	Reactor	102	No	No	42
J4810	1-EG-LSH-2364A		Level Sw. High	Reactor	102	No	No	42
J4810	1-EG-LSH-2364B		Level Sw. High	Reactor	102	No	No	42
J4810	1-ED-LSH-2365A		Level Sw. High	Reactor	77	No	No	42
J4810	1-ED-LSH-2365B		Level Sw. High	Reactor	77	No	No	42
J4810	1-ED-LSH-2365C		Level Sw. High	Reactor	77	No	No	42
J4810	1-BD-LE-4151-1		Level Element	Reactor	54	No	No	42
J4810	1-BD-LSH-4151-1		Level Sw. High	Reactor	54	No	No	42
J4810	1-BD-LE-4151-2		Level Element	Reactor	54	No	No	42
J4810	1-BD-LSH-4151-2		Level Sw. High	Reactor	54	No	No	42
J4810	1-BC-LE-4403A1		Level Element	Reactor	54	No	No	42
J4810	1-BC-LSH-4403A1		Level Sw. High	Reactor	54	No	No	42
J4810	1-BC-LE-4403A2		Level Element	Reactor	54	No	No	42
J4810	1-BC-LSH-4403A2		Level Sw. High	Reactor	54	No	No	42
J4810	1-BC-LE-4403B1		Level Element	Reactor	54	No	No	42
J4810	1-BC-LSH-4403B1		Level Sw. High	Reactor	54	No	No	42
J4810	1-BC-LE-4403B2		Level Element	Reactor	54	No	No	42
J4810	1-BC-LSH-4403B2		Level Sw. High	Reactor	54	No	No	42
J4810	1-BC-LE-4151-1		Level Sw. High	Reactor	54	No	No	42
J4810	1-BC-LE-4151-1		Level Element	Reactor	54	No	No	42
J4810	1-BC-LSH-4151-2		Level Sw. High	Reactor	54	No	No	42
J4810	1-BC-LE-4151-2		Level Element	Reactor	54	No	No	42

HCGS FSAR
TABLE 3.11-5

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EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-26-1

SYSTEM: RADIOLOGICAL MONITORING SYS
SP

P.O.	ID NO. Note (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	FESS REF. NO.
				BLDG.	ELEV.			
M001	1-SP-RE-N006A	D11	Radiation Element	Reactor	145	No	No	121
M001	1-SP-RE-N006B	D11	Radiation Element	Reactor	145	No	No	121
M001	1-SP-RE-N006C	D11	Radiation Element	Reactor	145	No	No	121
M001	1-SP-RE-N006D	D11	Radiation Element	Reactor	145	No	No	121
J1730	1-SP-RE-4825A		Radiation Element	Reactor	145	Yes	Yes	121
J1730	1-SP-RE-4825B		Radiation Element	Reactor	145	Yes	Yes	121
J1730	1-SP-RE-4856A		Radiation Element	Reactor	201	No	No	121
J1730	1-SP-RE-4856B		Radiation Element	Reactor	201	No	No	121
J1730	1-SP-RE-4856C		Radiation Element	Reactor	201	No	No	121
J1730	1-SP-RE-4857A		Radiation Element	Reactor	178	No	No	121
J1730	1-SP-RE-4857B		Radiation Element	Reactor	178	No	No	121
J1730	1-SP-RE-4857C		Radiation Element	Reactor	178	No	No	121
J1730	RM4	Rockbestos/Coax Cable GA		Note (3)		Yes	Yes	121

J1730	1-SP-RE-4825A		ION Chamber Detector	Reactor	145	Yes	Yes	35
J1730	1-SP-RE-4825B		ION Chamber Detector	Reactor	145	Yes	Yes	35
J1730	1-SP-RE-4856A		RFE Two Channel Duct Monitor	Reactor	201	No	No	36
J1730	1-SP-RE-4856B		RFE Two Channel Duct Monitor	Reactor	201	No	No	36
J1730	1-SP-RE-4856C		RFE Two Channel Duct Monitor	Reactor	201	No	No	36
J1730	1-SP-RE-4857A		RRE Two Channel Duct Monitor	Reactor	178	No	No	36
J1730	1-SP-RE-4857B		RRE Two Channel Duct Monitor	Reactor	178	No	No	36
J1730	1-SP-RE-4857C		RRE Two Channel Duct Monitor	Reactor	178	No	No	36
J1740	1-SP-FE-4811A		EVA Sensor	Reactor	178	Yes	No	37, 38
J1740	1-SP-FE-4811B		EVA Sensor	Reactor	178	Yes	No	37, 38

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 TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

 P&ID
 M-41-1

 SYSTEM: NUCLEAR BOILER
 AB

P.O.	ID NO. Note (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	EESS REF. NO.
				BLDG.	ELEV.			
P303AQ	1-AB-HV-F067D		Control Valve	Reactor	102	No	No	137
P303AQ	1-AB-ZS-F067D		Limit Switch	Reactor	102	Yes	No	137
P303AQ	1-AB-HV-F071		Control Valve	Reactor	102	No	No	139
P303AQ	1-AB-ZS-F071		Limit Switch	Reactor	102	No	No	139
P302Q	1-AE-ZS-F074A		Limit Switch	Reactor	102	Yes	No	134
P302Q	1-AE-ZS-F074B		Limit Switch	Reactor	102	Yes	No	134
M001	1-AB-PDT-N086A	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N086B	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N086C	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N086D	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N087A	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N087B	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N087C	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N087D	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N088A	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N088B	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N088C	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N088D	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N089A	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N089B	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N089C	B21	Press Diff. Trans.	Reactor	77	No	No	155
M001	1-AB-PDT-N089D	B21	Press Diff. Trans.	Reactor	77	No	No	155

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HCGS PSAR
TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-41-1SYSTEM: NUCLEAR BOILER
AB

P.O.	ID NO. Note (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TNI ACTION		EFSS REP. NO.
				BLDG.	ELEV.		PLAN EQUIP. NOTE (2)		
J5560	1-AB-TE-3648A		Temp. Elemt.	Reactor	54	Yes	No	43	
J5560	1-AB-TE-3648B		Temp. Elemt.	Reactor	54	Yes	No	43	
J5560	1-AB-TE-3648C		Temp. Elemt.	Reactor	54	Yes	No	43	
J5560	1-AB-TE-3648D		Temp. Elemt.	Reactor	54	Yes	No	43	
M001	1-SN-SV-3652A	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3652B	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3653A	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3653B	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3654A	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3654B	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3655A	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3655B	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3656A	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3657A	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3658A	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3659A	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3660A	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3661A	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3662A	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3663A	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3664A	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3665A	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
M001	1-SN-SV-3665B	B21-F013	Solenoid Valve	Reactor	121	No	No	159	
J8000	1-AB-XE-4507A		Acoustic Elemt.	Reactor	121	Yes	Yes	51	
J8000	1-AB-XT-4507A		Acoustic Trans.	Reactor	102	Yes	Yes	50	
J8000	1-AB-XE-4507B		Acoustic Elemt.	Reactor	121	Yes	Yes	51	
J8000	1-AB-XT-4507B		Acoustic Trans.	Reactor	102	Yes	Yes	50	
J8000	1-AB-XE-4507C		Acoustic Elemt.	Reactor	121	Yes	Yes	51	
J8000	1-AB-XT-4507C		Acoustic Trans.	Reactor	102	Yes	Yes	50	
J8000	1-AB-XE-4507D		Acoustic Elemt.	Reactor	121	Yes	Yes	51	
J8000	1-AB-XT-4507D		Acoustic Trans.	Reactor	102	Yes	Yes	50	
J8000	1-AB-XE-4507E		Acoustic Elemt.	Reactor	121	Yes	Yes	51	
J8000	1-AB-XT-4507E		Acoustic Trans.	Reactor	102	Yes	Yes	50	
J8000	1-AB-XE-4507F		Acoustic Elemt.	Reactor	121	Yes	Yes	51	
P30340	1-AE-HV-4444		CONTROL VALVE	REACTOR	102	No	No	139	

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TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-42-1SYSTEM: NUCLEAR BOILER VESSEL INSTRU
BH

P.O.	ID NO. Note (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	EISS REF. NO.
				BLDG.	ELEV.			
M001	1-SE-EAM-K002A	C51	Voltage Preamp	Reactor	102	No	No	154
M001	1-SE-EAM-K002B	C51	Voltage Preamp	Reactor	102	No	No	154
M001	1-SE-EAM-K002C	C51	Voltage Preamp	Reactor	102	No	No	154
M001	1-SE-EAM-K002D	C51	Voltage Preamp	Reactor	102	No	No	154
M001	1-SE-EAM-K002E	C51	Voltage Preamp	Reactor	102	No	No	154
M001	1-SE-EAM-K002F	C51	Voltage Preamp	Reactor	102	No	No	154
M001	1-SE-EAM-K002G	C51	Voltage Preamp	Reactor	102	No	No	154
M001	1-SE-EAM-K002H	C51	Voltage Preamp	Reactor	102	No	No	154
M001	1-SE-RE-N001A	C51	Radiation Elemt.	Reactor	121	Yes	No	162
M001	1-SE-RE-N001B	C51	Radiation Elemt.	Reactor	121	Yes	No	162
M001	1-SE-RE-N001C	C51	Radiation Elemt.	Reactor	121	Yes	No	162
M001	1-SE-RE-N001D	C51	Radiation Elemt.	Reactor	121	Yes	No	162
M001	1-SE-RE-N002A	C51	Radiation Elemt.	Reactor	121	Yes	No	103
M001	1-SE-RE-N002B	C51	Radiation Elemt.	Reactor	121	Yes	No	103
M001	1-SE-RE-N002C	C51	Radiation Elemt.	Reactor	121	Yes	No	103
M001	1-SE-RE-N002D	C51	Radiation Elemt.	Reactor	121	Yes	No	103
M001	1-SE-RE-N002E	C51	Radiation Elemt.	Reactor	121	Yes	No	103
M001	1-SE-RE-N002F	C51	Radiation Elemt.	Reactor	121	Yes	No	103
M001	1-SE-RE-N002G	C51	Radiation Elemt.	Reactor	121	Yes	No	103
M001	1-SE-RE-N002H	C51	Radiation Elemt.	Reactor	121	Yes	No	103
M001	1-BB-PT-N050A	C71	Pressure Transmitter	Reactor	162	No	No	155
M001	1-BB-PT-N050B	C71	Pressure Transmitter	Reactor	162	No	No	155
M001	1-BB-PT-N050C	C71	Pressure Transmitter	Reactor	162	No	No	155
M001	1-BB-PT-N050D	C71	Pressure Transmitter	Reactor	162	No	No	155
M001	1-BB-PT-N078A	B21	Pressure Transmitter	Reactor	77	No	No	155, ISSA
M001	1-BB-PT-N078B	B21	Pressure Transmitter	Reactor	77	No	No	155
M001	1-BB-PT-N078C	B21	Pressure Transmitter	Reactor	77	No	No	155

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TABLE 3.11-5

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EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-42-1

SYSTEM: NUCLEAR BOILER VESSEL INSTRU
BB

P.O.	ID NO. Note (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	EISS REF. NO.
				BLDG.	ELEV.			
M001	1-BB-PT-N078D	B21	Pressure Transmitter	Reactor	77	No	No	155
M001	1-BB-LT-N080A	B21	Level Transmitter	Reactor	77	No	Yes	155, ISSA
M001	1-BB-LT-N080H	B21	Level Transmitter	Reactor	77	No	Yes	155
M001	1-BB-LT-N080C	B21	Level Transmitter	Reactor	77	No	Yes	155
M001	1-BB-LT-N080D	B21	Level Transmitter	Reactor	77	No	Yes	155
M001	1-SM-LT-N081A	B21	Level Transmitter	Reactor	77	No	Yes	155, ISSA
M001	1-SM-LT-N081B	B21	Level Transmitter	Reactor	77	No	Yes	155, ISSA
M001	1-SM-LT-N081C	B21	Level Transmitter	Reactor	77	No	Yes	155
M001	1-SM-LT-N081D	B21	Level Transmitter	Reactor	77	No	Yes	155
M001	1-BB-LT-N085A	B21	Level Transmitter	Reactor	77	Yes	Yes	155
M001	1-BB-LT-N085B	B21	Level Transmitter	Reactor	77	Yes	Yes	155
M001	1-BB-PT-N090A	B21	Pressure Transmitter	Reactor	77	No	No	155, ISSA
M001	1-BB-PT-N090B	B21	Pressure Transmitter	Reactor	77	No	No	155
M001	1-BB-PT-N090E	B21	Pressure Transmitter	Reactor	77	No	No	155
M001	1-BB-PT-N090F	B21	Pressure Transmitter	Reactor	77	No	No	155, ISSA
M001	1-BE-PT-N090J	B21	Pressure Transmitter	Reactor	77	No	No	155
M001	1-BE-PT-N090K	B21	Pressure Transmitter	Reactor	77	No	No	155
M001	1-BE-PT-N090N	B21	Pressure Transmitter	Reactor	77	No	No	155, ISSA
M001	1-BE-PT-N090P	B21	Pressure Transmitter	Reactor	77	No	No	155, ISSA
M001	1-BB-LT-N091A	B21	Level Transmitter	Reactor	77	Yes	Yes	155, ISSA
M001	1-SB-LT-N091B	B21	Level Transmitter	Reactor	77	Yes	Yes	155
M001	1-SB-LT-N091C	B21	Level Transmitter	Reactor	77	No	Yes	155, ISSA
M001	1-SB-LT-N091D	B21	Level Transmitter	Reactor	77	No	Yes	155
M001	1-SB-LT-N091E	B21	Level Transmitter	Reactor	77	No	Yes	155
M001	1-SB-LT-N091F	B21	Level Transmitter	Reactor	77	No	Yes	155
M001	1-SB-LT-N091G	B21	Level Transmitter	Reactor	77	No	Yes	155
M001	1-SB-LT-N091H	B21	Level Transmitter	Reactor	77	No	Yes	155, ISSA
M001	1-BB-PT-N094A	B21	Pressure Transmitter	Reactor	162	No	No	155
M001	1-BB-PT-N079D	B21	PRESSURE TRANSMITTER	REACTOR	77	No	No	155
M001	1-BB-PT-N079H	B21	PRESSURE TRANSMITTER	REACTOR	77	No	No	155

HCCS ESAR
TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-42-1

SYSTEM: NUCLEAR BOILER VESSEL INSTRU
BB

P.O.	ID NO. Note (5)	MPL NO.	COMPONENT	LOCATION		PAH EQUIP. NOTE (1)	TMI ACTION PL/4 EQUIP. NOTE (2)	EESS REF. NO.
				BLDG.	ELEV.			
M001	1-BB-PT-N094B	B21	Pressure Transmitter	Reactor	162	No	No	155
M001	1-BB-PT-N094C	B21	Pressure Transmitter	Reactor	162	No	No	155
M001	1-BB-PT-N094D	B21	Pressure Transmitter	Reactor	162	No	No	155
M001	1-BB-PT-N094E	B21	Pressure Transmitter	Reactor	162	No	No	155
M001	1-BB-PT-N094F	B21	Pressure Transmitter	Reactor	162	No	No	155
M001	1-BB-PT-N094G	B21	Pressure Transmitter	Reactor	162	No	No	155
M001	1-BB-PT-N094H	B21	Pressure Transmitter	Reactor	162	No	No	155
M001	1-SN-LT-N095B	B21	Level Transmitter	Reactor	77	No	Yes	155
M001	1-SN-LT-N095D	B21	Level Transmitter	Reactor	77	No	Yes	155
M001	1-BB-LT-N097D	B21	Level Transmitter	Reactor	77	No	No	155
M001	1-BB-LT-N097H	B21	Level Transmitter	Reactor	77	No	No	155
M001	1-BB-LT-N402A	B21	Level Transmitter	Reactor	77	No	No	114, ISSA
M001	1-BB-LT-N402B	B21	Level Transmitter	Reactor	77	No	No	114
M001	1-BB-LT-N402E	B21	Level Transmitter	Reactor	77	No	No	114, ISSA
M001	1-BB-LT-N402F	B21	Level Transmitter	Reactor	77	No	No	114
M001	1-BB-PT-N403A	B21	Press Transmitter	Reactor	77	No	No	110, ISSA
M001	1-BB-PT-N403B	B21	Press Transmitter	Reactor	77	No	No	110
M001	1-BB-PT-N403E	B21	Press Transmitter	Reactor	77	No	No	110, ISSA
M001	1-BB-PT-N403F	B21	Press Transmitter	Reactor	77	No	No	110
M001	1-BB-RE-12D193	B11	Radiation Elemt.	Reactor	120	Yes	No	107
M001	1-BB-RE-13D193	B11	Radiation Elemt.	Reactor	120	Yes	No	107
M001	1-BB-RE-14D193	B11	Radiation Elemt.	Reactor	120	Yes	No	107
M001	1-BB-RE-15D193	B11	Radiation Elemt.	Reactor	120	Yes	No	107
M001	1-BB-RE-16D193	B11	Radiation Elemt.	Reactor	120	Yes	No	107
M001	1-BB-RE-21D193	B11	Radiation Elemt.	Reactor	120	Yes	No	107
M001	1-BB-RE-22D193	B11	Radiation Elemt.	Reactor	120	Yes	No	107

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EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-43-1SYSTEM: REACTOR RECIRC SYS
MH

P.C.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)		EISS REF. NO.
				BLDG.	ELEV.				
M001	1-BB-FT-N014A	B31	Flow Trans.	Reactor	77	Yes	No	155, ISSA	
M001	1-BB-FT-N014B	B31	Flow Trans.	Reactor	77	No	No	155	
M001	1-BB-FT-N014C	B31	Flow Trans.	Reactor	77	No	No	155	
M001	1-BB-FT-N014D	B31	Flow Trans.	Reactor	77	No	No	155, ISSA	
M001	1-BB-FT-N024A	B31	Flow Trans.	Reactor	77	Yes	No	155	
M001	1-BB-FT-N024B	B31	Flow Trans.	Reactor	77	No	No	155	
M001	1-BB-FT-N024C	B31	Flow Trans.	Reactor	77	No	No	155	
M001	1-BB-FT-N024D	B31	Flow Trans.	Reactor	77	No	No	155	
P101A0	1-BF-HV-3800A		Control Valve	Reactor	54	No	No	139	
P101A0	1-BF-ZS-3800A		Limit Switch	Reactor	54	Yes	No	139	
P101A0	1-BF-HV-3800B		Control Valve	Reactor	102	No	No	139	
P101A0	1-BF-ZS-3800B		Limit Switch	Reactor	102	Yes	No	139	
J6010	1-BB-SV-4310		Solenoid Valve	Reactor	121	No	No	46	
J6010	1-BB-ZS-4310		Position Switch	Reactor	121	Yes	No	46	
J6010	1-BB-SV-4311		Solenoid Valve	Reactor	145	No	No	46	
J6010	1-BB-ZS-4311		Position Switch	Reactor	145	Yes	No	46	

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TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-44-1

SYSTEM: REACTOR WATER CLEAN-UP
BG

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	THE ACTION PLAN EQUIP. NOTE (2)		FEES REF. NO.
				BLDG.	ELEV.				
P1020	1-BG-HV-F001		Contr. Valve	Reactor	145	No	No	132	
P1020	1-BG-ZS-F001		Limit Switch	Reactor	145	Yes	No	132	
P1020	1-BG-HV-F004		Contr. Valve	Reactor	145	No	No	132	
P1020	1-BG-ZS-F004		Limit Switch	Reactor	145	Yes	No	132	
P1020	1-BG-HV-F034		Contr. Valve	Reactor	77	No	No	132	
P1020	1-BG-ZS-F034		Limit Switch	Reactor	77	No	No	132	
P1020	1-BG-HV-F035		Contr. Valve	Reactor	77	No	No	132	
P1020	1-BG-ZS-F035		Limit Switch	Reactor	77	No	No	132	
P1020	1-AE-HV-F039		Contr. Valve	Reactor	102	No	No	132	
P1020	1-AE-ZS-F039		Limit Switch	Reactor	102	Yes	No	132	
M001	1-BG-FT-N012A	G33	Flow Trans.	Reactor	77	No	No	155	
M001	1-BG-FT-N012D	G33	Flow Trans.	Reactor	102	No	No	155	
M001	1-BG-FT-N036A	G33	Flow Trans.	Reactor	77	No	No	155	
M001	1-BG-FT-N036D	G33	Flow Trans.	Reactor	77	No	No	155, ISSA	
M001	1-BG-FT-N041A	G33	Flow Trans.	Reactor	102	No	No	155	
M001	1-BG-FT-N041D	G33	Flow Trans.	Reactor	102	No	No	155	

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EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-47-1

SYSTEM: CONTROL ROD DRIVE HYD-PART B
Bz

F.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		P&I EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	FESS REF. NO.
				BLDG.	ELEV.			
M001	1-BF-LS-N013D	C11	Level Switch	Reactor	102	No	No	151
M001	1-BF-LS-N013E	C11	Level Switch	Reactor	102	No	No	151
M001	1-BF-LS-N013F	C11	Level Switch	Reactor	102	No	No	151
M001	1-BF-LS-N013G	C11	Level Switch	Reactor	102	No	No	111
M001	1-BF-LS-N013H	C11	Level Switch	Reactor	102	No	No	111
M001	1-BF-SV-139	C11	Solenoid Valve	Reactor	102	No	No	122 Typical of 185 Solenoid Valves

HCCS FSAR
TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-48-1

SYSTEM: STANDBY LIQUID CONTROL
BH

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	EISS REF. NO.
				BLDG.	ELEV.			
M001	1A-P-208	C41 C001	Pump Motor	Reactor	162	No	No	104
M001	1B-P-208	C41 C001	Pump Motor	Reactor	162	No	No	104
M001	1-BH-XV-F004A	C41	Explosive Control Valve	Reactor	162	No	No	102
M001	1-BH-XV-F004B	C41	Explosive Control Valve	Reactor	162	No	No	102
P101AQ	1-BH-HV-F006A		Control Valve	Reactor	145	No	No	139
P101AQ	1-BH-ZS-F006A		Limit Switch	Reactor	145	Yes	No	139
P101AQ	1-BH-HV-F006B		Control Valve	Reactor	145	No	No	139
P101AQ	1-BH-ZS-F006B		Limit Switch	Reactor	145	Yes	No	139
M001	1-BH-PT-N004A	C41	Press Trans.	Reactor	162	Yes	No	115
M001	1-BH-PT-N004B	C41	Press Trans.	Reactor	162	Yes	No	115
M001	1-BH-LT-N010A	C41	Level Trans.	Reactor	162	No	No	113
M001	1-BH-LT-N010B	C41	Level Trans.	Reactor	162	No	No	113
M001	1-BH-LI-N010E	C41	Level Trans.	Reactor	162	No	No	113
M001	1-BH-LT-N010F	C41	Level Trans.	Reactor	162	No	No	113
MC 31	1-BH-HS-S4A	C41A	Hand Sw.	Reactor	162	No	No	152
M001	1-BH-HS-S4B	C41A	Hand Sw.	Reactor	162	No	No	152
M001	1-BH-LT-N012	C41	Level Trans.	Reactor	162	Yes	No	152

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EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P41D
M-49-1

