



June 11, 1996
LD-96-021

Docket 52-002

Attention: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Subject: Changes to the System 80+™ Standard Plant Design

Reference: Letter, B. K. Grimes to C. B. Brinkman, "Preparation of Final Design Control Document for the System 80+ Standard Plant Design," 5/15/96

Dear Sirs:

The purpose of this letter is to submit the attached draft changes to the System 80+ Design Control Document (DCD), as requested in the Reference letter, for Staff review and approval. These design changes are itemized on the attached table with details shown on the attached marked-up DCD pages; these changes were discussed with the Staff on June 6, 1996.

Approved changes will be documented by printed replacement DCD pages containing a margin bar adjacent to each change and a revision date of 6/96. All such margin bars and revision dates will be deleted from the DCD when it is reprinted following design certification rulemaking.

Please call me, Stan Ritterbusch (860-285-5206) or Virgil Paggen (860-285-4700) if you have any questions.

Very truly yours,
COMBUSTION ENGINEERING, INC.

C. B. Brinkman
Director, Nuclear Licensing

cc: J. N.. Wilson (NRC)
S. M. Franks w/o enclosure (DOE)

Attachments: As stated

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ABB Combustion Engineering Nuclear Systems

Potential System 80+ Design Changes

Item No.	Ref DCD Section	Design Change Description
1	Tbl 1.8-7, Tbl in 3.9; Ch 5, 16 Ch 6	ASME Section XI Code has been revised to eliminate Inservice hydrostatic testing. Code now requires shop test and initial in-plant test at 125% of design pressure, while remaining in-plant leak-rate tests would be conducted at 100% of design pressure. (Change is approved by NRC). Code Cases N-453, N-498 address issue. No impact on ITAAC or FSER. Action; modify P-T curves to remove "Inservice test" line, add Code Case N-498.
2	Sect 6.5.3.4, Figures 6.3.2-1A & 6.3.2-1B.	Insufficient net positive suction head is available to the S80+ shutdown cooling pumps when aligned for containment spray due to frictional loss in crossover piping. Increased SCS piping & valve sizes; reduce allowable containment spray runout flow to provide reqd NPSH when used for cmtt spray function.
3	Tbl 3.2-1	Correct the seismic classification, quality class of the vacuum breakers, the piping from the PSVs to the IRWST, and spargers; correct DG heading (pg 3.2-13), correct potable water heading (pg 3.2-18).
4	Ch 16, Fig 3.4.5-1	Evaluate need to revise Tech Spec on IRWST temperature. Fig 3.5.4-1 indicates IRWST limit is 40F; analysis limit is 60F. Confirm whether Tech Spec Figure in Ch 16 should be updated to 60F.
5	Ch 4; Figs in Ch 3, 5 ITAAC	Twelve (12) CEA locations in core center region should be added and designated as "spares." These would allow additional maneuvering capability. Requires correction of CEA locations shown in ITAAC Fig 2.2.1-3, ADM Fig 4.2-11, ADM Fig 4.3-46, ADM Fig 4.3-47. Also modify Section 4.2.2.4 to accommodate extra CEAs.
6	Ch 5	Revise mid-loop level monitor to accommodate ex-vessel device.
7	Ch 6, Ch 3.6	Reduce SDS line/valve size from 6" to 4" per YGN qualification data. Revise Figure 5.1.2-3.
8	Ch 3	Modify Piping Evaluation Diagrams to be consistent with NRC-approved criteria for AP600. Change load combination factor from 1.4 to 1.0.
9	Pg 9.3-29	Section 9.3.4.2.1 states interlock provided so only one chg pump can be operated during all modes of plant operation. [Ch 15 limits chg flow to 150 gpm for boron dilution; also Tier 1 commitment.] This means that one charging pump must be stopped before the standby pump can be started. Momentary loss of RCP seal injection may occur.
10	Ch 3	Increase damping for envelope response spectrum analysis of piping to 5% to be consistent with NRC-approved criteria for AP-600.
11	Tbl 3.6-3	Delete PSV relief lines, items 40...43 since valves mounted directly on pzz; correct items 58, 59.
12	Tbl 3.9-2	Correct Table title to include component supports in loading combinations.
13	Tbl 4.2-3	Add ANO inspection program scope for year 1989/cycle 7.
14	Tbl 5.4.7-2	Correct failure mode entry to "fail closed" for shutdown cooling pump dischg isolation valve.
15	Pg 9.3-30	Correct statement regarding fluid return to RCS when in shutdown cooling mode.
16	Pg 9.3-37	Revise CVCS system "redundancy" statements to be consistent.
17	Tbl 9.3.4-4	Change VCT normal operating pressure to be 20-50 psig.
18	Pg 10.3-7, 13, 14	Revise MSIV bypass valve closing time to be 5 seconds or less, rather than 10 seconds. 5 seconds was used in safety analysis.
19	Ch 19.7.5	Delete Tables 19.7.5.1-1 through 19.7.5.4-7. These tables should be intentionally blank.
20	Pg 19.11-145	Revise statement re: core uncover at 7700 seconds to be consistent with referenced Fig & Table.