SYSTEM: REACTOR CORE ISOL COOL.
BD

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)		FESS REF. NO.
				BLDG.	ELEV.				
P103AQ	1-FC-ZS-F025		Position Switch	Reactor	54	No	No	141	
P103AQ	1-FC-SV-F026		Solenoid Valve	Reactor	54	No	No	140	
P103AQ	1-FC-ZS-F026		Position Switch	Reactor	54	No	No	141	
P301Q	1-BD-HV-F031		Control Valve	Reactor	54	No	No	126	
P301Q	1-BD-ZS-F031		Limit Switch	Reactor	54	Yes	No	126	
J601Q	1-FC-SV-F054		Solenoid Valve	Reactor	54	No	No	44	
P301Q	1-FC-ZS-F054		Position Switch	Reactor	54	No	No	44	
P301Q	1-FC-HV-F059		Control Valve	Reactor	77	No	No	126	
P301Q	1-FC-ZS-F059		Limit Switch	Reactor	77	Yes	No	126	
P303AQ	1-FC-HV-F060		Control Valve	Reactor	54	No	No	138	
P303AQ	1-FC-ZS-F060		Limit Switch	Reactor	54	Yes	No	138	
P401Q	1-FC-HV-F062		Control Valve	Reactor	54	No	No	127	
P301Q	1-FC-ZS-F062		Limit Switch	Reactor	54	Yes	No	127	
P303AQ	1-FC-HV-F076		Control Valve	Reactor	100	No	No	139	
P303AQ	1-FC-ZS-F076		Limit Switch	Reactor	100	Yes	No	139	
P301Q	1-FC-HV-F084		Control Valve	Reactor	77	No	No	127	
P301Q	1-FC-ZS-F084		Limit Switch	Reactor	77	Yes	No	127	
M001	1-BD-FT-N003	E51	Flow Trans	Reactor	54	Yes	No	155	
M001	1-FC-PT-N007	E51	Press. Trans	Reactor	54	No	No	106	
M001	1-FC-LSH-N010	E51	Level Switch High	Reactor	54	No	No	150	
M001	1-BD-PT-N050	E51	Press. Trans	Reactor	54	No	No	155	
M001	1-BD-PT-N051	E51	Flow Trans	Reactor	54	No	No	155	
M001	1-FC-PDI-N057B	E51	Press. Diff. Trans	Reactor	77	No	No	155	
M001	1-FC-PDT-N057D	E51	Press. Diff. Trans	Reactor	77	No	No	155	

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TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-49-1

SYSTEM: REACTOR CORE ISOL COOL.
HD

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		P&I EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)		EISS REF. NO.
				BLDG.	ELEV.				
M001	1-FC-PT-N058B	E51	Press. Trans	Reactor	77	No	No	155	
M001	1-FC-PT-N058D	E51	Press. Trans	Reactor	77	No	No	155, 155A	
M001	1-FC-PT-N058F	E51	Press. Trans	Reactor	77	No	No	155	
M001	1-FC-PT-N058H	E51	Press. Trans	Reactor	77	No	No	155	
J3010	1-HD-PT-4157		Press. Trans.	Reactor	54	No	No	155 &	
J3010	1-FC-FT-4158		Flow Trans.	Reactor	54	No	No	156 29	

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TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-51-1

SYSTEM: RESIDUAL HEAT REMOVAL SYSTEM
BC

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	EESS REF. NO.
				BLDG.	ELEV.			
P103AQ	1-BC-ZS-F146A		Position Switch	Reactor	100	Yes	No	141
P103AQ	1-BC-SV-F146B		Solenoid Valve	Reactor	100	No	No	140
P103AQ	1-BC-ZS-F146B		Position Switch	Reactor	100	Yes	No	141
P103AQ	1-BC-SV-F146C		Solenoid Valve	Reactor	100	No	No	140
P103AQ	1-BC-ZS-F146C		Position Switch	Reactor	100	Yes	No	141
P103AQ	1-BC-SV-F146D		Solenoid Valve	Reactor	100	No	No	140
P103AQ	1-BC-ZS-F146D		Position Switch	Reactor	100	Yes	No	141
M001	1-BC-FT-N013	E11	Flow Transmitter	Reactor	102	No	No	106
M001	1-BC-FT-N015A	E11	Flow Transmitter	Reactor	77	Yes	No	155
M001	1-BC-FT-N015B	E11	Flow Transmitter	Reactor	77	Yes	No	155
M001	1-BC-FT-N015C	E11	Flow Transmitter	Reactor	54	Yes	No	155
M001	1-BC-FT-N015D	E11	Flow Transmitter	Reactor	54	Yes	No	155
M001	1-BC-FT-N052A	E11	Flow Transmitter	Reactor	77	No	No	155
M001	1-BC-FT-N052B	E11	Flow Transmitter	Reactor	77	No	No	155
M001	1-BC-FT-N052C	E11	Flow Transmitter	Reactor	54	No	No	155
M001	1-BC-FT-N052D	E11	Flow Transmitter	Reactor	54	No	No	155
M001	1-BC-PT-N053A	E11	Press. Trans	Reactor	54	No	No	106
M001	1-BC-PT-N053B	E11	Press. Trans	Reactor	54	No	No	106
M001	1-BC-PT-N053C	E11	Press. Trans	Reactor	54	No	No	106
M001	1-BC-PT-N053D	E11	Press. Trans	Reactor	54	No	No	106
M001	1-BC-PT-N055B	E11	Press. Trans	Reactor	54	No	No	155
M001	1-BC-PT-N055D	E11	Press. Trans	Reactor	54	No	No	155
M001	1-BC-PT-N055F	E11	Press. Trans	Reactor	54	No	No	155
M001	1-BC-PT-N055H	E11	Press. Trans	Reactor	54	No	No	155
M001	1-BC-PT-N056B	E11	Press. Trans	Reactor	54	No	No	155
M001	1-BC-PT-N056D	E11	Press. Trans	Reactor	54	No	No	155
M001	1-BC-PT-N056F	E11	Press. Trans	Reactor	54	No	No	155

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TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P410
M-51-1

SYSTEM: RESIDUAL HEAT REMOVAL SYSTEM
RC

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)		EES REF. NO.
				BLDG.	ELEV.				
M001	1-BC-PT-N056H	E11	Press. Trans	Reactor	54	No	No	155	
M001	1-BC-PT-N057	E11	Press. Trans.	Reactor	77	No	No	106	
M001	1-BC-PT-N058A	E11	Press. Diff. Trans.	Reactor	77	No	No	155	
M001	1-BC-PT-N058B	E11	Press. Diff. Trans.	Reactor	77	No	No	155	ISSA
M001	1-BC-PT-N058C	E11	Press. Diff. Trans.	Reactor	77	No	No	155	
M001	1-BC-PT-N058D	E11	Press. Diff. Trans.	Reactor	77	No	No	155	ISSA
M001	1-BC-PDT-N060A	E11	Press. Diff. Trans.	Reactor	77	No	No	106	
M001	1-BC-PDT-N060B	E11	Press. Diff. Trans.	Reactor	77	No	No	106	
F6H159	1-BC-TE-N027A		Temp. Elemt.	Reactor	77	Yes	No	34	
F6H159	1-BC-TE-N027B		Temp. Elemt.	Reactor	77	Yes	No	34	
J5560	1-BC-TE-4401		Temp. Elemt.	Reactor	77	No	No	43	
J1010	1-BC-FT-4435		Flow Trans.	Reactor	77	No	No	29	
P1010	1-BC-HV-4439		Control Valve	Reactor	77	No	No	127	
P1010	1-BC-ZS-4439		Limit Switch	Reactor	77	No	No	127	
J1010	1-BC-FT-4461A		Flow Trans.	Reactor	77	Yes	No	29	
J1010	1-BC-FT-4461B		Flow Trans.	Reactor	77	Yes	No	29	
J1010	1-BC-FT-4462A		Flow Trans.	Reactor	102	Yes	No	29	
J1010	1-BC-FT-4462B		Flow Trans.	Reactor	77	Yes	No	29	

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-52-1

SYSTEM: CORE SPRAY
BE

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION		EISS REP. NO.
				BLDG.	ELEV.		PLAN EQUIP. NOTE (2)		
M001	1-BE-FT-N051A	ES=	Flow Trans.	Reactor	54	No	No	155	
M001	1-BE-FT-N051B	E21	Flow Trans. Flow	Reactor	54	No	No	155	
M001	1-BE-PT-N054A	E21	Press Trans.	Reactor	54	No	No	155, ISSA	
M001	1-BE-PT-N054B	E21	Press Trans.	Reactor	54	No	No	155, ISSA	
M001	1-BE-PT-N055B	E21	Press. Trans.	Reactor	54	No	No	155	
M001	1-BE-PT-N055D	E21	Press. Trans.	Reactor	54	No	No	155	
M001	1-BE-PT-N055F	E21	Press. Trans.	Reactor	54	No	No	155	
M001	1-BE-PT-N055H	E21	Press. Trans.	Reactor	54	No	No	155	
M001	1-BE-PDT-N056	E21	Press. Diff. Trans.	Reactor	77	No	No	155, ISSA	

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TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-55-1

SYSTEM: HIGH PRESS COOL. INJECTION
HJ

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	EESS REF. NO.
				BLDG.	ELEV.			
P102Q	1-FD-HV-F001		Control Valve	Reactor	54	No	No	133
P102Q	1-FD-ZS-F001		Limit Switch	Reactor	54	No	No	133
P102Q	1-FD-HV-F002		Control Valve	Reactor	102	No	No	132
P102Q	1-FD-ZS-F002		Limit Switch	Reactor	100	Yes	No	132
P102Q	1-FD-HV-F003		Control Valve	Reactor	102	No	No	132
P102Q	1-FD-ZS-F003		Limit Switch	Reactor	102	Yes	No	132
P101Q	1-BJ-HV-F004		Control Valve	Reactor	54	No	No	126
P101Q	1-BJ-ZS-F004		Limit Switch	Reactor	54	No	No	126
P102Q	1-BJ-HV-F006		Control Valve	Reactor	102	No	No	133
P102Q	1-BJ-ZS-F006		Limit Switch	Reactor	102	Yes	No	133
P102Q	1-BJ-HV-F007		Control Valve	Reactor	54	No	No	133
P102Q	1-BJ-ZS-F007		Limit Switch	Reactor	54	No	No	133
P102Q	1-BJ-HV-F008		Control Valve	Reactor	77	No	No	133
P102Q	1-BJ-ZS-F008		Limit Switch	Reactor	77	No	No	133
P102Q	1-AP-HV-F011		Control Valve	Reactor	77	No	No	133
P102Q	1-AP-ZS-F011		Limit Switch	Reactor	77	No	No	133
P102Q	1-BJ-HV-F012		Control Valve	Reactor	54	No	No	133
P102Q	1-BJ-ZS-F012		Limit Switch	Reactor	54	Yes	No	133
P103AQ	1-FD-SV-F028		Solenoid Valve	Reactor	54	No	No	140
P103AQ	1-FD-ZS-F028		Position Switch	Reactor	54	No	No	141
P103AQ	1-FD-SV-F029		Solenoid Valve	Reactor	54	No	No	140
P103AQ	1-FD-ZS-F029		Position Switch	Reactor	54	No	No	141
P101Q	1-BJ-HV-F042		Control Valve	Reactor	54	Yes	No	126
P101Q	1-BJ-ZS-F042		Limit Switch	Reactor	54	Yes	No	126
J601Q	1-FD-SV-F054		Solenoid Valve	Reactor	54	No	No	44
J601Q	1-FD-ZS-F054		Position Switch	Reactor	54	No	No	45
P101Q	1-FD-HV-F071		Control Valve	Reactor	77	No	No	126

HCGS FSAR
TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

SYSTEM: HIGH PRESS COOL. INJECTION
BJ

P&ID
M-55-1

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	FESS REF. NO.
				BLDG.	ELEV.			
P301Q	1-FD-ZS-F071		Limit Switch	Reactor	77	Yes	No	126
P301Q	1-FD-HV-F075		Control Valve	Reactor	77	No	No	127
P301Q	1-FD-ZS-F075		Limit Switch	Reactor	77	Yes	No	127
P301Q	1-FD-HV-F079		Control Valve	Reactor	77	No	No	127
P301Q	1-FD-ZS-F079		Limit Switch	Reactor	77	Yes	No	127
P303AQ	1-FD-HV-F100		Control Valve	Reactor	100	No	No	139
P303AQ	1-FD-ZS-F100		Limit Switch	Reactor	100	Yes	No	139
M001	1-FD-FT-N008	E41	Flow Trans.	Reactor	54	Yes	No	155
M001	1-FD-PT-N013	E41	Press. Trans.	Reactor	54	No	No	106
M001	1-FD-LSH-N014	E41	Level Switch High	Reactor	54	No	No	150
M001	1-RJ-PT-N050	E41	Press. Trans.	Reactor	54	No	No	155
M001	1-BJ-FT-N051	E41	Flow Trans.	Reactor	54	No	No	155
M001	1-FD-PDT-N057A	E41	Press. Diff. Trans.	Reactor	77	No	No	155
M001	1-FD-PDT-N057C	E41	Press. Diff. Trans.	Reactor	77	No	No	155
M001	1-FD-PT-N058A	E41	Press. Transmitter	Reactor	77	No	No	155
M001	1-FD-PT-N058C	E41	Press. Transmitter	Reactor	77	No	No	155
M001	1-FD-PT-N058E	E41	Press. Transmitter	Reactor	77	No	No	155
M001	1-FD-PT-N058G	E41	Press. Transmitter	Reactor	77	No	No	155
M001	1-BJ-LT-N062A	E41	Level Transmitter	Reactor	54	No	No	120
M001	1-BJ-LT-N062E	E41	Level Trans.	Reactor	54	No	No	120
J301Q	1-BJ-PT-4771		Press Trans.	Reactor	54	No	No	28
J301Q	1-BJ-LT-4801		Level Trans.	Reactor	54	Yes	Yes	139
P303AQ	1-BJ-HV-4803		Control Valve	Reactor	54	No	No	139
P303AQ	1-BJ-ZS-4803		Limit Switch	Reactor	54	Yes	No	139
P303AQ	1-BJ-HV-4804		Control Valve	Reactor	54	No	No	139
P303AQ	1-BJ-ZS-4804		Limit Switch	Reactor	54	Yes	No	139
M001	1-BJ-LT-4805-1		Level Trans.	Reactor	54	Yes	No	155

ISSA
ISSA

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EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-55-1

SYSTEM: HIGH PRESS COOL. INJECTION
BJ

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	EES REP. NO.
				BLDG.	ELEV.			
M001	1-BJ-LT-4805-2		Level Trans.	Reactor	54	No	No	155, ISCA
P103AQ	1-BJ-HV-4865		Control Valve	Reactor	54	No	No	139
P103AQ	1-BJ-ZS-4865		Limit Switch	Reactor	54	Yes	No	139
P103AQ	1-BJ-HV-4866		Control Valve	Reactor	54	No	No	139
P103AQ	1-BJ-ZS-4866		Limit Switch	Reactor	54	Yes	No	139
P102Q	1-BJ-HV-8278		Control Valve	Reactor	102	No	No	133
P102Q	1-BJ-ZS-8278		Limit Switch	Reactor	102	Yes	No	133

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

SYSTEM: HPCI PUMP TURBINE
FD

P&ID
M-56-1

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)		EES REF. NO.
				BLDG.	ELEV.				
M001	10-S-211	E41-C002	HPCI Turbine	Reactor	54	No	No	123	
M001	10-P-213	E41-C002	Aux. Oil Pump Motor	Reactor	54	No	No	123	
M001	10-P-215	E41-C002	HPCI Vac Tank Cond. Pump	Reactor	54	No	No	123	
M001	10-P-216	E41-C002	HPCI Gland Seal Cond. Vac. Pump	Reactor	54	No	No	123	
M001	No Tag No.	E41-C002	HPCI Turbine Governor	Reactor	54	No	No	123	
M0820	1A-P-228		Jockey Pump Motor, ECCS	Reactor	54	No	No	123	
J601Q	1-FD-SV-F025		Solenoid Valve	Reactor	54	No	No	80	
J601Q	1-FD-2S-F025		Position Switch	Reactor	54	No	No	44	
P303AQ	1-FD-SV-F026		Solenoid Valve	Reactor	54	No	No	45	
P303AQ	1-FD-2S-F026		Position Switch	Reactor	54	No	No	140	
P303AQ	1-BJ-HV-F059		Control Valve	Reactor	54	No	No	141	
P303AQ	1-BJ-2S-F059		Limit Switch	Reactor	54	No	No	138	
M001	1-BJ-PT-N052	E41	Press. Trans.	Reactor	54	No	No	138	
M001	1-BJ-PT-N053	E41	Press. Trans.	Reactor	54	No	No	106	
M001	1-FD-PT-N055A	E41	Press. Trans.	Reactor	77	No	No	155, ISSA 155, ISSA	

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TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-56-1

SYSTEM: HPCI PUMP TURBINE
FD

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	EESS REF. NO.
				BLDG.	ELEV.			
M001	1-FD-PT-N055C	E41	Press. Trans.	Reactor	77	No	No	155, ISSA
M001	1-FD-PT-N055E	E41	Press. Trans.	Reactor	77	No	No	155
M001	1-FD-PT-N055G	E41	Press. Trans.	Reactor	77	No	No	155, ISSA
M001	1-FD-PT-N056A	E41	Pres. Trans.	Reactor	54	No	No	155
M001	1-FD-PT-N056E	E41	Press. Trans.	Reactor	54	No	No	155
M001	1-FD-FV-4879	E41	Flow Control Valve Actuator	Reactor	54	No	No	123
M001	1-FD-FV-4880	E41	Flow Control Valve Actuator	Reactor	54	No	No	123
M001	1-FD-LSH-4890	E41	Level Switch High	Reactor	54	No	No	123
J101Q	1-BJ-PT-4891		Press Trans.	Reactor	77	No	No	29
M001	1-FD-LSL-4903	E41-C002	Level Switch Low	Reactor	54	No	No	123
M001	1-FD-PSH-4905	E41-C002	Press Switch High	Reactor	54	No	No	123
M001	1-FD-ZS-4907	E41-C002	Limit Switch	Reactor	54	No	No	123
M001	1-FD-PSL-4908	E41-C002	Press. Switch Low	Reactor	54	No	No	123
M001	1-FD-TS-4909	E41-C002	Temp Switch	Reactor	54	No	No	123
M001	1-FD-PDSH-4910	E41-C002	Press Diff. Switch High	Reactor	54	No	No	123
M001	1-FD-LSHL-4912	E41-C002	Level Switch High Low	Reactor	54	No	No	123
M001	1-FD-PS-4913	E41-C002	Press. Switch	Reactor	54	No	No	123
M001	1-FD-SE-4919	E41-C002	Speed Elemt.	Reactor	54	No	No	123
P103A	1-FD-HV-4922		Contr. Valve	Reactor	54	No	No	139

HCGS PSAR
TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-57-1SYSTEM: CONTAINMENT ATMOS. CONTRL.
GS

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	EISS REF. NO.
				BLDG.	ELEV.			
J3590	1A-C-200		H2/O2 Analyzer PNL A	Reactor	162	No	Yes	30
J3590	1B-C-200		H2/O2 Analyzer PNL B	Reactor	162	No	Yes	30
J3590	1C-C-200		H2/O2 Analyzer Heat Trace PNL	Reactor	162	No	No	31
J3590	1D-C-200		H2/O2 Analyzer Heat Trace PNL	Reactor	162	No	No	31
J3590	1-GS-TE-0351		Temperature Element	Reactor	145	No	Yes	31
J3590	1-GS-TE-0352		Temperature Element	Reactor	145	No	Yes	31
J3590	1-GS-TE-0353		Temperature Element	Reactor	77	No	Yes	31
J3590	1-GS-TE-0354		Temperature Element	Reactor	162	No	Yes	31
J3590	1-GS-TE-0355		Temperature Element	Reactor	102	No	Yes	31
J3590	1-GS-TE-0356		Temperature Element	Reactor	162	No	Yes	31
J3590	1-GS-TE-0357		Temperature Element	Reactor	162	No	Yes	31
J3590	1-GS-TE-0358		Temperature Element	Reactor	162	No	Yes	31
J3590	1-GS-TE-0359		Temperature Element	Reactor	162	No	Yes	31
P3050	1-GS-SV-4950		Solenoid Valve	Reactor	145	No	No	145
P3050	1-GS-ZS-4950		Position Switch	Reactor	145	Yes	No	146
P303AQ	1-GS-HV-4951		Control Valve	Reactor	145	No	No	139
P303AQ	1-GS-ZS-4951		Limit Switch	Reactor	145	Yes	No	139
P3050	1-GS-SV-4952		Solenoid Valve	Reactor	145	No	No	145
P3050	1-GS-ZS-4952		Position Switch	Reactor	145	Yes	No	146
P303AQ	1-GS-HV-4955A		Control Valve	Reactor	132	No	No	139
P303AQ	1-GS-ZS-4955A		Limit Switch	Reactor	132	Yes	No	139
P303AQ	1-GS-HV-4955B		Control Valve	Reactor	162	No	No	139
P303AQ	1-GS-ZS-4955B		Limit Switch	Reactor	162	Yes	No	139
P3050	1-GS-SV-4956		Solenoid Valve	Reactor	102	No	No	145
P3050	1-GS-ZS-4956		Position Switch	Reactor	102	Yes	No	146
P3050	1-GS-SV-4958		Solenoid Valve	Reactor	77	No	No	145
P3050	1-GS-ZS-4958		Position Switch	Reactor	77	Yes	No	146
P303AQ	1-GS-HV-4959A		Control Valve	Reactor	77	No	No	139
J3590	1-GS-TE-0360		TEMPERATURE ELEMENT	REACTOR	162	No	YES	31

HCCS FSAR
TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-57-1SYSTEM: CONTAINMENT ATMOS. CONTRL.
GS

P.O.	ID NO. NOTE (1)	MPL NO.	COMPONENT	LOCATION		PAH EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	ESS REF. NO.
				BLDG.	ELEV.			
J1590	1-GS-SV-5086A1		Solenoid Valve	Reactor	162	No	No	30
J1590	1-GS-SV-5086A2		Solenoid Valve	Reactor	162	No	No	30
J1590	1-GS-SV-5086B1		Solenoid Valve	Reactor	162	No	No	30
J1590	1-GS-SV-5086B2		Solenoid Valve	Reactor	162	No	No	30
J1590	1-GS-HS-5087A		Hand Sw.	Reactor	162	No	No	30
J1590	1-GS-SV-5087A1		Solenoid Valve	Reactor	162	No	No	30
J1590	1-GS-SV-5087A2		Solenoid Valve	Reactor	162	No	No	30
J1590	1-GS-HS-5087B		Hand Sw.	Reactor	162	No	No	30
J1590	1-GS-SV-5087B1		Solenoid Valve	Reactor	162	No	No	30
J1590	1-GS-SV-5087B2		Solenoid Valve	Reactor	162	No	No	30
J1590	1-GS-TAH-5092A		Temp. Alarm High	Reactor	162	No	No	30
J1590	1-GS-TAL-5092A		Temp. Alarm Low	Reactor	162	No	No	30
J1590	1-GS-TAH-5092B		Temp. Alarm High	Reactor	162	No	No	30
J1590	1-GS-TAL-5092B		Temp. Alarm Low	Reactor	162	No	No	30
J1590	1-GS-TSH-5092A		Temp Sw. High	Reactor	162	No	No	30
J1590	1-GS-TSL-5092A		Temp. Sw. Low	Reactor	162	No	No	30
J1590	1-GS-TSH-5092B		Temp. Sw. High	Reactor	162	No	No	30
J1590	1-GS-TSL-5092B		Temp. Sw. Low	Reactor	162	No	No	30
J1590	1-GS-FAL-5094A		Flow Alarm High	Reactor	162	No	No	30
J1590	1-GS-FAL-5094B		Flow Alarm Low	Reactor	162	No	No	30
J1590	1-GS-PDS-5094A1		Press. Diff. Sw.	Reactor	162	No	No	30
J1590	1-GS-PDS-5094A2		Press. Diff. Sw.	Reactor	162	No	No	30
J1590	1-GS-PDS-5094B1		Press Diff. Sw.	Reactor	162	No	No	30
J1590	1-GS-PDS-5094B2		Press Diff. Sw.	Reactor	162	No	No	30
P101AQ	1-GS-HV-5741A		Control Valve	Reactor	132	No	No	139
P101AQ	1-GS-ZS-5741A		Limit Switch	Reactor	132	No	No	139
P101AQ	1-GS-HV-5741B		Control Valve	Reactor	132	No	No	139
P101AQ	1-GS-ZS-5741B		Limit Switch	Reactor	132	No	No	139
J1590	NO TAG NO.		Heat Tracing	Reactor	N/A	No	No	11

HCGS FSAR
 TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

 P&ID
 M-58-1

 SYSTEM: CONTAINMENT HYDROGEN RECOMBINATION SYSTEM
 GS

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	EISS REF. NO.
				BLDG.	ELEV.			
M047AQ	1A-S-205		Hydrogen Recombiner	Reactor	162	No	No	52,53,56 thru 64
M047AQ	1B-S-205		Hydrogen Recombiner	Reactor	162	No	No	52,53,56 thru 64
M047AQ	1A-C-215		Power Panel	Reactor	162	No	No	54
M047AQ	1A-E-215		Heater	Reactor	162	No	No	56
M047AQ	1A-V-215		Fan Motor	Reactor	162	No	No	52
M047AQ	1B-C-215		Power Panel	Reactor	162	No	No	54,55,64,65
M047AQ	1B-E-215		Heater	Reactor	162	No	No	56
M047AQ	1B-V-215		Fan Motor	Reactor	162	No	No	52
P1010	1-GS-HV-5050A		Contr. Valve	Reactor	145	No	Yes	127
P1010	1-GS-ZS-5050A		Limit Switch	Reactor	145	Yes	No	127
P1010	1-GS-HV-5050B		Contr. Valve	Reactor	102	No	Yes	127
P1010	1-GS-ZS-5050B		Limit Switch	Reactor	102	Yes	No	127
P1010	1-GS-HV-5052A		Contr. Valve	Reactor	145	No	Yes	127
P1010	1-GS-ZS-5052A		Limit Switch	Reactor	145	Yes	No	127
P1010	1-GS-HV-5052B		Contr. Valve	Reactor	102	No	Yes	127
P1010	1-GS-ZS-5052B		Limit Switch	Reactor	102	Yes	No	127
P1010	1-GS-HV-5053A		Contr. Valve	Reactor	77	No	Yes	127
P1010	1-GS-ZS-5053A		Limit Switch	Reactor	77	Yes	No	127
P1010	1-GS-HV-5053B		Contr. Valve	Reactor	77	No	Yes	127
P1010	1-GS-ZS-5053B		Limit Switch	Reactor	77	Yes	No	127
P1010	1-GS-HV-5054A		Contr. Valve	Reactor	77	No	Yes	127
P1010	1-GS-ZS-5054A		Limit Switch	Reactor	77	Yes	No	127
P1010	1-GS-HV-5054B		Contr. Valve	Reactor	77	No	Yes	127
P1010	1-GS-ZS-5054B		Limit Switch	Reactor	77	Yes	No	127
P103AQ	1-HC-HV-5055A		Contr. Valve	Reactor	54	No	No	139
P103AQ	1-HC-ZS-5055A		Limit Switch	Reactor	54	No	No	139
P103AQ	1-HC-HV-5055B		Contr. Valve	Reactor	77	No	No	139

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-72-1

SYSTEM: MAIN STEAM ISO VLV SEAL SYS
KP

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		P&I EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	EISS REF. NO.
				BLDG.	ELEV.			
P 10 1A0	1-KP-HV-5815A		Control Valve	Reactor	102	No	No	139
M001	1-AB-PT-5815A		Press Trans.	Reactor	102	Yes	No	155
P 10 1A0	1-KP-ZS-5815A		Limit Switch	Reactor	102	Yes	No	139
P 10 1A0	1-KP-HV-5815B		Control Valve	Reactor	102	No	No	139
M001	1-AB-PT-5815B		Press Trans.	Reactor	102	No	No	155
P 10 1A0	1-KP-ZS-5815B		Limit Switch	Reactor	102	No	No	139
P 10 1A0	1-KP-HV-5816A		Control Valve	Reactor	102	No	No	139
M001	1-AB-PT-5816A		Press Trans.	Reactor	102	Yes	No	155
P 10 1A0	1-KP-ZS-5816A		Limit Switch	Reactor	102	Yes	No	139
P 10 1A0	1-KP-HV-5816B		Control Valve	Reactor	102	No	No	139
M001	1-AB-PT-5816B		Press Trans.	Reactor	102	No	No	155
P 10 1A0	1-KP-ZS-5816B		Limit Switch	Reactor	102	No	No	139
P 10 1A0	1-KP-HV-5817A		Control Valve	Reactor	102	No	No	139
M001	1-AB-PT-5817A		Press Trans.	Reactor	102	Yes	No	155
P 10 1A0	1-KP-ZS-5817A		Limit Switch	Reactor	102	Yes	No	139
P 10 1A0	1-KP-HV-5817B		Control Valve	Reactor	102	No	No	139

HCGS FSAR
TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-83-1

SYSTEM: REACTOR BLDG. SUPPLY CONTROL DIAG.
GR

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	EESS REF. NO.
				BLDG.	ELEV.			
M7860	IAC-043		Heater Control Panel	Reactor	132	No	No	100, 163
M7860	IBC-043		Heater Control Panel	Reactor	178	No	No	100, 163
M7860	ICC-043		Heater Control Panel	Reactor	132	No	No	100, 163
M7860	IDC-043		Heater Control Panel	Reactor	162	No	No	100, 163
M7860	IAC-044		Heater Control Panel	Reactor	162	No	No	100, 163
M7860	IBC-044		Heater Control Panel	Reactor	178	No	No	100, 163
M7110	1A-VH-208		Unit Cooler	Reactor	54	No	No	81
M7110	1B-VH-208		Unit Cooler	Reactor	54	No	No	81
M7110	1A-VH-209		Unit Cooler	Reactor	54	No	No	81
M7110	1B-VH-209		Unit Cooler	Reactor	54	No	No	81
M7110	1A-VH-210		Unit Cooler	Reactor	54	No	No	81
M7110	1B-VH-210		Unit Cooler	Reactor	54	No	No	81
M7110	1C-VH-210		Unit Cooler	Reactor	54	No	No	81
M7110	1D-VH-210		Unit Cooler	Reactor	54	No	No	81
M7110	1E-VH-210		Unit Cooler	Reactor	77	No	No	81
M7110	1F-VH-210		Unit Cooler	Reactor	77	No	No	81
M7110	1G-VH-210		Unit Cooler	Reactor	54	No	No	81
M7110	1H-VH-210		Unit Cooler	Reactor	54	No	No	81
M7110	1A-VH-211		Unit Cooler	Reactor	54	No	No	81
M7110	1B-VH-211		Unit Cooler	Reactor	54	No	No	81
M7110	1C-VH-211		Unit Cooler	Reactor	54	No	No	81
M7110	1D-VH-211		Unit Cooler	Reactor	54	No	No	81
M7110	1E-VH-211		Unit Cooler	Reactor	54	No	No	81
M7110	1F-VH-211		Unit Cooler	Reactor	54	No	No	81
M7110	1G-VH-211		Unit Cooler	Reactor	54	No	No	81
M7110	1H-VH-211		Unit Cooler	Reactor	54	No	No	81
M7110	1A-V-213		Fan & E-H Actuator	Reactor	132	No	No	81
M7110	1B-V-213		Fan & E-H Actuator	Reactor	178	No	No	81

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
M-83-1

SYSTEM: REACTOR BLDG. SUPPLY CONTROL DIAG.
GR

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION PLAN EQUIP. NOTE (2)	FESS REF. NO.
				BLDG.	ELEV.			
M7130	1C-V-213		Fan & E-H Actuator	Reactor	132	No	No	83
M7130	1D-V-213		Fan & E-H Actuator	Reactor	162	No	No	83
M7130	1E-V-213		Fan & E-H Actuator	Reactor	162	No	No	83
M7130	1F-V-213		Fan & E-H Actuator	Reactor	178	No	No	83
M7110	1A-VH-214		Unit Cooler	Reactor	102	No	No	81
M7110	1B-VH-214		Unit Cooler	Reactor	102	No	No	81
M7110	1C-VH-214		Unit Cooler	Reactor	102	No	No	81
M7110	1D-VH-214		Unit Cooler	Reactor	102	No	No	81
M780AQ	1A-C-281		Unit Cooler Ctrl Pnl.	Reactor	102	No	No	91,97,98,666
M780AQ	1B-C-281		Unit Cooler Ctrl Pnl.	Reactor	102	No	No	91,97,98,666
M780AQ	1C-C-281		Unit Cooler Ctrl Pnl.	Reactor	102	No	No	91,97,98,666
M780AQ	1D-C-281		Unit Cooler Ctrl Pnl.	Reactor	77	No	No	91,97,98,666

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TABLE 3.11-5

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EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

SYSTEM: REAC BLDG EXH CONTRL /
GU

P&ID
M-84-1

P.O.	ID NO. NOTE (5)	MPL NO.	COMPONENT	LOCATION		PAM EQUIP. NOTE (1)	TMI ACTION		EISS REF. NO.
				BLDG.	ELEV.		PLAN EQUIP. NOTE (2)		
M786Q	IAC-045		Heater Control Panel	Reactor	145	No	No	100, 163	
M786Q	IBC-045		Heater Control Panel	Reactor	145	No	No	100, 163	
M711Q	1A-V-206		Fan & EH Actuator	Reactor	145	No	No	83	
M711Q	1B-V-206		Fan & EH Actuator	Reactor	145	No	No	83	
M728Q	1-GU-SV-9414A1		Solenoid Valve	Reactor	178	No	No	88	
M728Q	1-GU-SV-9414A2		Solenoid Valve	Reactor	178	No	No	88	
M728Q	1-GU-ZS-9414A		Position Switch	Reactor	178	Yes	No	89	
M728Q	1-GU-SV-9414B1		Solenoid Valve	Reactor	178	No	No	88	
M728Q	1-GU-SV-9414B2		Solenoid Valve	Reactor	178	No	No	88	
M728Q	1-GU-ZS-9414B		Position Switch	Reactor	178	Yes	No	89	
J301Q	1-GU-FT-9425A		Flow Trans.	Reactor	145	No	No	29	
M780AQ	1-GU-PDT-9425A		Press. Diff. Trans.	Reactor	145	No	No	99	
M780AQ	1-GU-TE-9425A		Temp. Elemt.	Reactor	145	No	No	93	
M717Q	1-GU-HD-9425A1		Hand Damper Actuator	Reactor	162	No	No	87	
M717Q	1-GU-HD-9425A2		Hand Damper Actuator	Reactor	145	No	No	87	
M717Q	1-GU-FD-9425A3		Flow Damper Actuator	Reactor	145	No	No	87	
M717Q	1-GU-FD-9425A5		Flow Damper Actuator	Reactor	145	No	No	87	
J301Q	1-GU-FT-9425B		Flow Trans.	Reactor	145	No	No	29	
M780AQ	1-GU-PDT-9425B		Press. Diff. Trans.	Reactor	145	No	No	99	
M780AQ	1-GU-TE-9425B		Temp. Elemt.	Reactor	145	No	No	93	
M717Q	1-GU-HD-9425B1		Hand Damper Actuator	Reactor	162	No	No	87	
M717Q	1-GU-HD-9425B2		Hand Damper Actuator	Reactor	145	No	No	87	
M717Q	1-GU-FD-9425B3		Flow Damper Actuator	Reactor	145	No	No	87	
M717Q	1-GU-FD-9425B5		Flow Damper Actuator	Reactor	145	No	No	87	
M711Q	1-GU-FD-9426A		Flow Damper Actuator	Reactor	145	No	No	84	
M780AQ	1-GU-FT-9426A		Flow Trans.	Reactor	145	No	No	99	
M780AQ	1-GU-FSL-9426A1		Flow Sw. Low	Reactor	145	No	No	92	
M780AQ	1-GU-PDT-9426A1		Press. Diff. Trans.	Reactor	201	No	No	99	
M780AQ	1-GU-FSL-9426A2		Flow Sw. Low	Reactor	145	No	No	92	
M780AQ	1-GU-PDT-9426A2		Press. Diff. Trans.	Reactor	102	No	No	99	

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TABLE 3.11-5

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
N/A

SYSTEM: GENERIC

P.O.	ID NO. NOTE (5)	VENDOR	COMPONENT	LOCATION		P&I EQUIP. NOTE (1)	T&I ACTION PLAN EQUIP. NOTE (2)	E&S REF. NO.
				BLDG.	ELEV.			
E1700	R14	Brand-Rex	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	9
E1700	R16	Brand-Rex	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	9
E1700	R58	Brand-Rex	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	9
E1700	R59	Brand-Rex	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	9
E1700	R62	Brand-Rex	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	9
E170AQ	RM1	Rockbestos	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	12, 12B
E170AQ	RM2	Rockbestos	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	12
E170AQ	RM7	Rockbestos	300V Shielded Inst. Cable	Note (3)		N/A	N/A	10
E170AQ	RO6	Rockbestos	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	10
E170AQ	RF9	Rockbestos	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	10
E170AQ	RG11	Rockbestos	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	10
E170AQ	R22	Rockbestos	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	10
E170AQ	S13	Rockbestos	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	11, 12B
E170AQ	S110	Rockbestos	600V Control/Power Cable	Note (3)		N/A	N/A	11
E170AQ	S14	Rockbestos	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	11, 12B
E170AQ	S16	Rockbestos	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	11, 12B
E170AQ	S18	Rockbestos	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	11, 12B
E170AQ	VK4	Rockbestos	Co, Tri, and Twinaxial Cable	Note (3)		N/A	N/A	12A, 12B
E171Q	104	Eaton	600V Shielded Instru Cable	Note (3)		N/A	N/A	13
E171Q	DO1	Eaton	600V Shielded Instru Cable	Note (3)		N/A	N/A	13
E171Q	DO2	Eaton	600V Shielded Instru Cable	Note (3)		N/A	N/A	13
E171Q	DO3	Eaton	600V Shielded Instru Cable	Note (3)		N/A	N/A	13
E171Q	DO4	Eaton	600V Shielded Instru Cable	Note (3)		N/A	N/A	13
E171Q	DO5	Eaton	600V Shielded Instru Cable	Note (3)		N/A	N/A	13
E171Q	DO6	Eaton	600V Shielded Instru Cable	Note (3)		N/A	N/A	13

EQUIPMENT SELECTED FOR HARSH ENVIRONMENT QUALIFICATION

P&ID
N/A

SYSTEM: GENERIC

P.O.	ID NO.	VENDOR	COMPONENT	LOCATION		P&M	T&I ACTION	E&SS REF. NO.
	NOTE (5)			BLDG.	ELEV.	EQUIP.	PLAN EQUIP.	
						NOTE (1)	NOTE (2)	
F436200-1-1F		Raychem Corp.	High Voltage Terminations	Note (3)		N/A	N/A	19
F478880-1-1F	N-MCK	Raychem Corp.	Motor Connection Kit	Note (3)		N/A	N/A	20
F484790-1-1F	WCSP-N	Raychem Corp.	Cable Breakout Kit	Note (3)		N/A	N/A	21
F484790-1-1F		Raychem Corp.	Cable End Sealing Kit	Note (3)		N/A	N/A	21
F485610-1-1F	NPXS,NPKX,NPKC	Raychem Corp.	Cable Splice Assy	Note (3)		N/A	N/A	23
F485610-1-1F		Raychem Corp.	ThermoFit Insulation Sys	Note (3)		N/A	N/A	22
F512160-1-1F	N-21009-01	Conax Corp.	Electric Conductor Seal Assy	Note (3)		N/A	N/A	24
F61553	N-11176 N-11177	CONAX CORP.	MODIFIED PG GLAND ASSY	NOTE(3)		N/A	N/A	33

SAFETY-RELATED EQUIPMENT LOCATED IN A HARSH ENVIRONMENT EXEMPTED
FROM ENVIRONMENTAL QUALIFICATION REQUIREMENTS

EQUIPMENT TAG NO.	MPL NO.	DESCRIPTION	REASON
1-AE-HV-F039		Motor Operated Valves	<p>These motor operated valves are not qualified for submergence caused by a feedwater line break in the steam tunnel. They have been provided with primary and backup IE bus protective devices located in the hazard free area.</p> <p style="text-align: center;">AND SOLENOID VALVES</p>
1-AB-HV-F071		Motor Operated Valves	
1-KP-HV-5829A,B		Motor Operated Valves	
1-KP-HV-4834A,B		Motor Operated Valves	
1-KP-HV-5835A,B		Motor Operated Valves	
1-KP-HV-5836A,B		Motor Operated Valves	
1-KP-HV-5837A,B		Motor Operated Valves	
1-AB-HV-F067A		Motor Operated Valves	
1-AB-HV-F067B		Motor Operated Valves	
1-AB-HV-F067C		Motor Operated Valves	
1-AB-HV-F067D		Motor Operated Valves	
No Tag No.	C11-F010	Position Switch	<p>These NAMCO limit switches perform no safety functions. Failure modes and effect analysis has shown that there are no possible failure modes which can adversely effect the IE power supply.</p>
No Tag No.	C11-F011	Position Switch	
No Tag No.	C11-F180	Position Switch	
No Tag No.	C11-F181	Position Switch	
1-BE-SV-F006A	E21	Solenoid Valve	<p>These solenoid valves and position switches perform no safety functions. However, because of their association with a IE power supply, they have been provided with primary and backup protective devices.</p>
No Tag No.	E21-F006A	Position Switch	
1-BE-SV-F006B	E21	Solenoid Valve	
No Tag No.	E21-F006B	Position Switch	
1-BC-SV-F041A	E11	Solenoid Valve	
No Tag No.	E11-F041A	Position Switch	
1-BC-SV-F041B	E11	Solenoid Valve	
No Tag No.	E11-F041B	Position switch	
1-BC-SV-F041C	E11	Solenoid Valve	
No Tag No.	E11-F041C	Position Switch	

1-AE-HV-4144	MOTOR OPERATED VALVES
1-KL-PDV-5825A	SOLENOID VALVE
1-KL-PDV-5825B	SOLENOID VALVE

from its drive without disturbing the remainder of the control system. The bottom-mounted drives permit the entire control system to be left intact and operable for tests with the reactor vessel open.

4.1.3.2 Description of Control Rods

REFERENCED A description of the control rods is ~~given~~ in Section 4.2.2.1.

4.1.3.3 Supplementary Reactivity Control

The initial- and reload-core control requirements are met by use of the combined effects of the movable control rods, supplementary burnable poison, and variation of reactor coolant flow. A description of the supplementary burnable poison is ~~given~~ in Section 4.2.

4.1.4 ANALYSIS TECHNIQUES

4.1.4.1 Reactor Internal Components

Computer codes used for the analysis of the internal components are as follows:

- a. MASS
- b. SNAP (MULTISHELL)
- c. GASP
- d. NOHEAT
- e. FINITE
- f. DYSEA
- g. SHELL 5

- e. Mechanical reactivity control permits criticality checks during refueling and provides maximum plant safety. At any time in its operating history, the core is designed to be subcritical with any one control rod fully withdrawn.
- f. The selected control rod pitch represents a practical value of individual control rod reactivity worth, and allows adequate clearance between control rod drive (CRD) mechanisms below the pressure vessel for ease of maintenance and removal.

4.1.2.1.2 Core Configuration

The reactor core is arranged as an upright circular cylinder containing a large number of fuel cells and is located within the reactor vessel. The coolant flows upward through the core. The core arrangement (plan view) and the lattice configuration are given in Section 4.3.

4.1.2.1.3 Fuel Assembly Description

REFERENCED Descriptions of the fuel assembly and the fuel rods are ~~given~~ in Section 4.2.

4.1.2.1.4 Fuel Assembly Support and Control Rod Location

A few peripheral fuel assemblies are supported by the core plate. Otherwise, individual fuel assemblies in the core rest on fuel support pieces mounted on top of the control rod guide tubes. Each guide tube, with its fuel support piece, bears the weight of four fuel assemblies and is supported by a CRD penetration nozzle in the bottom head of the reactor vessel. The core plate provides lateral support and guidance at the top of each control rod guide tube.

The top guide, mounted inside the shroud, provides lateral support and guidance for each fuel assembly. The reactivity of the core is controlled by cruciform control rods and their associated mechanical-hydraulic drive system. The control rods occupy alternate spaces between fuel assemblies. Each independent CRD enters the core from the bottom, accurately positions its associated control rod during normal operation, and

4.1.4.3 Reactor Systems Dynamics

RESULTS OF THE The analysis techniques and computer codes used in reactor systems dynamics are described in Section 4 of References 4.1-10, 4.1-10A, and 4.1-10B. Section 4.4.4 ~~also~~ provides a complete stability analysis for the reactor coolant system (RCS).

4.1.4.4 Nuclear Engineering Analysis

The analysis techniques are described and referenced in Section 3 of Reference 4.1-1.

4.1.4.5 Neutron Fluence Calculations

Vessel neutron-fluence calculations are carried out using a one-dimensional, discrete-ordinates, Sn transport code with general anisotropic scattering.

This code is a modification of a widely used discrete-ordinates code that solves a wide variety of radiation transport problems. The program solves both fixed source and multiplication problems. Slab, cylinder, and spherical geometries are allowed with various boundary conditions. The fluence calculations incorporate, as an initial starting point, neutron fission distributions prepared from core physics data as a distributed source. Anisotropic scattering is considered for all regions. The cross sections are prepared with 1/E-flux-weighted, P matrices for anisotropic scattering, but do not include resonance self-shielding factors. Fast neutron fluxes at locations other than the core midplane are calculated using a two-dimensional, discrete-ordinates code. The two-dimensional code is an extension of the one-dimensional code.

4.1.4.6 Thermal-Hydraulic Calculations

A description of the thermal-hydraulic models is given in Section 4 of Reference 4.1-1.

4.1.5 REFERENCES

- 4.1-1 "General Electric Standard Application for Reactor Fuel," including the "United States Supplement,"

-7-

-7-

NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision)

- 4.1-2 L. Beitch, "Shell Structures Solved Numerically by Using a Network of Partial Panels," AIAA Journal, Volume 5, No. 3, March 1967.
- 4.1-3 E. L. Wilson, "A Digital Computer Program For the Finite Element Analysis of Solids With Non-Linear Material Properties," Aerojet General Technical Memo No. 23, Aerojet General, July 1965.
- 4.1-4 I. Farhoomand and E. L. Wilson, "Non-Linear Heat Transfer Analysis of Axisymmetric Solids," SESM Report SESM71-6, University of California at Berkeley, Berkeley, California, 1971.
- 4.1-5 J. E. McConnelee, "Finite-Users Manual," General Electric TIS Report DF 69SL206, March 1969.
- 4.1-6 R. W. Clough and C. P. Johnson, "A Finite Element Approximation For the Analysis of Thin Shells," International Journal of Solid Structures, Vol. 4, 1968.
- 4.1-7 "A Computer Program For the Structural Analysis of Arbitrary Three-Dimensional Thin Shells," Report No. GA-9952, Gulf General Atomic, 1969.
- 4.1-8 A. B. Burgess, "User Guide and Engineering Description of HEATER Computer Program," General Electric, NEDE-20731-02 March 1974.
- 4.1-9 L. J. Young, "FAP-71 (Fatigue Analysis Program) Computer Code," GE/NED Design Analysis Unit R. A. Report No. 49, January 1972.
- 4.1-10 L. A. Carmichael and G. J. Scatena, Stability and Dynamic Performance of the General Electric Boiling Water Reactor, NEDO-21506, January 1977.