System 80+ DCD - Potential Design Changes

Item Number: 1

Summary Description: Delete Inservice RCS Hydrostatic Testing

Affected DCD Section(s): CDM: None
ADM: Table 3.9-1
Chapters 5 and 6

Description of Change:

A recent ASME B&PV Code Case has been published which provides an alternative to performing the 10-year hydrostatic pressure test. Code Case N-498, "Alternate Rules for 10 Year Hydrostatic Pressure Testing for Class 1 and 2 Systems Section XI, Division 1", indicates that a system leakage test can be conducted at or near the end of each inspection period, prior to reactor startup instead of the Hydrostatic Pressure Test. This code case was approved by the NRC as indicated in Reg Guide 1.147, Rev 09. Currently the DCD (Table 3.9-1) specifies that 15 RCS and secondary hydrostatic tests are included in the stress analysis of Code Class 1 and CS components. This number will be reduced from 15 to 10 occurrences during the plant life time.

The RCS and secondary leak tests have adequate margin in the number of occurrences listed in Table 3.9-1 to account for the additional leak tests which will be performed instead of the hydrostatic pressure tests. Table 1.8-7, "ASME Section III Code Cases applicable to System 80+ will be revised to include Code Case N-498. In addition, Section 5.2.4.6 of the DCD will be revised to include reference to the Code Case.

DCD Markups Attached? Yes

Table 3.9-1 Transients Used in Stress Analysis of Code Class 1 and CS Components
(Cont'd.)

Test Conditions ^[1]		Occurrences ^[2]
1. RCS hydrostatic test [Primary pressure cycles from atmospheric to 3125 psia at a temperature between 120 and 400°F]		16. 10.
2. RCS leak test [Primary pressure cycles from atmospheric to 2250 psia at a temperature between 120 and 400°F]		200.
3. Secondary hydrostatic test [Secondary pressure cycles from atmospheric to 1500 psia at a maximum temperature of 190°F]		16. 10.
4. Secondary leak test [Secondary pressure cycles from atmospheric to 1200 psia at a maximum temperature of 200°F]		200.
5. SIS/SCS check valve operability test		500.
6. SIS/SCS preoperational and maintenance test		240.
Upset Conditions ^[1]		Occurrences ^[2]
1. Decrease in feedwater temperature		20.
2. Increase in feedwater flow rate		20.
3. Increase in steam flow rate		20.
4. Inadvertent opening of a steam generator relief or safety valve		10.
5. Loss of load (turbine speed control system operates normally) [Loss of electrical load and normal turbine/generator runback to house load]		19.
6. Turbine trip		20.
7. Loss of condenser vacuum		20.
8. Loss of non-emergency AC power to the station auxiliaries		10.
9. Loss of normal feedwater flow [Subsequent actuation and cycling of cold emergency feedwater to the steam generators]		20.
10. Loss of forced reactor coolant flow		20.
11. Uncontrolled CEA withdrawal from subcritical or low power condition		10.