4.2 FUEL SYSTEM DESIGN

INSERT →

The fuel system design for the HCGS is identical to that which the NRC reviewed and approved for GESSAR II (Reference 4.2-1). Methods and criteria used to evaluate fuel system performance are also identical to those used for GESSAR II, except for the evaluation of combined fuel-lift loadings from a safe shutdown earthquake and a loss-of-coolant accident. See Section 3.9.1.4.10 and Table 3.9-5 for the results of the fuel-lift evaluation. The results of the NRC review of Section 4.2 of GESSAR II documented in References 4.2-2 and 4.2-3 are therefore applicable to the HCGS.

4.2.1 REFERENCES

- 4.2-1 General Electric Standard Safety Analysis Report, Docket No. 50-447
- 4.2-2 NUREG-0979, "Safety Evaluation Report Related to the Final Design Approval of the GESSAR II BWR/6 Nuclear Island Design", April, 1983
- 4.2-3 NUREG-0979 (Supplement No. 1), "Safety Evaluation Report Related to the Final Design Approval of the GESSAR II BWR/6 Nuclear Island Design", July, 1983

INSERT FOR PAGE 4.2-1

The format of this section corresponds to Standard Review Plan 4.2 in NUREG-0800. Most of the information is presented by reference to GESTAR II (Ref. 4.2-1).

4.2.1 DESIGN BASES

Reference to design bases are given in Subsection A.4.2.1 of GESTAR II (Ref. 4.2-1).

4.2.2 DESCRIPTION AND DESIGN DRAWINGS

Reference to the fuel system description and design drawings are given in Subsection A.4.2.2 of GESTAR II (Ref. 4.2-1).

4.2.2.1 Reactivity Control Assembly (Control Rods)

The control rod description is given in Subsection 2.2.4 and is shown in Figures 2.6a, 2.6b, and 2.7 of NEDE 20944-P-1 (Ref. 4.2-2).

4.2.2.2 Reactivity Control Assembly Evaluation

The control rod evaluation is given in Subsection 2.3.3 of NEDE 20944-P-1 (Ref. 4.2-2).

4.2.3 DESIGN EVALUATION

Compliance with the design bases is discussed in Subsection A.4.2.3 of GESTAR II (Ref. 4.2-1), with the exception that Paragraphs 4.2.3.2.9 and 4.2.3.3.5 appear as below.

4.2.3.2.9 Mechanical Fracturing Evaluation

All mechanical breaking under normal operation and abnormal operational transients is bounded by the analysis for LOCA plus SSE given in Section 3.9.1.4.10.

4.2.3.3.5 Structural Deformation Evaluation

Results of the Hope Creek specific SSE plus LOCA analysis are documented in Section 3.9.1.4.10.

4.2.4 TESTING, INSPECTION AND SURVEILLANCE PLANS

Descriptions of fuel assembly testing, inspection, and surveillance are referenced in Subsection A.4.2.4 of GESTAR II (Ref. 4.2-1).

INSERT FOR PAGE 4.2.1 (Continued)

4.2.5 REFERENCES

- 4.2-1 "General Electric Standard Application for Reactor Fuel", including the "United States Supplement", NEDE-24011-P-A-7, and NEDE-24011-P-A-7-US.
- 4.2-2 "BWR/4 and BWR/5 Fuel Design", NEDE-20944-P-1 (Proprietary) and NEDO-20944-1, October 1976, and "Amendment 1", January 1977.

4.3.2.4.2 Reactivity Variations

Information on reactivity variations is referenced in Subsection A.4.3.2.4.2 of Reference 4.3-1. The combined effects of the individual constituents of reactivity are accounted for in each K_{eff} in Table ~~4.3-1~~.

4.3-7

4.3.2.5 Control Rod Patterns and Reactivity Worths

Control rod patterns and reactivity worths are discussed in Section 3.2.5 of NEDE-20944-P-1 (Reference 4.3-2). Typical control rod patterns and the associated power distributions are presented in Appendix A of Reference 4.3-2. These control rod patterns are calculated with the BWR Core Simulator. Qualification for this model is discussed and referenced in Section 3.1 of Reference 4.3-1.

~~4.3.2.5.1 Seram Reactivity~~

~~Seram reactivity is calculated as described in Section 5.2 of Reference 4.3-1 and is discussed in Section 3.2.5.3 of Reference 4.3-2.~~

4.3.2.6 Criticality of Reactor During Refueling

4.3.2.7 Stability

4.3.2.7.1 Xenon Transients

4.3.2.7.2 Thermal Hydraulic Stability

4.3.2.8 Vessel Irradiations

The neutron fluxes at the vessel have been calculated using the one-dimensional, discrete-ordinates, transport code described in Section 4.1.4.5. The discrete-ordinates code is used in a distributed source mode with cylindrical geometry. The geometry describes six regions from the center of the core to a point beyond the vessel. The core region is modeled as a single, homogenized cylindrical region. The coolant water region between

4.3.5 REFERENCES

- 4.3-1 "General Electric Standard Application for Reactor Fuel," including the "United States Supplement," NEDE-24011-P-A and NEDE-24011-P-A-US. ~~(latest approved revision)~~
- 4.3-2 "BWR/4 and BWR/5 Fuel Design," NEDE-20944-1 (Proprietary) and NEDO-20944-1, October 1976, and "Amendment 1," January 1977.

4.4 THERMAL AND HYDRAULIC DESIGN

Most of the information in Section 4.4 is provided in the licensing topical report GESTAR II (Reference 4.4-1). The section numbers in Section 4.4 directly correspond to subsection numbers of Appendix A of GESTAR II. The differences are discussed below.

4.4.1 DESIGN BASES

The thermal and hydraulic design bases are referenced in Section A.4.4.1 of Reference 4.4-1. The design steady-state operating limit for the minimum critical power ratio (MCPR) and the linear heat generation rate (LHGR) are given in Table 4.4-1.

4.4.2 DESCRIPTION OF THERMAL AND HYDRAULIC DESIGN OF THE REACTOR CORE

A description of the thermal and hydraulic design of the reactor core is referenced in Section A.4.4.2 of Reference 4.4-1. Any additions or differences are given in the appropriate section below.

An evaluation of plant performance from a thermal and hydraulic standpoint is provided in Section 4.4.3.

4.4.2.1 Summary Comparison


Summary A comparison of the thermal and hydraulic design parameters of this reactor with those of reactors of similar design is given in Table 4.4-1.


4.4.2.2 Linear Heat Generation Rate


4.4.2.3 Void Fraction Distribution

The core average and maximum exit void fractions in the core at rated condition are given in Table 4.4-1. The axial distribution of core void fractions for the average radial channel and the maximum radial channel (end-of-node value) for the core are given


in Table 4.4-3. The core average and maximum exit value are also provided. Similar distributions for steam quality are provided in Table 4.4-5. The core average axial power distribution used to produce these tables is given in Table 4.4-4.

4.4.2.  Core Coolant Flow Distribution and Orificing Pattern


4.4.2.  Core Pressure Drop and Hydraulic Loads

4.4.2.  Correlation and Physical Data

GE has obtained substantial amounts of physical data in support of the pressure drop and thermal-hydraulic loads. This information is given in Appendix B of Reference 4.4-1 where responses are provided to NRC questions on Section 4 of GESTAR II.

4.4.2.  Thermal Effects of Operational Transients

4.4.2.  Uncertainties in Estimates

4.4.2.  Flux Tilt Considerations

The inherent design characteristics of the BWR are particularly well suited to handle perturbations due to flux tilt. The stabilizing nature of the moderator void coefficient effectively damps oscillations in the power distribution. In addition to this damping, the in-core instrumentation system and the associated on-line computer provide the operator with prompt and reliable power distribution information. Thus, the operator can readily use control rods or other means to limit effectively the undesirable effects of flux tilting. Because of these features and capabilities, it is not necessary to allocate a specific peaking factor margin to account for flux tilt. If, for some reason, the power distribution could not be maintained within normal limits using control rods, then the operating power limits would have to be reduced as prescribed in Chapter 16.

4.4.3 DESCRIPTION OF THE THERMAL AND HYDRAULIC DESIGN OF THE REACTOR COOLANT SYSTEM

4.4.7 SRP RULE REVIEW

Acceptance criterion II.8 of SRP Section 4.4 specifies, in part, that the effects of crud should be accounted for in the thermal-hydraulic design, and also that process monitoring provisions be capable of detecting a three percent pressure drop in the reactor coolant flow.

In general, the critical power ratio (CPR) is not affected as crud accumulates on fuel rods (References 4.4-2 and 4.4-3). Therefore, no modifications to GEXL are made to account for crud deposition. For pressure drop considerations, the amount of crud assumed to be deposited on the fuel rods and fuel rod spacers is greater than is actually expected at any point in the fuel lifetime. This crud deposition is reflected in a decreased flow area, increased friction factors, and increased spacer loss coefficients, the effect of which is to increase the core pressure drop by approximately 1.7 psi, an amount which is large enough to be detected in monitoring of core pressure drop. It should be noted that assumptions made with respect to crud deposition in core thermal-hydraulic analyses are consistent with established water chemistry requirements. More detailed discussion of crud (service-induced variations) and its uncertainty is found in Section III of Reference 4.4-4.

4.4.8 REFERENCES

- 4.4-1 General Electric Standard Application for Reactor Fuel," including the "United States Supplement," NEDE-24011-P-A and NEDE-24011-P-A-US. (latest approved revision)
- 4.4-2 R.V. McBeth, R. Trenberth, and R. W. Wood, "An Investigation Into the Effects of Crud Deposits on Surface Temperature, Dry-Out, and Pressure Drop, with Forced Convection Boiling of Water at 69 Bar in an Annular Test Section," AEEW-R-705, 1971.
- 4.4-3 S.J. Green, B.W. LeTourneau, and A.C. Peterson, "Thermal and Hydraulic Effects of Crud Deposited on Electrically Heated Rod Bundles," WAPD-TM-918 September 1970.
- 4.4-4 General Electric, "General Electric Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application," NEDO-10958A, January 1977.

THERMAL AND HYDRAULIC DESIGN CHARACTERISTICS
OF THE REACTOR CORE

<u>General Operating Conditions</u>	HCGS (251-764)	WNP-2 (251-764)	NMP-2 (251-764)
Reference design thermal output, MWT	3293	3323	3323
Power level for engineered safety features, MWT	3436	3489	3461
Steam flow rate, at 419.9°F final feedwater temperature (FFWT), millions lb/h	14.159	14.30	14.30
Core coolant flow rate, millions lb/h	100.0	100.5	100.5
Feedwater flow rate, millions lb/h	14.127	14.26	14.27
System pressure, nominal in steam dome, psia	1020	1020	1020
System pressure, nominal core design, psia	1035	1035	1035
Coolant saturation temperature at core design pressure, °F	549	549	549
Average power density, kW/liter	48.7	49.15	49.15
Maximum linear heat generation rate, kW/ft	13.4	13.4	13.4
Average linear heat generation rate, kW/ft	5.34	5.40	5.39
Core total heat transfer area, ft ²	74,841	74,781	74,781
Maximum heat flux, Btu/h-ft ²	361,600	361,500	361,600
Average heat flux, Btu/h-ft ²	144,100	145,100	145,050
Design operating minimum critical power ratio (MCPR)	(see Chapter 15)		
Core inlet enthalpy, at 419.9°F FFWT, Btu/lb	526.1	527.6	527.6

HCGS FSAR

TABLE 4.4-1 (cont)

Core inlet temperature, at 419.9°F FFWT, °F	531.6	533.0 (1) 532.0 (1)
Core maximum exit voids within assemblies, %	77.1	76.0 — 76.2
Core average void fraction, active coolant	0.419	0.418 — 0.408
Maximum fuel temperature, °F	3435	3435 — 3435
Active coolant flow area per assembly, in. ²	15.824	15.824 — 15.824
Core average inlet velocity, ft/s	6.41	6.88 — 6.91
Maximum inlet velocity, ft/s	6.803	7.28 — 8.00
Total core pressure drop, psi	21.25	24.74 — 23.7
Core support plate pressure drop, psi	16.82	20.32 — 19.28
Average orifice pressure drop Central region, psi	7.16	6.03 — 6.05
Peripheral region, psi	14.53	16.54 — 16.0
Maximum channel pressure loading, psi	10.88	13.28 — 12.16
Average-power assembly channel pressure loading (bottom), psi	9.61	— 10.83
Shroud support ring and lower shroud pressure loading, psi	22.87	— 27.52
Upper shroud pressure loading, psi	6.05	— 7.2

(1) At 420°F FFWT

TABLE 4.4-10

BYPASS FLOW PATHS

Flow Path Description⁽¹⁾	Driving Pressures	Number of Paths
1a. Between fuel support and the control rod guide tube (upper path)	Core plate differential	One/control rod
1b. Between fuel support and the control rod guide tube (lower path)	Core plate differential	One/control rod
2. Between core plate and control rod guide tube	Core plate differential	One/control rod
3. Between core support and the in-core instrument guide tube	Core plate differential	One/instrument
4. Between core plate and shroud	Core plate differential	One
5. Between control rod guide tube and control rod drive housing	Core plate differential	One/control rod
6. Between fuel support and lower tie plate	Channel wall differential plus lower tie plate differential	One/channel
7. Control rod drive coolant	Independent of core	One/control rod
8. Between fuel channel and lower tie plate	Channel wall differential	One/channel
9. Holes in lower tie plate	Lower tie plate/ bypass region differential	Two/assembly

~~(1) See Figure 4.4-2.~~

HCGS FSAR

rod indicates indirectly that the rod and drive are coupled. The over-travel position feature provides a positive check on the coupling integrity, for only an uncoupled drive can reach the over-travel position.

- d. During operation, accumulator pressure and level at the normal operating value is verified.

Experience with CRD systems of the same type indicates that weekly verification of accumulator pressure and level is sufficient to ensure operability of the accumulator portion of the CRD system.

- e. At each refueling outage, each operable control rod is subjected to scram time tests from the fully withdrawn position.

Experience indicates that the scram times of the control rods do not significantly change over the time interval between refueling outages. A test of the scram times at each refueling outage is sufficient to identify any significant lengthening of the scram times.

INSERT →

4.6.3.1.6 Functional Tests

The functional testing program of the CRDs consists of the 5-year maintenance life and the 1.5X design life test programs, as described in Section 3.9.4.

There are a number of failures that can be postulated on the CRD, but it would be very difficult to test all possible failures. A partial test program with postulated accident conditions and imposed single failures is available.

The following tests with imposed single failures have been performed to evaluate the performance of the CRDs under these conditions:

- a. Simulated ruptured scram line test

INSERT FOR PAGE 4.6-43

- f. Prior to startup after the first refueling outage, PSE&G shall:
1. Confirm that the leak rates, loading conditions and material properties for the scram discharge volume piping system are bounded by the limiting values for those parameters identified in the May 10, 1984 BWR Owners Group letter,
 2. Comply with BWR Owners Group recommendations for leak detection capability,
 3. Comply with the applicable generic secondary containment Emergency Procedure Guidelines, and
 4. Provide assurance that the expected radiation fields and contact exposure levels at the scram discharge volume piping systems in the facility will not impair the performance of routine tests, inspections, and post-scram reset walkdowns.

automatically realign from system flow test modes to the emergency core cooling mode of operation following receipt of an automatic initiation signal. The core spray and LPCI systems begin injection into the reactor pressure vessel (RPV) when reactor vessel pressure decreases to system discharge shutoff pressure. HPCI injection begins as soon as the HPCI turbine-pump is up to speed. The injection valve is open, since the HPCI system is capable of injecting water at full flow into the RPV over a pressure range from 200 psig to reactor pressure specified in mode A of Figure 6.3-3.

6.3.6 REFERENCES

- 6.3-1 General Electric, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566-P, November 1975.
- 6.3-2 H. M. Hirsch, Methods for Calculating Safe Test Intervals and Allowable Repair Times for Engineered Safeguard Systems, NEDO-10739, General Electric, January 1973.
- 6.3-3 General Electric, "General Electric Standard Application for Reactor Fuel," including the "United States Supplement," NEDE-24011-P-A and NEDE-24011-P-A-US. (latest approved revision)

-7

RHR pump A or C or core spray pump A, and RHR pump A or C or core spray pump C.

After receipt of the initiation signals and after a delay provided by time delay relays, each of the two solenoid pilot air valves for all ADS valves are energized. This allows pneumatic pressure from each ADS valve accumulator to act on the air cylinder operator of its respective ADS valve. Each ADS trip system timer can be reset manually to delay system initiation. If reactor vessel water level is restored by the HPCI system prior to the end of the time delay, ADS initiation will be prevented.

A manual inhibit switch is provided in each division of the ADS initiation logic. By placing this switch in the inhibit position, the operator will inhibit automatic depressurization. This will be indicated by a white status light and an annunciator window in the main control room. If the ADS has already begun and the initiation signal is sealed in, the inhibit switch will not break the seal-in, and the operation of the ADS will not be terminated.

The ADS trip system B actuates the A solenoid pilot valve on each ADS valve. Similarly, the ADS trip system D actuates the B solenoid pilot valve on each ADS valve. Actuation of either solenoid pilot valve causes the ADS valve to open to provide depressurization.

Manual initiation of the ADS trip systems or individual ADS valves is possible from the main control room. To manually initiate an ADS trip system, the control room operator must actuate two armed pushbutton switches, one for each of the logic channels associated with that trip system. Manual initiation bypasses the ADS trip system time delay and all the trip logic. The control room operator can manually open an individual ADS valve by depressing one of the two pushbutton switches (one for each pilot solenoid) that will bypass the trip logic and energize the associated pilot solenoid allowing air to open the valve.

- INSERT
- c. ADS testability - The ADS has two complete trip systems, one in trip system B and one in trip system D.

INSERT FOR PAGE 7.3-9

In addition, controlled access (key-locked) hand switches provide local manual control for certain ADS valves.

detect low water level in the CST. Either switch can automatically cause suction transfer. To prevent losing suction to the pump, the two suction valves are interlocked so that one suction path must be open before the other closes. See Figure 7.4-2, Sheets 4, 5, and 6 for valve operation logic.

One of the RCIC pump suction automatic switchover level switches is also used to provide CST low-low level indication at the remote shutdown panel (See Section 7.4.1.4.5.2).

The RCIC turbine is functionally-controlled as shown on the RCIC functional control diagram (FCD), Figure 7.4-1. The turbine governor control system limits the turbine speed and adjusts the turbine steam control valve so that design pump discharge flow rate is obtained. The flow signal used for automatic control of the turbine is derived from a differential pressure measurement across a flow element in the RCIC system pump discharge line.

The turbine is automatically tripped and the throttle valve closed if any of the following conditions are detected:

- a. Turbine overspeed
- b. High turbine exhaust pressure
- c. Low pump suction pressure
- d. Auto-isolation signal
 1. High area temperature
 2. Steam line high differential pressure or instrument line break
 3. Steam supply pressure low
 4. Exhaust diaphragm high pressure.

~~e. Reactor vessel high water level~~

Instrument ranges for the RCIC system controls and instrumentation are listed in Table 7.4-1.

HCGS FSAR

sources to the HCGS switchyard, all of which are physically independent sources of offsite power to the HCGS unit. They are:

- a. A tie of approximately 2112 feet between the HCGS and Salem switchyards
- b. A tie of approximately 42.9 miles to the New Freedom switching station
- c. A tie of approximately 30.1 miles to the Keeney switching station.

The maximum winter capacity of each of these lines is 3500 MVA. The maximum summer capacity of each of these lines is 3220 MVA.

The 500-kV switchyard provides preferred power through its interconnections with two sets of two station power transformers, to the 13.8-kV ring bus as shown on Figure 8.3-1. Station power transformers T1 and T4 each supply two 13.8/4.16-kV and one 13.8/7.2-kV station service transformer. Station power transformers T2 and T3 each supply one 13.8/4.16-kV station service transformer ~~and one, 13.8/4.16-kV/208-V~~ station lighting and power transformer. In the event a station power transformer is unavailable, alternate feed is made available to the affected buses.

13.8-kV/
208-120V

AND ONE
13.8/13.8 kV
ISLAND
SUBSTATION
TRANSFORMER.

The offsite power systems and their interconnections are described in Section 8.2.

8.1.2 ONSITE POWER SYSTEMS

The onsite power system for the unit consists of two major categories:

- a. Class 1E power system - The Class 1E power system supplies Class 1E loads that are necessary for safe and orderly shutdown, for maintaining the plant in a safe shutdown condition, and for mitigating the consequences of an accident. A limited number of non-Class 1E loads important to the power generating equipment integrity are also supplied from the Class 1E power system. These non-Class 1E loads are listed in Table 8.3-1.

of the 500-kV bus. Direct stroke lightning protection is provided by overhead 19/#9 Alumoweld shield wires. The control and status indication for the 500-kV main step-up transformer disconnect switch is installed on a breaker control relay rack in the switchyard control house. The switchyard control house will provide an auxiliary switch contact for input to generating station computer systems via a data input/output (I/O) cabinet for status indication. For safety reasons, the control switch for the transformer disconnect switch is provided with a lock-in handle. The generating station control room operator must release the key in his possession to permit operation of this control switch.

8.2.1.4 Switchyard

The 500-kV switchyard, located to the east of the Hope Creek plant, is designed with tapered tubular steel structures and rigid aluminum bus work. This yard consists of two breaker-and-a-half bays containing five SF-6 circuit breakers connected to two 500-kV main buses, 10X and 20X, as shown on Figure 8.2-2. Bus 10X is protected by primary and backup differential relays. Breaker failure relaying detects a failure-to-trip or failure-to-interrupt condition at the line terminal and trips associated breakers necessary to isolate the line. The 500 kV and 13.8 kV circuit breakers are pneumatically operated and each breaker has sufficient stored air for a minimum of three operations without compressor actuation. Compressor motors are supplied by dual ac feeds from separate panels in the switchyard.

The control room and the switchyard control house have independent and simultaneous control of the 500 kV circuit breakers. The electric system operation center, located in Newark, N.J., has limited control of the line breakers 51X, 60X, and 61X and the tie breaker 50X, and no control of the generator breaker 52X.

Restoration of the 500 kV lines after a LOP would generally consist of the following procedural steps:

The system load dispatcher would be contacted to verify availability of 500 kV circuits.

Verify 4 kV and 7.2 kV non-1E bus infeed breakers are opened.

Verification of 500/14.4 kV ~~500/13 kV~~ transformer and 13 kV ring bus breaker positions aligned to restore offsite power.

The load dispatcher is contacted for final clearance to reclose 500 kV breakers.

Once 500 kV power is reestablished, 4 kV and 7.2 kV power is provided to the respective non-1E buses, loading of these non-1E buses can then commence.

Final transfer of Class 1E loads from the standby to the preferred power source can be made when plant conditions are stable.

Generating station auxiliary services are supplied via two 13.8-breaker bays by four 500/14.4 kV, 42/56/70-MVA, oil-immersed, self-cooled/forced-air-forced-oil-cooled (OA/FOA/FOA) three-phase transformers connected to the 500-kV busses 10X and 20X, as shown on Figure 8.2-2. Station power transformers T1 and T4 each supply two 13.8/4.16-kV and one 13.8/7.2-kV station service transformers. The remaining two transformers, T2 and T3, each supply one 13.8/4.16-kV station service transformer ~~and~~ one ~~13.8 kV/208V~~ station light and power transformer. Each 13.8-kV breaker bay consists of three breakers in series. To prevent paralleling of the transformers, one of the breakers is normally open. This breaker is closed in case one of the transformers is out of service.

AND ONE 13.8/
13.8 kV
ISLAND
SUBSTATION
TRANSFORMER

13.8 kV/
208-120V

As shown on Figure 8.2-2, there are six 13.8-kV, 1500-MVA oil circuit breakers. Breaker failure protection detects the failure to trip or failure to interrupt conditions at the line terminals and electrically isolates faulty equipment. Primary and backup relay protection on the 500/14.4-kV station power transformers is provided by the use of harmonic restraint differential relays.

The 13.8-kV system is ungrounded and connected to the delta side of all station power and station service transformers. To detect a phase-to-ground fault in the system, a ~~13.8 kV/120 V~~ grounded-
wye grounding transformer is installed on the secondary side of each station transformer. The neutrals of the grounding transformers are connected to neutral resistors and relays for phase-to-ground fault detection and annunciation.

13.2 kV/
120 V

To locate the fault, a current transformer is installed on each of the 13.8-kV cables. The current transformers are connected to a milliammeter via selector switch to measure the residual cable current and locate the faulty cable.

This configuration of the offsite power system, with provisions for periodic testing, is in full conformance with NRC GDC 17 and 18 of Appendix A to 10 CFR Part 50, which is further discussed in Section 8.3.1.2.1.

HCGS complies with Regulatory Guide 1.32. Clarifications and exception are noted in Section 1.8.

Station light and power transformers SLP 1 and SLP 2 are tapped from 13.8-kV bus sections S3 and S8, respectively. These transformers provide service power to the switchyard via distribution panels located in the switchyard and in the control house. Control power for the protection of switchyard equipment is supplied by two 125-V dc switchyard service batteries (regular and backup), equipped with two full-capacity chargers each. In the event of a relay operation, the relays can be reset and the equipment returned to service within one hour. This scheme ensures that primary and backup relay protection of switchyard electrical equipment will not be lost. The two batteries, dc distribution cabinets and battery chargers are electrically independent and located more than 8 feet apart in the switchyard control house. Control cables from the dc distribution cabinets to regular and backup protection racks run in separate cable trays.

ARE

8.2.1.4.1 13.8-kV Supply

Station power is supplied from the 13.8-kV switchyard via multiple runs of 15-kV, 2000-kcmil power cable in polyvinyl chloride (PVC) conduit. The PVC conduit runs are encased in concrete and run underground from the 13.8-kV feeder positions to in-plant station service transformers. These duct banks are routed to minimize the possibilities of simultaneous failure under operating, postulated accident, and environmental conditions.

8.2.1.5 System Monitoring

PSE&G transmission lines and rights of way are patrolled at least five times each year to ensure that the physical and electrical integrity of transmission line supports, hardware, insulators, and conductors are acceptable for safe and reliable service. This periodic transmission line patrol is conducted by

ISLAND SUBSTATION TRANSFORMERS SLP3 AND SLP4 ARE TAPPED FROM 13.8-KV BUS SECTIONS S4 AND S9 RESPECTIVELY, AND PROVIDE SERVICE FOR 13.8-KV ISLAND POWER DISTRIBUTION SYSTEM VIA 13.8-KV SWITCHGEAR LOCATED IN THE HCGS SWITCHYARD.

helicopter and ground patrols. Climbing inspections of structures are performed at least every 3 years depending on the age of the line.

Monitoring of the offsite power sources in the plant control room is provided for by a hard-wired, console-mounted, mimic bus arrangement that shows the status of station power, station service, ~~and station light and power~~ transformers. Potential indication of the 500/13.8 kV systems and status indication of the transformer secondary and bus tie disconnect switches are provided by the plant computer systems. Control and status indications of all 500-kV and 13.8-kV breakers are also shown. Annunciation accompanies status changes of circuit breakers, loss of potential, transformer trouble, fire protection system actuation, carrier equipment failure, and fault recorder failure.

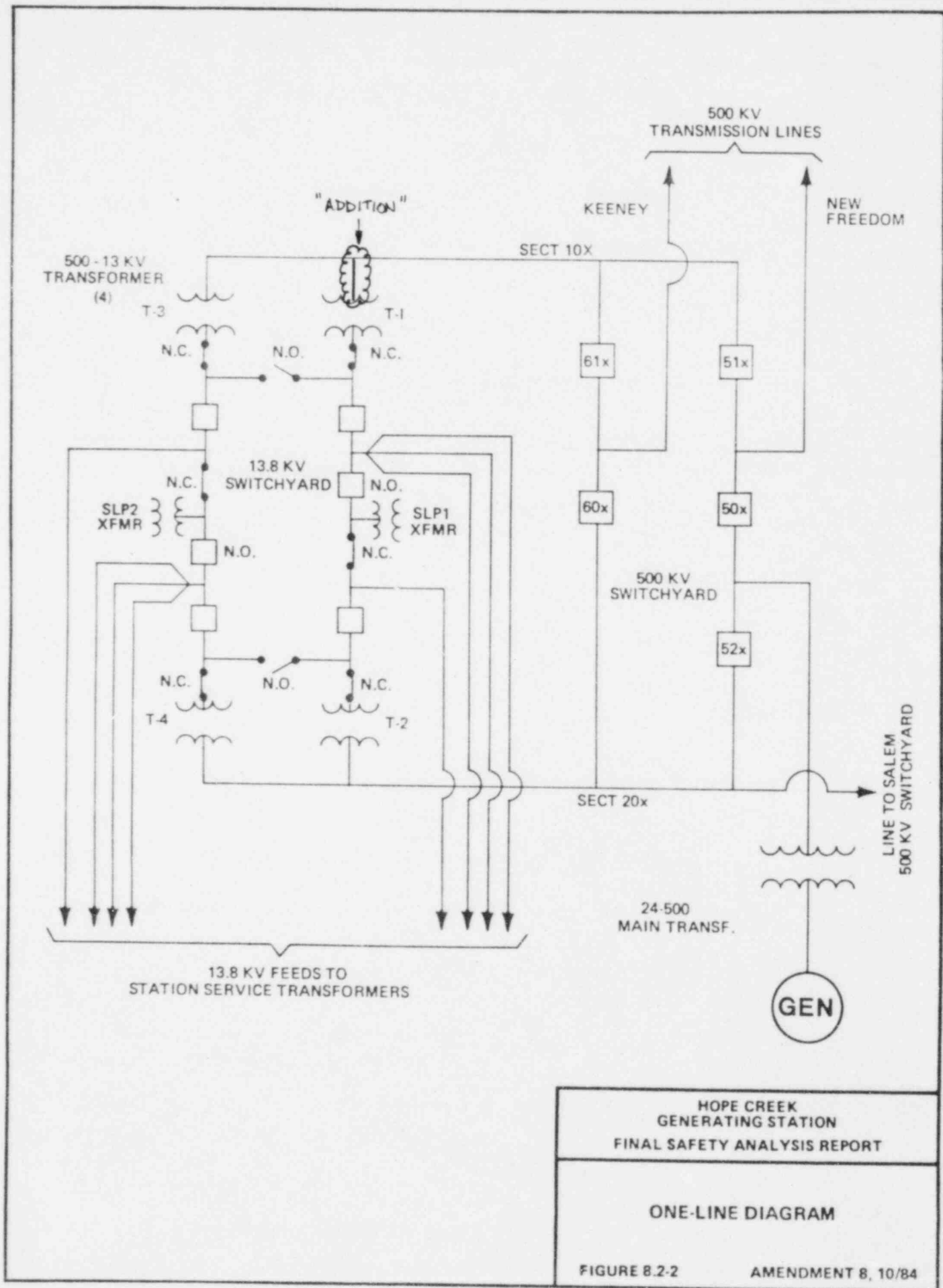
AND

The switchyard fault recorder inputs include phase currents, voltages, and carrier information for all three Hope Creek switchyard offsite power sources. Inputs to the plant fault recorder include the following:

- a. Voltage and current information on the generator and main transformer
- b. Voltage information on the 13.8-kV bus sections 1, 5, 6, and 10
- c. Voltage information on the station service transformers
- d. Voltage information on all 7.2-kV and 4.16-kV buses.

The plant computer system displays additional offsite power system information for the operator on CRTs. Each display is a mimic bus arrangement similar to the hard-wired mimic bus and includes the status of switchyard power circuit breakers (PCBs).

The main generator output leads to the 500-kV switchyard are monitored in the control room. A mimic bus arrangement provides control and status indication of the synchronizing PCB. Potential indication and monitoring of current, watts, volt-amperes reactive (VARs), wathours, and voltage are provided. Annunciation accompanies an abnormal change in the status of the synchronizing PCB and failure of the supervisory system.



HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

ONE-LINE DIAGRAM

FIGURE 8.2-2

AMENDMENT 8, 10/84

8.3 ONSITE POWER SYSTEMS

The onsite power systems consist of ac and dc power systems.

8.3.1 AC POWER SYSTEMS

8.3.1.1 Description

The onsite ac power systems include a Class 1E system and a non-Class 1E system. Figure 8.3-1 is the single line drawing of both the systems.

The onsite ac power is defined in Section 8.1.2.

8.3.1.1.1 Non-Class 1E AC Power System

The non-Class 1E portion of the onsite power system supplies ac power to non-Class 1E loads. A limited number of non-Class 1E loads, important to the power generating equipment integrity, are supplied from the Class 1E distribution system through isolation devices. These non-Class 1E loads are listed in Table 8.3-1.

The offsite power for the plant is fed through the 500-kV system via the 13.8-kV yard ring bus. Two separate buses, 10X and 20X, of the 500-kV switchyard, feed the 13.8-kV ring bus via 500 GND Y/14.4 kV station power transformers, T1, T3 and T2, T4, respectively, as shown on Figure 8.3-1. Physically independent routing of bus 10X and 20X feeders from sections 1 and 2 of the 500-kV switchyard to the station power transformers T1, T3 and T2, T4, minimizes the likelihood of simultaneous failure of the two 500-kV buses. The 13.8-kV ring bus provides auxiliary power during startup, normal operation, shutdown, and post-shutdown operation of the unit. Station power transformers T1 and T4 each feed two 13.8-4.16-kV and one 13.8-7.2-kV station service transformer. Station power transformers T2 and T3 each feed one 13.8-4.16-kV station service transformer ~~and one 13.8-7.2-kV~~ station lighting power transformer. Two 7.2-kV and six 4.16-kV non-Class 1E buses are supplied from the above eight station service transformers. The 7.2-kV buses are 10A110 and 10A120, and the 4.16-kV buses are 10A101, 10A102, 10A103, 10A104, 10A501, and 10A502. The configuration of the non-Class 1E power system is described below for the normal operation of the 13.8-kV ring bus and all the 4.16-kV and 7.2-kV non-Class 1E buses.

AND ONE
13.8/13.8KV
ISLAND
SUBSTATION
TRANSFORMER.

13,800-
2E/100-V

be closed only if one of the infeed breakers of the double-ended unit substation is open.

The 480-V unit substations feed 480-V motor control centers (MCC), motors of 100- to 250-horsepower rating, and 480-V power panels. MCCs supply power to motors of up to 75 horsepower rating, battery chargers, 480/277-V power distribution panels, and 480 and 208/120-V power distribution panels. Uninterruptible power supply (UPS) panels of 120 V ac supply the security system, public address system, NSSS computer, BOP computer, etc. The distribution panels feed miscellaneous loads such as lighting, space heaters, and unit heaters.

The non-Class 1E equipment ratings are listed below:

a. Transformers

1. Main step-up transformer: 3-1 ϕ , 362.5/406 MVA each, FOA 55°C/65°C, 24-500 kV; impedance 16% on 362.5 MVA base, NLTC $\pm 5\%$ in 2-1/2% step
2. Station power transformer: 4-3 ϕ , 42/56/70 MVA each, OA/FOA/FOA, 65°C, 500-14.4 kV, impedance 5.1% on 42 MVA base NLTC $\pm 5\%$ in 2-1/2% step
3. Station service transformers: 2-3 ϕ , 15/20/25 MVA, OA/FOA/FOA 55°C, and 16.8/22.4/28.0 MVA, OA/FOA/FOA 65°C, 13.8-7.2 kV GNDY, impedance 5.5% on 15 MVA base. HV-LTC= -15% to +5%

4-3 ϕ , 17.41/23.21/29 MVA, OA/FOA/FOA 55°C and 19.5/26/32.5 MVA, OA/FOA/FOA 65°C, 13.8-4.16 kV, GNDY, impedance 7.7% on 17.41 MVA base. HV-LTC= -15% to +5%

2-3 ϕ , 14.7/19.6 MVA OA/FA 55°C, 16.5/21.95 MVA OA/FA 65°C, 13.8-4.16 kV, GNDY, impedance 5.48% on 14.7 MVA base, HV-LTC= -15% to +5%
4. Station lighting and power transformer: 2-3 ϕ , 500 kVA, ~~4400~~ 208 V GNDY/120 V

13,800

INSERT

b. Switchgear

1. 7.2-kV switchgear: 1200/2000 A continuous rating, 500 MVA 3 ϕ class, 35,000 A interrupting rating at 8250 V (maximum rated voltage)
2. 4.16-kV switchgear: 1200/2000 A continuous rating 350 MVA 3 ϕ class, 42,400 A interrupting rating at 4760 V (maximum rated voltage)

INSERT 

c. 480-V unit substations

1. Transformers:
 - (a) 1000 kVA, 3 ϕ , 4160-480 V
 - (b) 1500 kVA, 3 ϕ , 4160-480 V
2. Bus: 2000 A continuous rating for 1000 kVA unit substations, 3200 A continuous rating for 1500 kVA unit substations
3. Breakers (metal clad): 30,000 A

d. 480-V MCCs control centers

1. Horizontal bus: 800 A continuous rating, 42,000 A bracing
2. Vertical bus: 300 A continuous rating, 42,000 A bracing
3. Breakers (molded case); 150 A and 250 A frame sizes, 25,000 A, symmetrical rms interrupting rating

INSERT FOR PAGE 8.3-3

5. Island Substation Transformer: 2-3 \emptyset , 12/16 MVA, OA/FA 65°C, 13.8-13.8 kV Y/7970 V, impedance 5.89% (SLP3), 5.94% (SLP4) on 12 MVA base at 13.8 kV.

INSERT FOR PAGE 8.3-4

3. 13.8 kV switchgear: 1200 A continuous rating, 19,300 A short circuit (symmetrical) rating.

HCGS FSAR

Each MCC cubicle derives its 120-V ac control power from a control power transformer located within the cubicle.

8.3.1 1.2.10 Electric Circuit Protection Systems

Protective relay schemes and trip devices on the primary and backup circuit breakers are provided throughout the power system in order to:

- a. Isolate faulted equipment and/or circuits from unfaulted equipment and/or circuits
- b. Prevent damage to equipment
- c. Protect personnel
- d. Minimize system disturbances
- e. Maintain power continuity of power supply in the unaffected part of the system.

The short-circuit protective system is analyzed to ensure that the various adjustable devices are applied within their ratings and set to be coordinated with each other to attain selectivity necessary to isolate a faulted area quickly with a minimum of disturbance to the rest of the system. Major types of protection measures employed include the following:

- a. Differential relaying - Differential relaying schemes are provided for the main generator, main generator-main transformer, station power transformers, station service transformers, ~~√43.8-kV cables from disconnect switches to the station service transformers,~~ 4.16-kV buses, SDGs, motors above 3000 horsepower rating, and 4.16- and 7.2-kV buses. These schemes provide high-speed disconnection by opening appropriate breakers to prevent severe damage in case of faults occurring within the bounds of the areas served by these relays.
- b. Overcurrent relaying - Each Class 1E 4.16-kV bus incoming feeder circuit breaker is equipped with three

ISLAND
SUBSTATION
TRANSFORMERS,

TABLE 8.3-1 (cont)

Item	Description	Equipment No.	Rating, each, hp	Operating kW each(5)	Number Connected To Class 1E Distribution System				Loading Sequence(2)			
					Diesel Buses				Time		Time	
					A	C	B	D	Min No.	From DBA(13)	Min No.	From LOP(14)
Class 1E loads												
16.	RB FRVS recirculation system fans	1A-V213 thru 1F-V213	150	120	2	1	2	1	4	19 s 30 s(7)	-	-
17.	Control room supply fans	1A-VH403 1B-VH403	40	32	-	1	-	1	1	30 s	1	30 s
18.	208Y/120-V ac XPMRS to power dist panels	10X201,202,203 204 10X411,412,413 414 10X421,422,423, 424 10X501,502,503, 504	-	75 (4 sets)	4	4	4	4	12	13 s	12	13 s
19.	Deleted											
20.	Intake structure exhaust fans	1A,B,C,D-V504	40	32	1	1	1	1	2	13 s(15)	2	13 s(15)
21.	Control room chilled water circulating pumps	1A-P400 1B-P400	60	48	-	1	-	1	1	65 s	1	65 s
22.	Control room supply unit heating coils	1A-VH403 1B-VH403	-	90	-	1	-	1	1	60 s	1	60 s
23.	Control room water chillers	1A-K400 1B-K400	680	506	-	1	-	1	1	100 s (16)	1	60 s
24.	Diesel generator room recirc system fans	1A-V412 thru 1H-V412	125	100	2	2	2	2	3	30 s(9)	3	30 s
25.	Primary containment instrument gas compressor	1A-K202 1B-K202	15	12	-	1	-	1	1	30 min	1	30 min
26.	Battery chargers, 250-V dc	10D423 10D433	-	15	1	-	1	-	2	13 s	2	13 s
27.	Control area battery room exhaust fans	1A-V410 1B-V410	5	4	-	1	-	1	1	60 s	1	60 s
28.	RB FRVS recirculation unit unit heating coils	1A-VH213 thru 1F-VH213	-	100	2	1	2	1	4	19 s 30 s(7)	-	-

TABLE 8.3-1 (cont)

Page 4 of 10

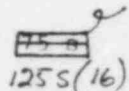
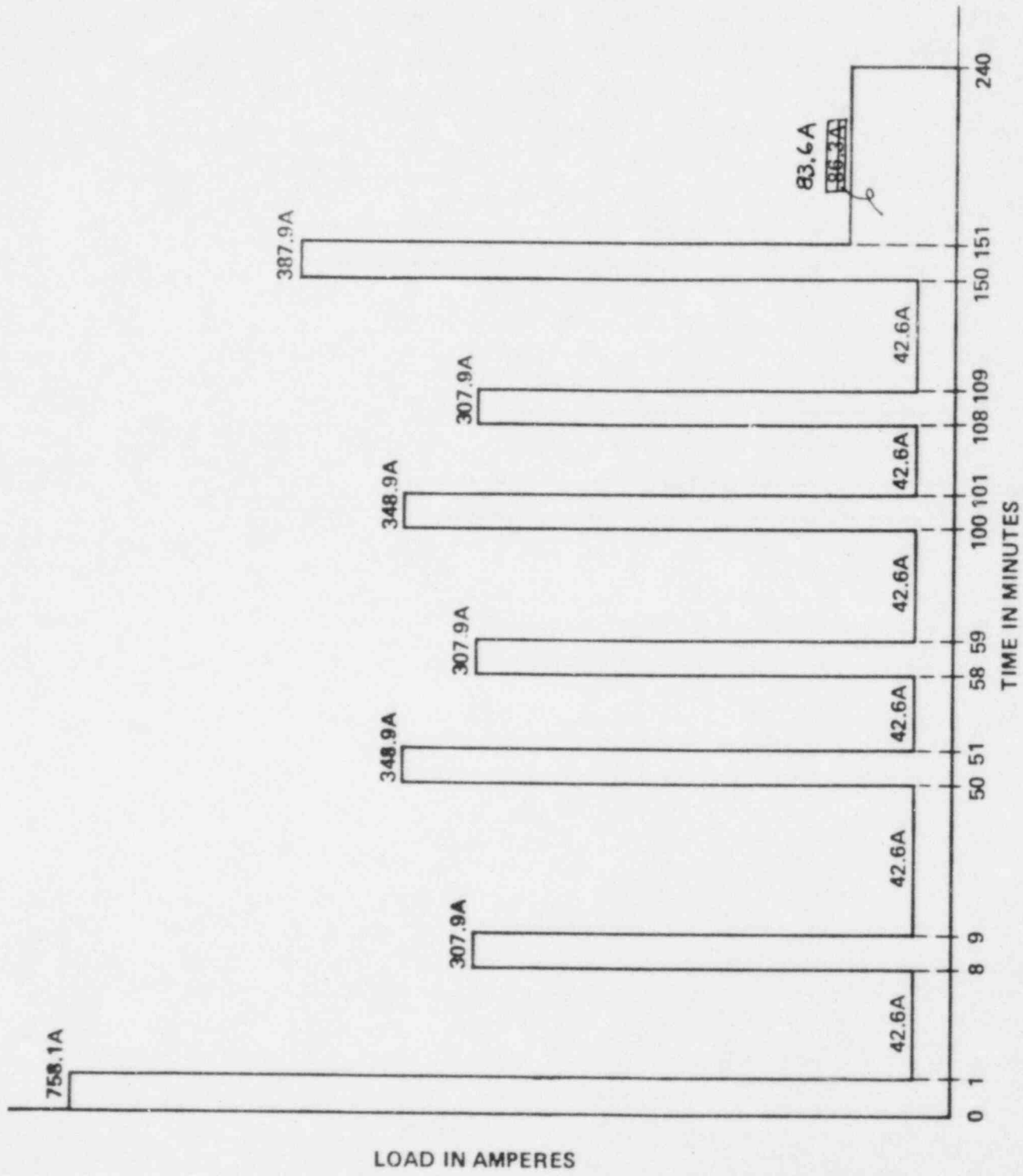
Item	Description	Equipment No.	Rating, each, hp	Operating kW each ^(*)	Number Connected To Class 1E Distribution System Diesel Buses				Loading Sequence ⁽²⁾			
					A	C	B	D	Min No.	Time From DBA ⁽¹⁾	Min No.	Time From LOP ⁽¹⁾
Class 1E Loads												
45.	Deleted.											
46.	480-V power supply to Class 1E chiller panels	1AC488, 1BC488 1AC491, 1BC491	-	4	1	1	1	1	2	13 s	1	13 s
47.	Traveling screens	1A-S501 1B-S501 1C-S501 1D-S501	5	4	1	1	1	1	3	55 s	3	55
48.	ECCS jockey pump	1A-P228 1B-P228 1C-P228 1D-P228	10	8	1	1	1	1	3	13 s	3	13 s
49.	Motor-driven diesel generator fuel oil standby pumps	1A-P402 1B-P402 1C-P402 1D-P402	2	1.6	1	1	1	1	3	13 s	3	13 s
50.	Standby liquid control pump room duct heaters	1A-VE261 1B-VE261	-	45	1	-	-	1	1	15 min	1	15 min
51.	480-V power supply to hydrogen and oxygen analyzer panels	1A-C200 1B-C200	-	1	1	-	1	-	1	13 s	1	13 s
52.	250-V dc battery room duct heaters	10-VE418	-	10	1	-	-	-	1	13 s	1	13 s
53.	125-V dc diesel area battery room duct heaters	1A-VE420 1B-VE420 1C-VE420 1D-VE420	-	21	1	1	1	1	3	13 s	1	13 s
54.	HPCI pump room duct heater	10-VE260	-	11	1	-	-	-	1	13 s	1	13 s
55.	RCIC pump room duct heaters	10-VE259	-	7	-	-	1	-	1	13 s	1	13 s
56.	250-V dc battery room duct heater	10-VE417	-	8	-	-	1	-	1	13 s	1	13 s
57.	Class 1E panel room water chillers	1A-K403 1B-K403	268	198	1	-	1	-	1		1	75 s

TABLE 8.3-1 (cont)

- (2) Loading sequence is based on availability of three standby diesel generators and their associated electric power distribution systems.
- (3) DBA = Design basis accident
LOCA = Loss-of-coolant accident
LOP = Loss of offsite power
DBA = LOCA + LOP
- (4) MOV's maximum stroking time will vary from 20 to 70 seconds except for the main steam stop valves, with a stroking time of 120 seconds.
MOV loads are not included in this table and the diesel generator loading tables that follow because of their small magnitude and short period of operation.
- (5) "Operating Kw" is taken as 0.8 motor hp rating for motors 250 hp and smaller
- (6) During a DBA, any two core spray pumps and three RHR pumps can be manually tripped after 10 minutes from the occurrence of LOCA, depending upon the load on each standby diesel generator. Either the A or the B RHR pump must be retained in service after 10 minutes from the occurrence of LOCA.
- (7) Buses A and B each have two FRVS recirculating fans and two unit heating coils connected to them. Buses C and D each have one FRVS recirculating fan and one unit heating coil connected to them. In the case of a DBA, one fan and one unit heating coil will start on each of buses A, B, C, and D and the remaining fans and heating coils will start in diesel buses A and B at the times shown in the loading chart.
- (8) Deleted.
- (9) In case the lead fan fails to start, the lag fan will start automatically at 95 seconds.
- (10) Deleted.
- (11) Upon the occurrence of a LOCA, non-Class 1E loads are tripped by LOCA signals in 3 seconds by tripping the unit substation circuit breakers feeding the non-Class 1E MCCs and motors. These loads can be reenergized manually at 10 minutes after the occurrence of LOCA.
- (12) Two redundant 25-hp pumps are provided.
- (13) Times shown are from the occurrence of LOCA or loss of offsite power.
- (14) Loads are not sequenced but power is required to be available to the loads in the event of LOP or be available within 25 seconds after an ATWS initiation signal. The time shown (13 sec) is when power is available after LOP.
- (15) Loads are not sequenced but are controlled by process signals. For SDG loading purposes, these loads are assumed to start and run after 13 seconds from the DBA/LOCA event.