Table 1.8-7 ASME Section III Code Cases Applicable to System 80+

Case	Title
N-4-11	[1337-11] Special Type 403 Modified Forgings or Bars, Class 1 and CS; 7/13/87.
N-60-4	Material for Core Support Structures; 7/27/88.
N-71-15	Additional Materials for Subsection NF, Classes 1,2,3 and MC Component Supports Fabricated by Welding; 12/16/89.
N-122-1	Evaluation of the Design of Rectangular Cross-Section Attachments on Class-1 Piping; 7/27/92.
N-192-2	Use of Braided Flexible Connectors, Class 2 and 3; 9/17/87.
N-247	Certified Design Report Summary for Component Standard Support, Class 1,2,3, and MC; 1/21/88.
N-249-10	Additional Materials for Subsection NF, Classes 1,2,3 and MC Component Supports Fabricated without Welding; 5/06/89.
N-262	Resistance Spot Welding for Structural Use in Component Supports; 7/28/88.
N-284	Metal Containment Shell Buckling Design Methods; Section III, Division 1, Class MC; 8/25/80.
N-309-1	Identification of Material for Component Supports; 7/28/88.
N-313	Alternate Rules for half-Coupling Branch Connections, Class 2; 11/28/86.
N-318-4	Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping; 12/11/89.
N-319-1	Evaluation of Stresses in Butt Welded Elbows for Class 1 Piping; 7/24/89.
N-391-1	Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping; 7/24/89.
N-392-1	Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 2 and 3 Pippings; 12/11/89.
N-393	Repair Welding Structural Steel Rolled Shapes and Plates for Components Supports; 7/30/89.
N-411-1	Alternative Damping Values for Response Spectra Analysis for Class 1,2, and 3 Piping; 2/20/89.
N-420	Linear Energy Absorbing Supports for Subsection NF, Class 1, 2, and 3 Construction; 2/14/88
N-430	Alternative Requirements for Welding Workmanship and Visual Acceptance Criteria for Class 1,2, 3, and MC Linear-type and Standard Supports; 2/28/89.
N-433	Non-threaded Fasteners for Class 1, 2, and 3 Components Piping Supports; 12/16/89.
N-474-1	Design Stress Intensities and Yield Strength Values for UNS NO6690 with a minimum specific Yield Strength of 35 ksi, Class 1 Components; 3/05/90.
N-476	Class 1, 2, 3, and MC Linear Component Supports - Design Criteria for Single Angle Members, Subsection NF; 5/06/89.

N-498 Alternative Rules for 10-year Hydrostatic Pressure Testing for Class 1 and 2 Systems SECTION XI, Division 1

5.2.4.4 Inspection Intervals

The IWA-2000 examination program for the 120 month inspection interval will be defined in the ISI plan. The ISI Plan for all Code Class 1 systems and components will be in accordance with the ASME Code Section XI edition in effect per 10 CFR 50.55a 12 months prior to the issuance of an operating license (initial interval).

5.2.4.5 Evaluation of Examination Results

Evaluation of examination results for Class 1 components will be conducted in accordance with Articles IWA-3000 and IWB-3000 of ASME Section XI.

Unacceptable indications will be repaired in accordance with the requirements of Articles IWA-4000 and IWB-4000 of ASME Section XI. Criteria for establishing need for repair or replacement shall be per IWB-3000.

5.2.4.6 System Leakage and Hydrostatic Tests

The hydrostatic and system leak tests for the reactor pressure vessel and reactor coolant pressure boundary will be conducted in accordance with the requirements of Articles IWA-5000 and IWB-5000 of ASME Section XI. Examinations performed during these tests will be conducted without the removal of insulation. Technical Specifications requirements on operating limits during heatup, cooldown, and system hydrostatic pressure testing shall be employed for these tests.

5.2.4.7 Code Exemptions

As provided in ASME Section XI, IWB-1220, certain portions of Class 1 systems are exempt from the volumetric and surface examination requirements of IWB-2500. The following components (or parts of components) are exempt from the volumetric and surface examination requirements of IWB-2500:

1. Components that are connected to the reactor coolant system and part of the reactor coolant pressure boundary, and that are of such a size and shape so that upon postulated rupture the resulting flow of coolant from the reactor coolant system under normal plant operating conditions is within the capacity of makeup systems which are operable from on-site emergency power;
 - piping of 1 inch nominal pipe size and smaller, except for steam generator tubing;
 - components and their connections in piping of 1 inch nominal pipe size and smaller;
2. reactor vessel head connections and associated piping, 2 inch nominal pipe size and smaller, made inaccessible by control rod drive penetrations.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection Systems

Means for the detection of leakage from the Reactor Coolant Pressure Boundary are provided to alert operators to the existence of leakage above acceptable limits, which may indicate an unsafe condition for the facility. The leakage detection systems are sufficiently diverse and sensitive to meet the criteria of Regulatory Guide 1.45 for leaks from identified and unidentified sources. The leakage detection systems are capable of performing their functions following seismic events that do not require plant shutdown.

as modified by ASME Code Case
N-498 (Alternate rules for 10-year Hydrostatic
Pressure Testing)

System 80+ DCD - Potential Design Changes

Item Number: 2

Summary Description: NPSH for SCS and CSS Pumps

Affected DCD Sections: CDM: None
ADM: Section 6.5.3.4
Figures 6.3.2-1A and 6.3.2-1B

Description of Change:

Reason for Change: There are two root causes for this design change:

1. While bidding System 80+ for the Lungmen Project in Taiwan, subsequent to the issuance of the Final Design Approval, ABB-CE received data from prospective pump vendors KSB and Ingersoll-Dresser. The data showed that the net positive suction head required (NPSHR) by the proposed shutdown cooling (SC) and containment spray (CS) pumps exceeded the NPSHR assumed in preparation of DCD Section 6.5.3.4. We had assumed a maximum NPSHR of 20 feet at pump runout flow of 6500 gpm. The vendors require approximately 25 feet at 6500 gpm.
2. During the engineering work to support CESSAR-DC, ABB-CE did not evaluate the net positive suction head (NPSH) available to the SC pumps when aligned for containment spray.