(16) TIME INDICATED IS SEQUENCE TIME REQUIRED IF CHILLERS ARE OFF PRIOR TO DBA OR LOP.
IF CHILLERS ARE ON PRIOR TO DBA OR LOP, SEQUENCE TIME REQUIRED IS 160 SECONDS.



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GENERATING STATION
FINAL SAFETY ANALYSIS REPORT


CLASS I E 250 VDC
BATTERY LOAD
PROFILE

FIGURE 8.3-16
SHEET 7 OF 8

AMENDMENT 8, 10/84

FIGURE 9.1-10 (CONTINUED)

OVERHEAD

Item Number	Crane or Hoist	Tag Number	Building	Floor Elev (ft)	Equipment Loc Fig Number	Col Ar
1	Reactor building polar crane	10H200	Reactor	201	1.2-32	N-V
2	Personnel air lock hoist	10H217	Reactor	102	1.2-28	P-R
3	Recirculation pump motor hoist	1AH201 1BH201	Reactor	102 (Drywell)	1.2-28	SA- SA-
4	Reactor water clean-up filter/demineralizer hoist	1AH220 1BH220	Reactor	178-6	1.2-31	R-Q
5	HPCI pump and turbine hoist	1AH211 1BH211	Reactor	54	1.2-26	W-V
6	RCIC pump and turbine hoist	10H212	Reactor	54	1.2-26	W-V
7	Main steam tunnel underhung crane	10H214 10H223	Reactor	102	1.2-28	P-Q
8	Inboard MSIV hoist	10H203	Reactor	102 (Drywell)	1.2-28	Q-R
9	Vacuum breaker valve removal hoist	10H207	Reactor	54 (Torus)	1.2-27	N-V
10	Main steam line relief valve removal hoist	10H202	Reactor	135-6 (Drywell)	1.2-29	
11	Turbine building bridge crane	10H102	Turbine	137	1.2-16	E-F
12	Feedwater heater removal hoist	1AH103 1BH103	Turbine	102	1.2-14	E-Eg
13	H&V equipment removal hoist	10H104	Turbine	171	1.2-17	E-F
14	Motor-generator set hoist	0AH105 0BH105	Turbine	137	1.2.-16	Eu-B

BOGS PSAR
TABLE 9.1-10

VY LOAD HANDLING SYSTEMS DATA SUMMARY

	Capacity (tons)	MAX Vert Lift (ft in)	Seismic Cat I	Design Standard(2)	Is Load Over Safety-Related(5) Equipment?	Is Load Over Safety-Related(5) Equipment on Next Lower Elev	Exclusion Criterion(1)
-23R	150 main 10 aux	129-0	Yes	a, b	Yes	Yes	None
-23R	30	16-1	No(3)	c, d	No	Yes	None
-20R -9-17R	24	12-0	No(3)	c, d	Yes	Yes	None
-17R	10	26-0	No	d	No	No	B
-21R	4	9-10	No(3)	c, d	Yes	NA	None
-18P	3	9-0	No(3)	c, d	Yes	NA	C
20	2-1/2	16-0 (10H214) 10H223	No(3)	a, d(4)	Yes	Yes	None
20R	2	16-0	No(3)	d	Yes	Yes	None
-22R	2	7-0	No	d	Yes	No	C
20	1	32-2	No(3)	c, d	No	Yes	C
220 main 45 aux	72-3 main 122-0 aux		No	a, b	No	No	B
22	24	12-6	NO	d	NO	No	B
3	15	37-0	No	c, d	NO	No	B
29	15	16-5	NO	c, d	NO	NO	B

Q-T; 17R-20R

36-0

Amendment 14

TABLE 9.3-5

STANDBY LIQUID CONTROL SYSTEM OPERATING PRESSURE/TEMPERATURE CONDITIONS

Piping	Standby Mode ⁽¹⁾		Test Modes ⁽¹⁾				Operating Mode ⁽¹⁾	
	Pressure, psig ⁽³⁾	Temperature, °F	Pressure, psig ⁽³⁾	Temperature, °F	Pressure, psig ⁽³⁾	Temperature, °F	Pressure, psig ⁽³⁾	Temperature, °F
Pump suction	Storage tank static head	70/110 ⁽⁴⁾	Test tank static head ⁽⁵⁾	70/110 ⁽⁴⁾	Test tank static head ⁽⁵⁾	70/110 ⁽⁴⁾	Storage tank static head	70/110 ⁽⁴⁾
Pump discharge to explosive valve inlet	0	70/110	0/1190	70/100	40 (plus reactor static head)	70/100	40 (plus reactor static head) to 125	70/110
Explosive valve outlet to, but not including, motor-operated stop check globe valves	Reactor static head to 1140 ⁽⁶⁾	70/110	Reactor static head to 1140 ⁽⁶⁾	70/100	<40 (plus reactor static head)	70/100	<40 (plus reactor static head) to <125	70/110
Motor-operated stop check globes valves to core spray line	Reactor static head to 1140 ⁽⁶⁾	70/560 ⁽⁷⁾	Reactor static head to 1140 ⁽⁶⁾	70/560 ⁽⁷⁾	Reactor static head ⁽²⁾	≤ 125 ⁽²⁾	Reactor static head to 1140 ⁽⁶⁾	70/560 ⁽⁷⁾

(1) The pump flow rate is zero (pump not operating) during the standby mode, at 43 gpm during the test modes (one pump operation) and is at rated system flow during the operating mode (two pump operation).

(2) Reactor to be at 0 psig and 125°F before changing from the standby mode to the injection test mode.

(3) Pressures tabulated represent pressure at the points identified below. To obtain pressure at intermediate points in the system, the pressures tabulated must be adjusted for elevation difference and pressure drop between such intermediate points and the pressure points identified below:

Piping

Pressure Point

Pump suction

Pump suction flange inlet

Pump discharge to explosive valve inlet

Pump discharge flange outlet

Explosive valve outlet to, but not including, motor-operated stop check globe valves

Explosive valve outlet

11

STEAM DOME PRESSURE

13.5 PLANT PROCEDURES

Plant procedures are prepared by the plant staff, support organizations, or contract organizations under the direction of the General Manager - Hope Creek Operations and implemented by the Operations Manager, Maintenance Manager, Technical Manager, and Radiation Protection/Chemistry Manager.

Plant procedures are prepared for applicable safety-related activities delineated in Regulatory Guide 1.33 and provide the controls necessary to comply with applicable Regulatory Guides as listed in Section 1.8.

Preparation of plant procedures necessary for fuel load has begun and is scheduled to be completed ~~six months~~ prior to the event. Review of procedures ~~affecting nuclear safety~~ and changes thereto are performed by the Station Operations Review Committee (SORC) and approved by the individual department managers for Hope Creek Operations.

Procedures are periodically reviewed and revised when changes are necessary or desirable. Similarly, procedures are reviewed and revised when necessary following the completion of system design changes or equipment modifications. Subsequent to an abnormal occurrence, e.g., an unexpected plant transient, significant operator error, etc, the appropriate procedure(s) receive a review. The purpose of this review is to ascertain whether the procedure may have contributed to the cause of the abnormal occurrence or was adequate in its capacity to mitigate the consequences.

Circumstances may develop during maintenance, operation, or testing of the station systems when an existing instruction or procedure is not entirely applicable as written or otherwise interferes with performance. In these instances, the existing procedure may be temporarily changed. All such changes to procedures are made so as not to change the intent of the existing procedure and are reviewed and approved by two members of station management, knowledgeable in the area affected, prior to its implementation. ~~For changes to procedures which may affect the operational status of plant systems or equipment, one member shall hold an SRO license.~~

INSERT A FOR PAGE 13.5-1

A technical review and control system utilizing qualified reviewers functions to perform periodic or routine review of procedures.

INSERT B FOR PAGE 13.5-1

Station Administrative Procedures, changes thereto and implementing

INSERT C FOR PAGE 13.5-1

that involve a significant safety issue

INSERT D FOR PAGE 13.5-1

The members of station management must be knowledgeable in the area affected and at least one must hold an SRO license.

13.5.1 ADMINISTRATIVE PROCEDURES

Station administrative procedures are written to provide stationwide direction in areas that are common to all station departments. Administrative procedures are prepared using the following format:

- a. 1.0 Purpose
- b. 2.0 References
- c. 3.0 Definitions
- d. 4.0 Responsibilities
- e. 5.0 Procedure
- f. List of Attachments and Forms

Furthermore, Section 5.0 may be subdivided, as appropriate, to facilitate the use of a specific procedure. Additional administrative procedures may be written as required. The following is a list of station administrative procedures:

- a. SA-AP.77-001 Preparation and Approval of Station Procedures

This procedure describes the methods used in preparation (format and organization), indexing, ~~review, and approval~~ of all HCGS procedures.

~~Furthermore, this procedure defines the level of review and approval required for each individual procedure or procedure category with respect to its designation as a functional unit procedure, and whether or not the procedure affects Q, R or F designated items.~~

- b. SA-AP.77-002 Station Organization and Operating Practices

AND
REVISION

establishes the administrative controls required for implementing and maintaining such a program. These controls are generally applicable but may be altered by the various departments using approved implementing procedures to meet their specific needs.

INSERT

V. ~~y~~. SA-AP.ZZ-024 Radiological Protection Program

An overview of the radiation protection program is provided, which includes pertinent information and instructions to plant staff, as dictated by 10 CFR 19. Additional requirements and instructions which affect all or a majority of the station staff are also specified. These include: ALARA requirements, dosimetry and exposure limits, emergency exposure criteria, radiation work permit guidelines, and radiological protection requirements for visitors.

W. ~~y~~. SA-AP.ZZ-025 Station Fire Protection Program

This procedure provides a program for prevention, detection, and control of fire hazards, the safeguarding of life, and prevention of property damage or loss due to fire at HCGS.

X. ~~y~~. SA-AP.ZZ-026 Nuclear Mutual Limited and Jurisdiction Boiler and Machinery Inspection Program

This procedure identifies the requirements of, and assigns responsibilities for, complying with inspection and testing requirements of the station's insurance program.

Y. ~~x~~. SA-AP.AA-027 Station Inservice Inspection Program

This procedure establishes administrative controls, identifies general requirements, and assigns responsibilities for inservice inspection (ISI) commitments of the HCGS.

Z. ~~y~~. SA-AP.ZZ-028 Reporting of Defects and Noncompliances

INSERT FOR PAGE 13.5-7

u. SA-AP.ZZ-023 Scaffolding Program

This procedure provides administrative controls for the erection and use of scaffolding in the plant.

This procedure describes the action necessary to implement NRC 10 CFR 21, Reporting of Defects and Noncompliances.

aa. z. SA-AP.ZZ-029 Radioactive Waste and Material Control

This procedure delineates responsibilities for the use, unconditional release, collection, processing, handling, packaging, inspection, receipt, storage and shipping of radioactive waste and material.

bb. aa. SA-AP.ZZ-030 Station Response and Commitment Control Program

This procedure establishes a program for tracking, implementing review, and providing resolution to various action items brought to the attention of the Hope Creek Generating Station

cc. bb. SA-AP.ZZ-031 Station Housekeeping Program

This procedure delineates the housekeeping responsibilities and controls used to ensure the cleanliness of facilities, materials, and equipment used at HCGS.

dd. cc. SA-AP.ZZ-032 ~~Revisions and Changes to~~ Station Procedures

This procedure describes the methods used in ~~recommending, as well as implementing, permanent and temporary (on-the-spot) changes to approved procedures and documents.~~

ee. dd. SA-AP.ZZ-033 Station Security Program

The requirements of the security plan, as described in Section 13.6 applicable to station personnel, are discussed in this procedure. In addition, description of responsibilities and authorities, employee and visitor access control criteria, security badging

REVIEW AND APPROVAL OF

AND PROCEDURE REVISIONS

INSERT

INSERT FOR PAGE 13.5-8

Reviewing and approving newly created and changed station procedures.

system, physical security system, etc, are also provided. Security documents contain safeguard information and thus their distribution is strictly limited.

ff. ~~ee.~~ SA-AP.ZZ-035 Station Reporting Requirements

This procedure serves as a reference for the station staff to indicate which reports are required by federal, state, and local agencies, and provides the administrative controls necessary for ensuring that such reports are prepared, reviewed, and submitted correctly.

gg. ~~ff.~~ SA-AP.ZZ-036 Phase III Startup Test Program |

This procedure defines the responsibilities and procedures used during Phase III - Plant Operational Testing, which includes initial fuel load and the tests requiring the application of nuclear heat. The testing activities will be listed with a brief overview of their scope. This procedure will be deleted at the start of the first refueling.

hh. ~~gg.~~ SA-AP.ZZ-037 Environmental Control Program

This procedure provides an overview of the basic philosophies, policies, and objectives of HCGS to systematically control environmental conditions to avoid accidental discharges, thereby minimizing environmental impact of plant operations.

ii. ~~hh.~~ SA-AP.ZZ-038 Control of Materials Usage Program

This procedure establishes controls for the use and disposal of those chemicals that can be harmful to personnel or cause damage to plant systems or equipment.

jj. ~~ii.~~ SA-AP.ZZ-039 Receipt of New Fuel |

This procedure identifies major refueling tasks and defines those departmental duties and responsibilities necessary to conduct refueling operations.

kk. ~~jjj~~. SA-AP.ZZ-040 Master Equipment List

This procedure defines the requirements for the control and use of the Hope Creek Master Equipment List.

ll. ~~kkk~~. SA-AP.ZZ-041 Confined Space Entry

This procedure specifies the methods for personnel protection for entering confined spaces.

mm. ~~lll~~. SA-AP.ZZ-042 Station Field Questionnaires

This procedure describes the use of the FQ to request Site Engineering Support.

nn. ~~mmm~~. SA-AP.ZZ-043 Operability and Maintainability Enhancement

This procedure establishes a mechanism to document a review of station design to ensure compatibility with operating requirements and recommending improvements to enhance operability and maintainability of the as-built plant.

oo. ~~nnn~~. SA-AP.ZZ-044 Station Aids Program

This procedure describes the approval, documentation, and review requirements to ensure Station Aids are current, complete, and necessary.

pp. ~~ooo~~. SA-AP.ZZ-045 Respirator Protection

PROGRAM

This procedure provides methods for personnel protection related to the inhalation of both radiological and nonradiological agents.

99. ~~999~~. SA-AP.ZZ-046 Radiological Access Control

This procedure describes the methods, policies, guidelines, limits and requirements for Restricted Area and Radiological Control Area access.

11. ~~999~~. SA-AP.ZZ-047 Operating Experience Evaluation

This procedure establishes a program which provides rapid dissemination of information pertaining to industry operating experience.

33. ~~999~~. SA-AP.ZZ-048 Station Performance and Reliability Monitoring

This procedure establishes a program for monitoring and trending plant process data to identify reductions in unit efficiency or component performance, and to further evaluate the root cause and recommend appropriate corrective actions.

OPERATIONS

44. ~~999~~. SA-AP.ZZ-049 Conduct of Refueling and Core Alterations

This procedure describes the administrative controls and departmental responsibilities during fuel handling and core alterations.

44. ~~999~~. SA-AP.ZZ-050 Station Retest Program

This procedure establishes guidelines for determining post-maintenance retest procedures.

44. ~~999~~. SA-AP.ZZ-051 Leakage Reduction Program

This procedure describes the program for reducing leakage from systems located outside containment that would contain highly radioactive fluids following an accident.

ww. ~~344~~. SA-AP.ZZ-052 Chemistry Control Program

This procedure establishes the program for ensuring that the water chemistry requirements for the NSSS and NSSS support systems are maintained within vendor and industry standards.

In addition to these station administrative procedures, operationally oriented administrative procedures provide guidelines for the operations senior shift supervisors and their shift crews, as well as procedures for night order book usage and control. Operations administrative procedures meet the requirements of 10 CFR 50.54(i), (j), (l), and (m). Figure 13.5-1 indicates the main control room area designated as "at the controls," the area restricted to licensed personnel and the limitations of the reactor operator while manipulating the controls.

13.5.2 OPERATING AND MAINTENANCE PROCEDURES

The operating and maintenance procedures meet the relevant requirements as discussed in Section 1.8.

It is planned that most operating and maintenance procedures will be completed at least three months prior to fuel load and will be available for review in advance draft form at least six months prior to fuel load. This will provide sufficient lead time to ensure that plant personnel can become familiar with them. Where practical the preoperational testing phase will be used to demonstrate the adequacy of the operating procedures.

13.5.2.1 Main Control Room Operating Procedures

The following categories delineate those procedures that are performed primarily within the main control room. Operator familiarization with these procedures is acquired through initial, requalification and replacement training programs. Furthermore, these procedures will be utilized in simulator training.

13.5.2.1.1 System Operating Procedures

The procedures for startup, operation and shutdown of safety-related BWR systems at HCGS will be called System Operating Procedures (SOPs). SOPs will be developed to cover the operating activities listed in Regulatory Guide 1.33, Appendix A, item 4 and will include the following procedures:

Reactor Pressure Control System Malfunction |

Reactor Vessel Level Control System Malfunction |

Neutron Monitoring System Malfunction |

Fuel Pool Cooling and Cleanup System Malfunction |

Standby Liquid Control System Initiation |

13.5.2.1.4 Alarm Response Instructions |

Alarm response instructions guide operators in their response to main control room alarm conditions. The alarm system at HCGS consists of control room overhead annunciators, console pushbutton alarms, computer (digital) alarms, and local/back panel alarms. A color code system (red, amber and white) is utilized for prioritizing control room overhead annunciators. The computer and local panel alarms are associated with an overhead annunciator. The priority of these alarms would be the same as the associated overhead alarm.

The alarm response procedures will be available in the main control room for the operators use. These procedures will be compiled in a manner which is consistent with the alarm system layout in the control room. For example, the overhead annunciator response procedures will be indexed by window box identification number. |

13.5.2.1.5 Temporary Procedures |

Temporary procedures may be issued to direct operations during activities such as testing, refueling, maintenance, and modifications, and to provide guidance in unusual situations not within the scope of existing procedures. Temporary procedures ~~are only in effect for specified periods of time, and~~ require the same review and approval process as other plant procedures, including independent review, as described in Section 13.4.

13.5.2.2 Additional Operating and Maintenance Procedures

The following categories delineate those procedures that are performed primarily outside the limits of the main control room.

b. Prerequisites

Section 14.2.10 (initial fuel loading) describes the prerequisites for commencing fuel loading.

c. Test Procedure

AND — The fuel loading procedure includes tests performed during the fuel loading evolution, including subcriticality checks, shutdown margin demonstration, and control rod functional tests.

d. Acceptance Criteria

Level 1:

INSERT — The partially loaded core shall be subcritical by at least 0.38% $\Delta K/K$ with the analytically determined strongest rod fully withdrawn.

14.2.12.3.4 Full Core Shutdown Margin

a. Objective

The test objective is to demonstrate that the reactor will remain subcritical throughout the first fuel cycle with the most reactive control rod fully withdrawn.

b. Prerequisites

The core is fully loaded and in the xenon-free condition.

c. Test Method

The shutdown margin demonstration is performed by withdrawing selected control rods until criticality is reached. The empirical data are used to correct calculated values to obtain true shutdown margin.

INSERT FOR PAGE 14.2-156

or by at least 0.38% delta k/k with the reactivity equivalent of the strongest rod added by the withdrawal of other control rods.

times following planned reactor scrams as detailed on Figure 14.2-5. In addition, proper response of the CRD flow control valve will be verified.

d. Acceptance Criteria

Level 1

The normal withdrawal speeds and scram times shall meet the requirements of the GE startup test specifications.

Level 2

The friction test results should meet the requirements of the GE startup test specifications.

14.2.12.3.6 Source Range Monitor Performance

a. Objective

The test objective is to demonstrate that the neutron sources, SRM instrumentation, and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner.

b. Prerequisites

Fuel loading is complete, ~~neutron sources have been installed, and~~ all control rods ~~have been~~ inserted, The CRD system is operational.

AND

ARG

c. Test Method

With the neutron sources installed, source range monitor count rate data is taken and compared to the required signal count and signal count-to-noise count ratio. Source range data is taken during rod withdrawals to the point of criticality. Rods will be withdrawn in accordance with a pre-established withdrawal sequence. Movement of rods in a prescribed sequence is monitored by the RWM and RSCS which prevent out of sequence movement.

d. Acceptance Criteria

Level 1

There must be a neutron signal count-to-noise count ratio of at least two and a minimum neutron count rate of 0.7 counts/second on the required operable SRMs.

14.2.12.3.7 Rod Sequence Exchange

This Test Has Been Deleted

14.2.12.3.8 Intermediate Range Monitor Performance

a. Objective

The test objective is to determine the IRM system overlap with the SRMs and APRMs and adjust the IRMS as required.

b. Prerequisites

The reactor is critical and the IRM gains have been optimized. ~~For conservatism~~

SPEED

above the reactor pressure to simulate the largest expected pipeline pressure drop. This CST testing is done to demonstrate general system operability and for making ~~test~~ controller adjustments.

Reactor vessel injection tests follow to complete the controller adjustments and to demonstrate automatic starting from a cold standby condition. "Cold" is defined as a minimum 72 hours without any kind of RCIC operation. Data will be taken to determine the RCIC high steam flow isolation trip setpoint while injecting at rated flow to the reactor vessel.

After all final controller and system adjustments have been determined, a defined set of demonstration tests must be performed. Two consecutive reactor vessel injections starting from cold conditions in the automatic mode must satisfactorily be performed to demonstrate system reliability. Following these tests, a set of CST injections are done to provide a benchmark for comparison with future surveillance tests.

After the auto start portion of certain of the above tests is completed, and while the system is still operating, small step disturbances in speed and flow command are input (in manual and automatic mode respectively) in order to demonstrate satisfactory stability. This is to be done at both low (above minimum turbine speed) and near rated flow initial conditions to span the RCIC operating range.

A demonstration of extended operation of up to two hours (or until ~~pump and~~ turbine oil temperature is stabilized) of continuous running at rated flow conditions is to be scheduled at a convenient time during the startup test program.

Depressing the manual initiation pushbutton is defined as automatic starting or automatic initiation of the RCIC system.

d. Acceptance Criteria

Level 1:

1. Following automatic initiation, the pump discharge flow must be equal to or greater than rated flow as specified in Section 5.4.6 within the time specified by the GE startup test specification.

which are based on not exceeding ASME Section III Code stress values. These specified displacements will be used as acceptance criteria in the appropriate startup test procedures.

Level 2:

1. All hangers and snubbers shall be within their normal operating range.
2. The displacements at the established transducer locations shall not exceed the expected values as provided by the piping designer.

14.2.12.3.16 [DELETED]

14.2.12.3.16 TIP Uncertainty

a. Objective

The test objective is to demonstrate the reproducibility of the TIP system readings.

b. Prerequisites

The core is at steady-state power level with equilibrium xenon, so as to require no rod motion or change in core flow to maintain power level during data acquisition by the TIP system.

c. Test Method

1. Core power distribution data are obtained during the power ascension test program. Axial power distribution data are obtained at each TIP location. At intermediate and higher power levels, several sets of TIP data are obtained to determine the overall TIP uncertainty.
2. TIP data are obtained with the reactor operating with a symmetric rod pattern and at steady-state conditions. The total TIP uncertainty for the

test is calculated by averaging the total TIP uncertainty determined from each set of TIP data. The TIP uncertainty is made up of random noise and geometric components.

d. Acceptance Criteria

Level 2:

The total TIP uncertainty shall be within the specified limits required in the GE startup test specification.

14.2.12.3.17 Core Performance

a. Objective

The test objective is to evaluate the principal thermal and hydraulic parameters associated with core behavior.

b. Prerequisites

The plant is operating at a steady-state power level.

c. Test Method

With the core operating in a steady-state condition, the core performance evaluation is used to determine the following principal thermal and hydraulic parameters associated with core behavior:

1. Core flow rate
2. Core thermal power level
3. MLHGR
4. MCPR

Electric startup test specification requirements when individually testing the MSIVs.

2. The RCIC and HPCI systems shall function in accordance with the GE startup test specification following the MSIV closure from high power.

14.2.12.3.24 Relief Valves

a. Objectives

1. To demonstrate proper operation of the main steam relief valves and verify that there are no major blockages in the relief valve discharge piping.
2. To demonstrate their leaktightness following operation.

b. Prerequisites

The reactor is on pressure control with adequate bypass or main steam flow to maintain pressure control throughout the relief valve opening transient.

c. Test Method

A functional test of each safety relief valve (SRV) shall be made ~~as early in the startup program as practical.~~ This test is normally performed during the first heatup cycle. The test is then repeated at rated reactor pressure if necessary. ~~response is monitored during the low pressure tests and~~ Bypass valves (BPV) ~~and~~ the electrical output response is monitored during ~~rated~~ pressure tests. ~~and~~ The test duration will be about 10 seconds to allow turbine valves and tailpipe sensors to reach a steady state.

AT RATED REACTOR PRESSURE BETWEEN 10 AND 20% OF RATED THERMAL POWER.

OR

THE

The tailpipe sensor responses will be used to detect the opening and subsequent closure of each SRV. The BPV ~~and~~ Mwe responses will be analyzed for anomalies indicating a restriction in an SRV tailpipe.

OR

Valve capacity will be based on certification by ASME code stamp and the applicable documentation being

available in the onsite records. Note that the nameplate capacity/pressure rating assumes that the flow is sonic. This will be true if the back pressure is not excessive. A major blockage of the line would not necessarily be offset and it should be determined that none exists through the BPV response signatures.

OR MWe

Vendor bench test data of the SRV capacity and setpoint is evaluated during preoperational testing. The acoustic monitoring subsystem will be monitored during the relief valve test program and planned reactor trips.

d. Acceptance Criteria

Level 1:

1. There should be positive indication of steam discharge during the manual actuation of each valve.

Level 2:

1. Decay ratio for pressure control variables is as specified in the GE startup test specification.
2. The temperature measured by thermocouples on the discharge side of the valves should return to the temperature recorded before the valve was open as required in the GE startup test specification.
3. During the ~~Reduced and rated pressure~~ functional tests, steam flow through each relief valve as compared to average relief valve flow is as specified in the GE startup test specification.

14.2.12.3.25 Turbine Trip and Generator Load Rejection

a. Objective

The test objective is to demonstrate the proper response of the reactor and its control systems following trips of the turbine and generator.

b. Prerequisites

Power testing has been completed to the extent necessary for performing this test. The plant is stabilized at the required power level.

c. Test Method

This test is performed at ~~three~~ ^{two} different power levels in the power ascension program. For the turbine trip, the main generator remains loaded for a time so there is no rise in turbine generator speed, whereas, in the generator trip, the main generator output breakers open

TWO

and residual steam will cause a momentary rise in turbine generator speed. This speed will be monitored during each test.

~~At test condition 3, a turbine trip will be initiated manually from the control room.~~ At test condition 6, a generator trip (load rejection) will be initiated by simulating a condition that will cause the generator output breakers to open. During ~~both~~ transients it is expected that the reactor will scram and the recirculation pump trip (RPT) breakers will open. It is not expected the HPCI or RCIC will initiate. Reactor water level, pressure, and heat flux will be monitored. The action of relief valves will be monitored.

THE

TURBINE

A ~~generator~~ trip will be performed at low power such that nuclear boiler system steam generation is just within bypass valve capacity. The purpose of this test is to demonstrate scram avoidance.

BOTH

During ~~all three~~ transients, main turbine stop, control, and bypass valve positions will be monitored. Prior to the low power, ~~generator~~ trip, bypass valve capacity will be determined.

TURBINE


d. Acceptance Criteria

Level 1:


1. For turbine and generator trips at power levels greater than 50%, the response times of the bypass valves shall be as specified in the GE startup test specification.
2. Feedwater control system settings must prevent flooding the main steam lines.
3. The reactor recirculation pump drive flow coastdown shall be as specified in the GE startup test specification.
4. The positive change in vessel dome pressure and heat flux must not exceed the limits specified in the GE startup test specification.

5. The total time delay from start of turbine stop valve motion or turbine control valve motion to complete suppression of electrical arc between the fully open contacts of the RPT circuit breakers shall be less than the limit specified in the GE startup test specification.

Level 2:

1. The bypass valve capacity shall be equal to or greater than that required by the GE startup test specification, which compares bypass valve capacity to the accident analysis.
2. There shall be no MSIV closure during the first three minutes of the transient and operator action shall not be required during that period to avoid the MSIV trip.
3. For the ~~generator~~  trip within bypass valves capacity, the reactor shall not scram for initial thermal power valves within that bypass valve capacity and below the power level at which trip scram is inhibited.
4. Low water level recirculation pump trip, HPCI and RCIC shall not be initiated.
5. Feedwater level control shall avoid loss of feedwater due to high level trip during the event.

TURBINE



1. To determine transient responses and steady-state conditions following recirculation pump trips at selected power levels
2. To obtain recirculation system performance data
3. To verify that cavitation in the recirculation system does not occur in the operating region of the power/flow map.

~~4. To verify the adequacy of the recirculation runback to mitigate a scram upon loss of one feedwater pump.~~

4^g. To verify that the feedwater control system can control water level without causing a turbine trip/scram following a single recirculation pump trip.

5^g. To demonstrate the adequacy of the recirculation pump restart procedure at the highest possible power level.

b. Prerequisites

The reactor is operating at steady-state conditions at required power level.

c. Test Method

Single pump trips are performed at test condition 3 and 6. Dual pump trip is demonstrated at test condition 3. The one-pump trip tests are to demonstrate that water level will not rise enough to threaten a high level trip of the main turbine or the feedwater pumps. The dual pump trip verifies the performance of the RPT circuit and the recirculation pump flow coastdown prior to the high power turbine generator trip tests. Single pump trips are initiated by tripping the pump motor breakers. Adequate margins to scrams and capability of the feedwater system to prevent a high level trip will be monitored. The two pump trip will be initiated by

simultaneously tripping both recirculation RPT breakers using a test switch. The recirculation pump restart demonstrates the adequacy of the restart operating procedure at the highest possible power level.

At several power and flow conditions, and in conjunction with single pump trip recoveries, recirculation system parameters are recorded.

~~At test condition 3 and at near rated recirculation flow, a loss of a feedwater pump is simulated. This is done prior to an actual feedwater pump trip to determine the adequacy of recirculation pump runback feature in preventing a scram.~~

While at test condition 3, it will be demonstrated that the cavitation interlocks which runback the recirculation pumps on decreased feedwater flow are adequate to prevent operation where recirculation pump or jet pump cavitation can occur.

d. Acceptance Criteria

Level 1:

1. During recovery from one pump-trip, the reactor shall not scram.
2. The two pump drive flow coastdown time constant following a dual recirculation pump trip is as specified in the GE startup test specification.

Level 2:

1. Neutron flux and heat flux scram, and reactor water high level trip avoidance margins are as specified in the GE startup test specification.
2. System performance parameters, including core flow, drive flow, jet pump M-ratio, core delta-pressure, recirculation pump efficiency and jet pump nozzle and riser plugging criteria are as specified in the GE startup test specification.

3. Runback logic shall have settings adequate to prevent operation in areas of potential cavitation.

~~4. The recirculation pumps shall runback upon a trip of the runback circuit.~~

14.2.12.3.29 Recirculation System Flow Calibration

a. Objective

The test objective is to perform a complete calibration of the installed recirculation system flow instrumentation, including specific signals to the plant process computer.

b. Prerequisites

The reactor is operating at steady-state conditions. The initial calibration of the recirculation system flow instrumentation has been completed.

c. Test Method

During the testing program at operating conditions required for rated flow at rated power, the jet pump flow instrumentation is adjusted to provide correct flow indication based on the jet pump flow. The flow-biased APRM/RBM system is adjusted to correctly follow core flow based on drive flow. Additionally, the total core flow and recirculation flow signals to the process computer will be calibrated to read these two process variables.

d. Acceptance Criteria

Level 2:

1. Jet pump flow instrumentation shall be adjusted such that the jet pump total flow ~~indication~~ recorder provides a core flow indication at rated conditions.

CORRECT

d. Acceptance Criteria

Level 1:

The piping displacements at the established locations shall not exceed the limits specified by the piping designer, which are based on not exceeding ASME Section III Code stress values or ANSI B31.1 values. These acceptable vibration levels will be used as acceptance criteria in the appropriate piping vibration startup test procedures.

14.2.12.3.32 Reactor Water Cleanup System

a. Objective

The test objective is to demonstrate the operation of the RWCU system.

b. Prerequisites

The reactor has been operated at a near rated temperature and pressure long enough to achieve a steady-state condition.

c. Test Method

With the reactor at rated temperature and pressure, process variables are recorded during steady-state operation in ~~three~~ two modes of operation of the RWCU system: blowdown, ~~hot standby,~~ and normal. The bottom head drain flow indicator will be calibrated by taking flow from the bottom drain only and using the RWCU system inlet flow indicator as a standard to compare against.

d. Acceptance Criteria

Level 2:

1. The performance data recorded during operation in the listed modes shall be acceptable as specified by the GE startup test specification.

~~2. Recalibrate bottom head flow indicator against RWCU flow indicator if the deviation is greater than values specified in the GE startup test specifications.~~

2. Pump vibration as measured on the bearing housing and coupling end shall be less than or equal to the values specified in the GE startup test specifications.

14.2.12.3.33 Residual Heat Removal System

a. Objectives

1. To demonstrate the ability of the RHR system to remove residual and decay heat from the nuclear system, so that refueling and nuclear system servicing can be performed
2. To demonstrate the capability of the RHR system to reduce the suppression pool temperature below the established limit immediately following a blowdown.

b. Prerequisites

Preoperational testing has been completed. ~~The test procedure has been reviewed, approved, and released for testing.~~ Instrumentation has been checked or calibrated as appropriate.

c. Test Method

Two modes are tested to verify system capability under actual operating conditions. The modes to be tested are suppression pool cooling and shutdown cooling. During the operations, the heat transfer rate is controlled to maintain acceptable cooldown rates. Data are recorded and reviewed to verify the satisfactory operation of the RHR system within design limits.

TEST NO. (22)	TEST NAME	OPEN VESSEL	HEAT UP	1	2	3	4	5	6
1	Chemical and Radiochemical	X	X	X		X		X	X
2	Radiation Measurement		X			X			X
3	Fuel Loading	X							
4	Full Core Shutdown Margin		X						
5	Control Rod Drive	X	X	X(2)	X(2)	X(2)			X(2)
6	SRM Performance	X							
8	IRM Performance		X	X					
9	LPRM Calibration		X	X		X			X
10	APRM Calibration		X	X	X	X		X	X
11	Process Computer	X	X	X(3)		X			X
12	RCIC		X	X					
13	HPCI		X			X			
14	Selected Process Temp		X			X	X(4)		X
14	Water Level Ref Leg Temp		X			X			X
15	System Expansion	X	X	X		X			X
16	TIP Uncertainty					X			X
17	Core Performance			X	X	X	X	X	X
18	Steam Production								X
19	Core Pwr-Void Mode Response						X	X	X
20	Pressure Regulator			X	X	X	X	X	X
21	Feed Sys-Setpoint Changes		X	X	X	X	X	X	X
21	Feed Sys-Loss FW Heating								X(5)
21	Feedwater Pump Trip								X(6)
21	Max FW Runout Capability								X(7)
22	Turbine Valve Surveillance							X(8)	X(10)
23	MSIV Functional Test		X	X(11)	X(12)			X(13)	
23	MSIV Full Isolation								X
24	Relief Valves			X(20)	X(15)	X(20)			X(20)
25	Turbine Trip & Load Rejection					X(16)			X(17)
26	Shutdown Outside CRC				X				
27	Recirculation Flow Control				X(14)			X(18)	
28	Recirc-One Pump Trip					X			X
28	RPT Trip-Two Pumps					X(19)			
28	Recirc System Performance				X	X	X		X
28	Recirc Pump Runback					X			
28	Recirc Sys Cavitation					X			
30	Loss of Offsite Pwr			X					
31	Pipe Vibration		X	X	X	X			X
29	Recirc Flow Calibration					X			X
32	RWCU			X(23)					
33	RHR				X(23)				X(21)
34	Drywell & Steam Tunnel Cooling		X	X		X			X
35	Gaseous Radwaste			X		X			X
38	SACS Performance					X			X
40	Confirmatory In-Plant Test				X				

6

- (1) Test conditions refer to plant conditions on Figure 14.2-4
- (2) Perform Test 5, timing of 4 selected control rods, in conjunction with expected scrams
- X
X (3) Dynamic System Test Case to be completed between test conditions 1 and 3
- X(2) (4) After recirculation pump trips (natural circulation)
- X
X
X (5) Between 80 and 90 percent thermal power, and near 100 percent core flow
- (6) Max FW Runout Capability & Recirc Pump Runback must have already been completed
- X
X
X
X (7) Reactor power between 80 and 90 percent
- X
X (8) Reactor power between 45 and 65 percent and 75 and 90 percent
- (9) Deleted
- X
X (10) At maximum power that will not cause scram
- X(5) (11) Perform between test conditions 1 and 3
- X(6) (12) Reactor power between 40 and 55 percent
- X(7) (13) Reactor power between 60 and 85 percent
- X(10) (14) Between test conditions 2 and 3
- X
X(20) (15) ~~Generator load rejection~~, within bypass valve capacity
- X(17) (16) ~~Reactor power between 60 and 80 percent at core flow > 95 percent - turbine trip~~
- X
X (17) Load rejection
- X
X (18) Between test conditions 5 and 6
- X(21) (19) >50% power and >95 core flow, and perform before Turbine Trip & Load Rejection
- X
X (20) Check SRV operability during major scram tests
- (21) Performed during cooldown from test condition 6
- (22) The test number correlates to FSAR Section 14.2.12.3 x where x is the indicated test number.
- (23) May be performed any time test conditions permit.

TURBINE TRIP

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GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

TEST SCHEDULE AND CONDITIONS

CHAPTER 15

ACCIDENT ANALYSES

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15.0.1 - 15.0.4 DELETED

CHAPTER 15

FIGURES

FigureTitle

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15.0-1

~~Typical Power/Flow Map~~

15.0-2

Scram Position and Reactivity Characteristics |

15.0-3

Minimum Operating CPR Limit |

15.1-1

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15.2-2

Generator Load Rejection Trip, Reactor Scram, Bypass-Off |

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Turbine Trip, Reactor Trip, Bypass-Off, RPT-On |

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15.2-11

Activity C2 Alternate Shutdown Cooling Path Utilizing RHR Loop A |

15.3-1

Trip of One Recirculation Pump |

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Chapter 15

ACCIDENT ANALYSES

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15.0 GENERAL

In this chapter, the effects of anticipated process disturbances and postulated component failures are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and events.

The scope of the situations analyzed includes anticipated (expected) operational occurrences, e.g., loss of electrical load; abnormal (unexpected) operational transients that induce system operations condition disturbances; postulated accidents of low probability, e.g., the sudden loss of integrity of a major component; and hypothetical events of extremely low probability, e.g., an anticipated transient without the operation of the entire control rod drive (CRD) system.

15.0.1 ANALYTICAL OBJECTIVE

The spectrum of postulated initiating events is divided into categories based upon the type of disturbance and the expected frequency of the initiating occurrence. The limiting events in each combination of category and frequency are quantitatively analyzed. The plant safety analysis evaluates the ability of the plant to operate within regulatory guidelines, without undue risk to the public health and safety.

15.0.2 ANALYTICAL CATEGORIES

Transient and accident events contained in this report are discussed in individual categories as required by Regulatory Guide 1.70, Revision 3. The results of the events are summarized in Table 15.0-1. Each event evaluated is assigned to one of the following applicable categories:

- a. Decrease in core coolant temperature - Reactor vessel water/moderator temperature reduction results in an increase in core reactivity. This could lead to fuel-cladding damage.

HCGS FSAR

- b. Increase in reactor pressure - Nuclear system pressure increases threaten to rupture the reactor coolant pressure boundary (RCPB). Increasing pressure also collapses the voids in the core moderator, thereby increasing core reactivity and power level, which threaten fuel cladding due to overheating.
- c. Decrease in reactor core coolant flow rate - A reduction in the core coolant flow rate threatens to overheat the cladding as the coolant becomes unable to adequately remove the heat generated by the fuel.
- d. Reactivity and power distribution anomalies - Transient events included in this category are those that cause rapid power increases due to increased core flow disturbance events. Increased core flow reduces the void content of the moderator increasing core reactivity and power level.
- e. Increase in reactor coolant inventory - Increasing coolant inventory could result in excessive moisture carryover to the main turbine, feedwater turbines, etc.
- f. Decrease in reactor coolant inventory - Reductions in coolant inventory could threaten the fuel as the coolant becomes less able to remove the heat generated in the core.
- g. Radioactive release from a subsystem or component - Loss of integrity of a radioactive containment component is postulated.
- h. Anticipated transients without scram (ATWS) - To determine the capability of plant design to accommodate an extremely low probability event, a multisystem maloperation situation is postulated.

15.0.3 TRANSIENT AND ACCIDENT EVENT EVALUATION

15.0.3.1 Identification of Causes and Frequency Classification

Situations and causes that lead to the analyzed initiating event are described within the categories designated earlier. The frequency of occurrence of each transient or accident is summarized, based upon currently available operating plant history for the event. Events for which inconclusive data exists are discussed separately within each event section.

Each initiating event within the major groups is assigned to one of the following frequency groups:

- a. Incidents of moderate frequency - These are incidents that can occur from once during a calendar year to once in 20 years for a particular plant. They are referred to as "anticipated (expected) operational transients."
- b. Infrequent incidents - These are incidents that can occur occasionally during the life of a particular plant, spanning once in 20 years to once in 100 years. They are referred to as "abnormal (unexpected) operational transients."
- c. Limiting faults - These are incidents that are not expected to occur but are postulated because their consequences can result in the release of significant amounts of radioactive material. They are referred to as "design basis accidents (DBAs)."
- d. Normal operation - Operations of high frequency are not discussed here but are examined along with a., b., and c. above in the nuclear systems operational analyses in Section 15.9.

15.0.3.1.1 Unacceptable Results for Incidents of Moderate Frequency-Anticipated (Expected) Operational Transients

The following are considered to be unacceptable safety results for incidents of moderate frequency, i.e., anticipated (expected) operational transients:

- a. A release of radioactive material to the environs that exceeds the limits of 10 CFR 20
- b. Reactor operation induced fuel cladding failure
- c. Nuclear system stresses in excess of that allowed for the transient classification by applicable industry codes
- d. Containment stresses in excess of that allowed for the transient classification by applicable industry codes.

15.0.3.1.2 Unacceptable Results for Infrequent Incidents-Abnormal (Unexpected) Operational Transients

The following are considered to be unacceptable safety results for infrequent incidents, i.e., abnormal operational transients:

- a. Release of radioactivity that results in dose consequences that exceed a small fraction of 10 CFR 100, i.e., no more than 10% of the 10 CFR 100 limits
- b. Fuel damage that precludes resumption of normal operation after a normal restart
- c. Generation of a condition that results in consequential loss of function of the reactor coolant system (RCS)
- d. Generation of a condition that results in a consequential loss of function of a necessary containment barrier

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- e. Nuclear system stresses in excess of that allowed for the transient classification by applicable industry codes
 - f. Containment stresses in excess of that allowed for the transient classification by applicable industry codes.

15.0.3.1.3 Unacceptable Results for Limiting Faults - Design Basis Accidents, (DBAs)

The following are considered to be unacceptable safety results for limiting faults (DBAs):

- a. Radioactive material release that results in dose consequences that exceed the guideline values of 10 CFR 100
- b. Failure of fuel cladding, which would cause changes in core geometry such that core cooling would be inhibited
- c. Nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes
- d. Containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required
- e. Radiation exposure to plant operations personnel in the main control room in excess of 5 rem whole body, 30 rem inhalation, and 75 rem skin.

15.0.3.2 Sequence of Events and Systems Operations

Each transient or accident is discussed and evaluated in terms of:

- a. The sequence of events from initiation to final stabilized condition

- b. The extent to which normally operating plant instrumentation and controls are assumed to function
- c. The extent to which plant and reactor protection systems (RPSs) are required to function
- d. The credit taken for the functioning of normally operating plant systems
- e. The operation of engineered safety systems that is required
- f. The effect of a single active failure (SAF) or a single operator error (SOE).

15.0.3.2.1 Single Failures or Operator Errors

This section discusses the application of SAF and SOE to analyses of the postulated events. Single active component failure (SACF) criteria have been required and successfully applied on past NRC-approved docket applications to DBA categories only. Regulatory Guide 1.70, Revision 3, infers that SAF and SOE requirements should be applied to transient events, including high, moderate, and low probability occurrences, as well as accident or very low probability situations.

Transient evaluations have been judged against criteria of one SACF or one SOE as the initiating event with no SAF assumptions added to the protective sequences although a great majority of these protective sequences use safety systems that can accommodate SACF aspects. Even under these postulated events, the plant damage allowances or limits are very much the same as those for normal operation.

The original categorization of events was based on frequency of the initiating event alone, and thus the allowance or limit was accordingly established based on that frequency level. If additional assumptions and conditions (initial event and single component failure (SCF) and/or SOE), were to be introduced, the events would be shifted to a lower frequency category. Less

restrictive limits or allowances would be applied to the results of the analyses of transients and accidents.

Most events postulated for consideration are already the results of single equipment failures or SOEs that have been postulated during any normal or planned mode of plant operations. The types of operational single failures and operator errors considered as initiating events and subsequent protective sequence challenges are identified in the following paragraphs.

15.0.3.2.1.1 Initiating Event Analysis

To initiate an event, one of the following actions must occur:

- a. The undesired opening or closing of any single valve (a check valve is not assumed to close against normal flow)
- b. The undesired starting or stopping of any single component
- c. The malfunction or maloperation of any single control device
- d. Any single electrical component failure
- e. The nonmechanistic break of a process piping line or process instrument line.
- f. Any SOE.

Operator error is defined as an active deviation from written operating procedures or nuclear plant standard operating practices. An SOE is the set of actions that is a direct consequence of a single erroneous decision. This set of actions is limited as follows:

- a. Those actions that could be performed by one person

- b. Those actions that would have constituted a correct procedure had the initial decision been correct
- c. Those actions that are subsequent to the initial operator error and have an effect on the designed operation of the plant, but are not necessarily directly related to the operator error.

Examples of SOEs are as follows:

- a. An increase in power above the established flow control power limits by control rod withdrawal in the specified sequences
- b. The selection and complete withdrawal of a single control rod out of sequence
- c. An incorrect calibration of an average power range monitor (APRM)
- d. Manual isolation of the main steam lines as a result of operator misinterpretation of an alarm or indication.

15.0.3.2.1.2 Single Active Component Failure or Single Operator Error Analysis

These failures include:

- a. The undesired action or maloperation of a single active component
- b. Any SOE where operator errors are as defined in Section 15.0.3.2.1.1.

15.0.3.3 Core and System Performance

Section 4.4 describes the various fuel failure mechanisms. Avoidance of unacceptable results a. and b. in Section 4.4.1.4 for incidents of moderate frequency is verified statistically with consideration given to date, calculation, manufacturing, and operating uncertainties. An acceptable criterion is that 99.9% of the fuel rods in the core are not expected to experience boiling transition. For more detail, see Reference 15.0-1. This criterion is met by demonstrating that incidents of moderate frequency do not result in a minimum critical power ratio (MCPR) less than 1.06. The reactor steady-state critical power ratio (CPR) operating limit is derived by determining the decrease in MCPR for the most limiting event. All other events result in smaller MCPR decreases and are not reviewed in depth in this chapter. The MCPR during significant abnormal events is calculated using a transient core heat transfer analysis computer program. The computer program is based on a multinode, single-channel, thermal-hydraulic model that requires simultaneous solution of the partial differential equations for the conservation of mass, energy, and momentum in the bundle, and that accounts for axial variation in power generation. The primary inputs to the model include a physical description of the bundle, and channel inlet flow and enthalpy, pressure, and power generation as functions of time.

A detailed description of the analytical model may be found in Appendix C of Reference 15.0-1. Maintaining MCPR greater than the safety MCPR limit is a sufficient, but not necessary, condition to ensure that no fuel damage occurs. This is discussed further in Section 4.4.

For situations in which fuel damage is sustained, the extent of damage is determined by correlating fuel energy content, cladding temperature, fuel rod internal pressure, and cladding mechanical characteristics.

These correlations are substantiated by fuel rod failure tests and are discussed in Sections 4.4 and 6.3.

15.0.3.3.1 Input Parameters and Initial Conditions for Analyzed Events

In general, the events analyzed within this section have values for input parameters and initial conditions as specified in

Table 15.0-3. Analyses that assume data inputs different from these values are designated accordingly in the appropriate event discussion in this chapter. These transient analyses presuppose, for the end-of-cycle-one (EOC-1) conditions being simulated, that the plant design includes a recirculation pump trip (RPT) actuated by either fast closure of the turbine control valves or closure of the main stop valve. An EOC-1/RPT system is part of the current plant design.

15.0.3.3.2 Initial Power/Flow Operating Constraints

The basis for most of the transient safety analyses is the thermal power at the 100% rated core flow corresponding to 105% nuclear-boiler-rated (NBR) steam flow. This operating point is the apex of a bounded operating power/flow map that in response to any classified abnormal operational transients, yields the minimum pressure and thermal margins of any operating point within the bounded map. As shown on Figure 15.0-1, the apex of the bounded power/flow map is point A; the upper boundary is the design flow control line, which is 104.3% rod line A-D'; the lower boundary is the zero power line H'-J'; the right boundary is the rated pump speed line A-H'; and the left boundary is either the minimum pump speed line D-J or the natural circulation line D'-J'.

This power/flow map, A-D'-J'-H'-A, represents the acceptable operational constraints for abnormal operational transient evaluations.

Any other constraint that can truncate the bounded power/flow map, e.g., the recirculation valve and pump cavitation regions, the licensed power limit, and other restrictions based on pressure and thermal margin criteria, must be observed. For instance, if the licensed power is 100% NBR, the power/flow map is truncated by the line B-C and reactor operation must be confined within the boundary B-C-D'-J'-J-L-K-B. If the maximum operating power level, e.g., point F, has to be limited to satisfy pressure margin criteria, the upper constraint on power/flow is correspondingly reduced to the rod line, e.g., line F-G', which intersects the power/flow coordinate of the new operating basis. In this case, the operating boundaries are F-G'-J'-J-L-K-F. Operation is not allowed at any point along line F-M, removed from point F, at the derated power but at reduced flow. If, however, operating limitations are imposed by General Electric Boiling Water Reactor Thermal Analysis Basis (GETAB), Reference 15.0-1, derived from transient data with an operating basis at point A, the power/flow boundary for 100% NBR

licensed power is B-C-D'-J'-J-L-K-B. This power/flow boundary is truncated by the MCPR operating limit, for which there is no direct correlation to a line on the power/flow map. Operation is allowed within the defined power/flow boundary and within the constraints imposed by GETAB. If operation is restricted to point F by the MCPR operating limit, operation at point M is allowed, provided that the MCPR limit is not violated.

Consequently, the upper operating power/flow limit of a reactor is predicated on the operating basis of the analysis and the corresponding constant rod pattern line. This boundary can be truncated by the licensed power and the GETAB operating limit.

Certain localized events are evaluated at other than the above-mentioned conditions. These conditions are included in the appropriate event discussion in this chapter.

15.0.3.3.3 Results

The results of analytical evaluations are provided for each event. In addition, critical parameters are shown in Table 15.0-1. From the data in Table 15.0-1, an evaluation can be made of the limiting event for that particular category and parameter. Table 15.0-2 provides a summary of applicable accident analysis results.

15.0.3.3.4 Regulatory Guide 1.49; General Compliance or Alternate Approach Assessment

For commitment, revision number, and scope, see Section 1.10.

Regulatory Guide 1.49 requires that the proposed licensed power level be restricted to a reactor core power level of 3800 megawatts thermal or less, and that analyses and evaluations in support of the application should be made at 1.02 times the proposed licensed power level. The compliance is shown in Table 15.0-3.

15.0.3.4 Barrier Performance

During transients that occur with no release of coolant to the containment, only RCPB performance is considered. If release to

the containment occurs as in the case of limiting faults, then challenges to the containment are evaluated as well.

Containment integrity is maintained as long as internal pressures remain below the maximum allowable values. The design internal pressures are as follows:

- a. Drywell (primary containment), 58 psig
- b. Suppression chamber (primary containment), 58 psig
- c. Reactor building enclosure, 6.5 inches of water.

Damage to any of the radioactive material barriers as a result of accident-initiated fluid impingement and jet forces is considered in other sections of the FSAR where mechanical design features of the systems and components are described. DBAs are used to determine the size and strength requirements of the essential nuclear system components. A comparison of the accidents considered in this section with those used in the mechanical design of equipment reveals either no difference or less severe stresses than those assumed for mechanical design.

15.0.3.5 Radiological Consequences

Consequences of radioactivity release during the following three types of events are considered:

- a. Incidents of moderate frequency (anticipated operational transients)
- b. Infrequent incidents (abnormal operational transients)
- c. Limiting faults (DBAs).

For all events whose consequences are limiting, a detailed quantitative evaluation is presented. For nonlimiting events, a qualitative evaluation is presented, or the results are referenced from a more limiting or enveloping case.

For limiting faults (DBAs), two quantitative analyses are considered:

- a. The first is based on conservative assumptions considered acceptable to the NRC for the purposes of bounding the worst-case event and determining the adequacy of the plant design to meet 10 CFR 100 guidelines. This analysis is referred to as the "design basis analysis."
- b. The second is based on realistic assumptions considered to reflect expected radiological consequences. This analysis is referred to as the "realistic analysis."

Results for both are shown to be within NRC guidelines.

15.0.4 NUCLEAR SAFETY OPERATIONAL ANALYSIS (NSOA) RELATIONSHIP

Section 15.9 is a comprehensive, total-plant, system-level, qualitative failure mode and effects analysis (FMEA), relative to all the Chapter 15 events considered, the protective sequences used to accommodate the events and their effects, and the systems involved in the protective actions.

Interdependency of analysis and cross-referral of protective actions is an integral part of this chapter and appendix.

Contained in Section 15.9 is a summary table that classifies events by frequency only, i.e., not just within a given category such as "decrease in core coolant temperature."

15.0.5 REFERENCES

15.0-1

General Electric, General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application, NEDO-10958 and NEDE-10958, November 1973.

15.0-2

General Electric, Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154, October 1978.

DELETED

DELETED

15.0-3

~~R. B. Linford, Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor, NEDO-10802, General Electric, April 1973.~~

15.0-4

"GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL," INCLUDING THE "UNITED STATES SUPPLEMENT," NEDE-24011-P-A-7 AND NEDE-24011-P-A-7-US.

36. Pressure setpoint of RPT, psig

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(1) ~~Applicable to events analyzed using model described in Reference 15.0.3.~~

(2) The inertia time constant is defined by the expression:

$$t = \frac{2\pi J_0 n}{g T_0}$$

where:

- t = inertia time constant, s
J₀ = pump motor moment of inertia, lb-ft²
n = rated pump speed, rps
g = gravitational constant, ft/s²
T₀ = pump shaft torque, ft-lb.

FOR TRANSIENTS SIMULATED ON THE ODYN COMPUTER MODEL, THIS INPUT IS CALCULATED BY ODYN

HCGS FSAR

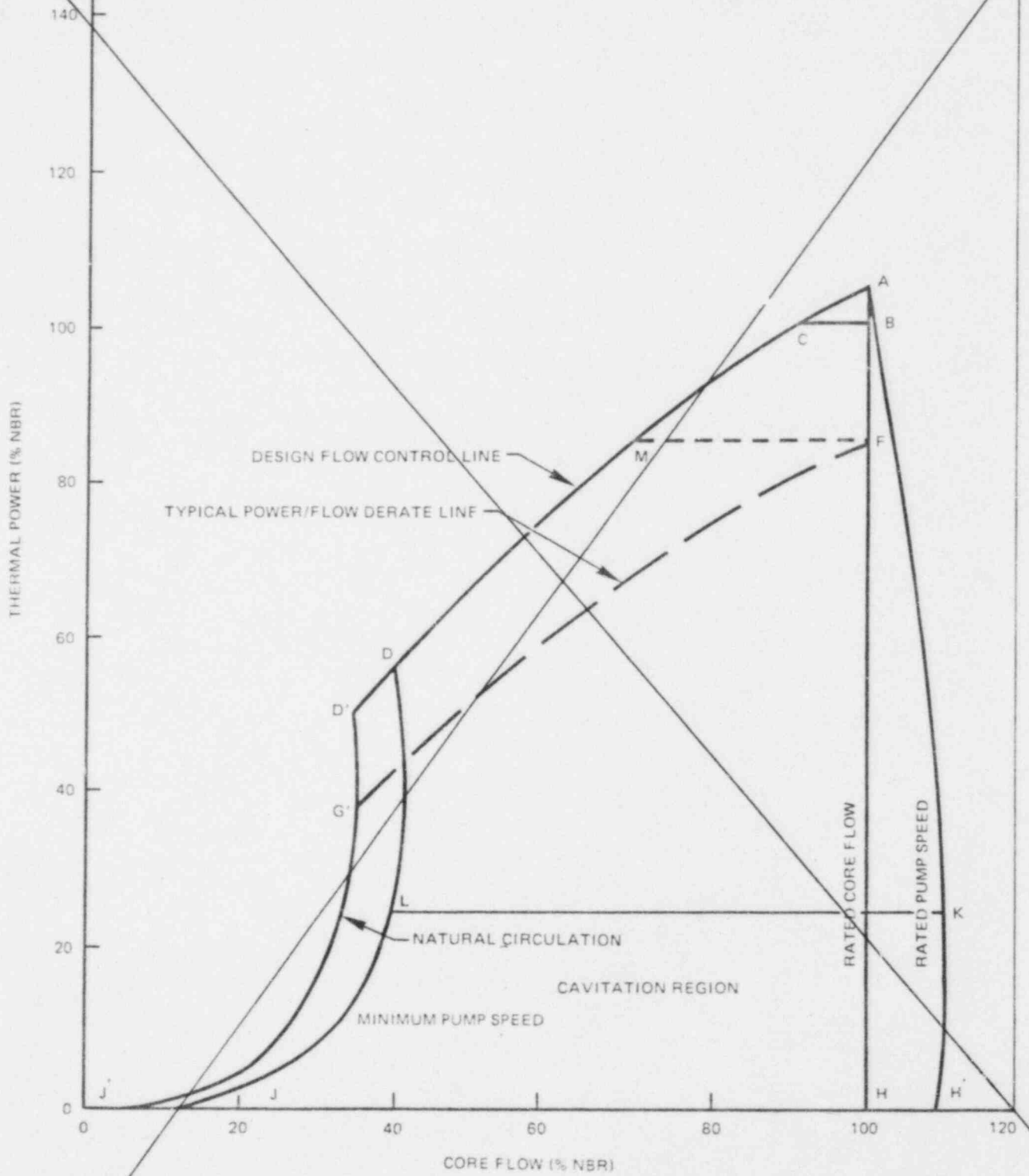
TABLE 15.0-4

Required Operating Limit MCPR Values

<u>Pressurization Events</u>	<u>OLMCPR (Option A)(1)</u>	<u>OLMCPR (Option B)(1)</u>
Load rejection with bypass	1.14	1.06
Load rejection without bypass	1.18	1.10
Turbine trip with bypass	1.12	1.06
Turbine trip without bypass	1.17	1.09
Feedwater controller failure with bypass	1.20	1.17
 <u>Nonpressurization Events</u>	 <u>OLMCPR</u>	
Loss of feedwater heating, MFC	1.20	
Rod withdrawal error (RBM=106%)	1.20	

(1) Option A and B include adjustment factors as specified in Reference 15.0.2

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~~TYPICAL POWER/FLOW MAP~~
 DELETED

FIGURE 15.0-1

15.3.4.5 Radiological Consequences

The radiological consequences of this event are the same as discussed in Section 15.3.1.5.

15.3.4.6 SRP Rule Review

SRP 15.3.3 - 15.3.4 acceptance criterion II.10 states that analysis for reactor coolant pump rotor seizure and reactor coolant pump shaft break events should include assumptions of turbine trip and coincidental loss of offsite power (LOP) and coastdown of undamaged pumps.

Coincidental LOP and turbine trip are not assumed in the HCGS analysis but would, if included, produce consequences less severe than those of Section 15.2-6.

The turbine trip or, indirectly, the loss of offsite power, will initiate reactor scram and rapid power reduction. The severity of pump shaft seizure or pump shaft break without assuming LOP is evidenced by the fast coastdown of core flow, which reduces the thermal margin significantly before a reactor scram is initiated by an L8 signal.

15.3.5 REFERENCES

- 15.3-1 General Electric, General Electric Standard Application for Reactor Fuel, including the United States Supplement, NEDE-24011-P-A and NEDE-24011-P-A, US. (latest approved revision)

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15.4.1.2.4 Barrier Performance

An evaluation of the barrier performance is not made for this event since there is no postulated set of circumstances for which this error could occur.

15.4.1.2.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

15.4.2 ROD WITHDRAWAL ERROR - AT POWER

This event is described in Section 5.2.1.5 of Reference 15.4-3. Analysis specific to the first cycle was performed. The limiting rod pattern is shown on Figure 15.4-1, and a summary of the results is presented in Table 15.4-22. For the selected rod block monitor setpoint, the change in the critical power ratio (Δ CPR) is 0.132 and the maximum linear heat generation rate (MLHGR) is 15.4 kw/ft.

15.4.3 CONTROL ROD MALOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

This event is covered by the evaluation cited in Sections 15.4.1 and 15.4.2.

15.4.4- ABNORMAL STARTUP OF IDLE RECIRCULATION PUMP

15.4.4.1 Identification of Causes and Frequency Classification

15.4.4.1.1 Identification of Causes

Abnormal startup of an idle recirculation pump results directly from the operator's manual initiation of pump operation. It assumes that the remaining loop is already operating.

15.4.10 REFERENCES

- 15.4-1 C. J. Paone, Bank Position Withdrawal Sequence, NEDO-21231, September 1976.
- 15.4-2 Deleted
- 15.4-3 -7 _____ General Electric, General Electric Standard Application For Reactor Fuel, including the United States Supplement, NEDE-24011-P-A, and NEDE-24011-P-A-US. (Latest Approved Revision).
- 15.4-4 Stancavage, P.P. and Morgan, E.J., Conservative Radiological Accident Evaluation - The CONACOI Code, NEDO-21143, March 1976.
- 15.4-5 Nguyen, D., Realistic Accident Analysis - The RELAC Code, NEDO-21142, October 1977. General Electric, General Electric Standard Application For Reactor Fuel, including the United States Supplement NEDE-24011-P-A and NEDE-24011-P-A-US, (latest approved revision).
- 15.4-6 N.R. Horton, W.A. Williams, K.W. Holtzclaw, Analytical Methods for Evaluating the Radiological Aspects of General Electric Boiling Water Reactors, APED-5756, March 1969.
- 15.4-7 Nuclear Regulatory Commission, Standard Review Plan, NUREG-75/037, Washington, D.C., November 24, 1975.

- v. Regulatory Guide 1.146, Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants
- w. BTP 9.5-1, Appendix A, Guidelines for Fire Protection for Nuclear Plants Docketed Prior to July 1, 1976.

Commitments to Regulatory Guides, with respect to revision level, exceptions, etc, are contained in Section 1.8.

Substantive changes to the QA program described herein will be submitted to the NRC within 30 days of implementation. Nonsubstantive changes will be identified in the annual FSAR updates.

The overall QA program is described in the Nuclear Department manual. This description is prepared and maintained by NQA.

PSE&G organizations performing activities affecting nuclear safety prepare and maintain implementing procedures and instructions. These procedures and instructions, and subsequent revisions thereto, are subject to NQA review and approval to the extent necessary to verify compliance with the QA Program and the applicable quality-related Regulatory Guides and standards identified above. NQA will monitor the preparation and issuance of required procedures to assure that they are in place prior to implementation of activities needed to support systems turnover to station operations and that all required procedures are in place ~~at least 90 days~~ prior to fuel load.

The General Manager - Hope Creek Operations has instituted and will maintain an administrative procedures (AP) manual for Hope Creek Generating Station (HCGS).

The station APs and all subsequent revisions thereto are prepared by the technical staff, are reviewed by the Technical Engineer, Technical Manager, NQA and SORC, and are approved by the General Manager - Hope Creek Operations.

Regulatory Guide 1.33 requires that plant activities affecting quality-related items and services be conducted in accordance with written administrative controls prepared by management. The procedures and instructions by which plant activities are

performed are prepared by the responsible station organization as required by station APs, reviewed by the organization responsible for the activity, reviewed by NQA for quality requirements, reviewed by the SORC (for procedures affecting safety), and approved by the department manager. In the absence of a department manager, procedures will be approved by the assistant general manager or his designee. Procedures cannot be implemented unless the review/approval process is accomplished. Station APs provide a means to accommodate on-the-spot changes to subtier implementing procedures. The routine practice for revising a procedure is to repeat the original review and approval sequence.

Implementation of the QA program is verified by means of independent inspections, monitoring, and audits conducted by NQA.

NQA reviews and analyzes problems affecting safety that occur during the operational phase. Items subject to review include:

- a. Documented nonconformances occurring at the vendor's facility and those during receiving, storage, installation, test, and operation, e.g., Deficiency Reports, Nonconformance Reports, Licensee Event Reports, etc
- b. Documented corrective actions taken on significant noncompliances and on audit findings
- c. NRC inspection findings, notifications, bulletins, etc.

The General Manager - Nuclear Quality Assurance, or his designee, has the authority to stop work through the issuance of a stop work order where continuance of an activity would seriously compromise safety or constitute a persistent and deliberate failure to correct a serious deficiency. Designees include the Manager - Station Quality Assurance for activities conducted at the station and the Manager - QA Engineering and Procurement for supplier activities.

NQA reports significant problems affecting the quality assurance program to respective management along with:

- a. Measures taken to improve QA program controls

THE QA PROGRAM SHALL BE IMPLEMENTED
90 DAYS PRIOR TO FUEL LOAD.

QUESTION 410.26 (SECTION 4.6)

Provide the information requested in our generic letter 81-34, dated August 31, 1981, regarding NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping."

RESPONSE

HCGS is participating in the BWROG activities related to the scram discharge pipe integrity. The BWROG's final response to the NRC ~~is being prepared for NRC review and approval. A HCGS plant specific response will be provided within 60 days of NRC resolution of the BWROG position. HCGS will implement any required fix by the end of the next refueling outage which is at least 12 months after NRC resolution. Pending material availability, this schedule may change with NRC approval.~~

HAS BEEN PROVIDED WHILE
THE HCGS PLANT SPECIFIC
RESPONSE IS SHOWN IN
SECTION 4.6.3.1.5.f.