As a result, two design concerns were identified:

1. There may be insufficient NPSH to the containment spray pumps if they operate at the maximum flow rate of 6500 gpm stated in the DCD.
2. There may be insufficient NPSH to the SC pumps when aligned for containment spray, due to large frictional losses caused by the length and diameter of the suction crossover piping.

[The NPSH available to the SC and CS pumps during all other operating modes is adequate.]

Description: The NPSH during containment spray operation was evaluated and some SCS/CSS suction lines were re-sized so that the minimum available NPSH exceeds the NPSHR specified by the pump vendors. Based on the revised NPSH calculations, and as shown on the attached markup of DCD Figures 6.3.2-1A and 1B, the changes are:

- Increased the nominal diameter of CS suction piping, including valves SI-104, 105, 157, and 158, from 18 to 20 inches
- Increased the nominal diameter of CS/SC pump suction crossover piping, including valves SI-340 and 342, from 18 to 20 inches
- Increased the nominal diameter of SC suction piping, including valves SI-107 and 106, from 14 (or 18) to 20 inches

Despite increasing these line sizes, it was not possible to obtain enough NPSHA for the CS or SC pumps during containment spray operation above 6000 gpm. Therefore, the maximum allowable

System 80+ DCD - Potential Design Changes

containment spray flow rate was reduced from 6500 gpm to 5500 gpm. Safety and containment analyses use CS flow rates of 5000 gpm minimum, 6500 gpm maximum, so this change provides margin to the maximum assumed value.

During the performance of these analyses, it was also determined that the IRWST water volume presented in DCD Section 6.5.3.4 was the volume above the pump suction piping. The volume will be represented as to the total volume remaining in the IRWST (198,000 gallons), to be consistent with the measurement in Technical Specification 3.5.4. **The minimum IRWST level (75.5 feet) is unchanged.**

The following table compares original and revised NPSH values for containment spray operation:

Flow Rate (gpm)	NPSH Required (Feet)		Minimum NPSH Available (Feet)			
	Assumed in DCD	Vendor Data	CS Pump Original	CS Pump Revised	SC Pump Original	SC Pump Revised
5000	none	16.5	24	21.9	n/c	20.7
5500	none	18	n/c	21.1	n/c	19.6
6000	none	20	n/c	20.3	n/c	18.5
6500	≤ 20	25	21.2	19.4	n/c	17.4

There are no design changes to the pumps themselves. The SC and CS pumps are still identical and interchangeable. The minimum pump design head and design flow will be confirmed by ITAAC, as presented in Certified Design Material Tables 2.3.2-1 and 2.4.6-1.

This change has no impact on safety. In fact, it refines the System 80+ design to ensure that the CS and SC pumps are more likely to meet their safety functions.

DCD Markups Attached? Yes

Figure 6.3.2-1B (similar on 6.3.2-1A)

Figure 6.2.1-16), the long-term airborne elemental concentration is calculated to be less than 20 percent of the organic. Thus, depression of the sump pH due to the long-term production of HCl by the irradiation of the electrical insulation and the radiation-induced nitric acid formation would not impact the dose assessment.

The transient spray removal lambdas for the 10 CFR 100 LOCA analysis are shown on Figure 6.5-6.

6.5.3.4 Available Net Positive Suction Head (NPSH)

The IRWST is the suction source for the SI pumps and CS pumps during short term injection and long term cooling modes of post-accident operation. As described in Section 6.8, the Holdup Volume Tank (HVT) performs water collection services after an accident. Spillways allow accumulated water in the HVT to spill back into the IRWST, thereby replenishing IRWST water volume during accident operations. The minimum available NPSH for the SI and CS pumps was determined based on the minimum water level in the IRWST during accident conditions. In addition, the following conservative assumptions are made:

- Fluid conditions in the IRWST are saturated; no credit is taken for an increase in containment pressure.
- The contribution of the volume of water spillage from the RCS and safety injection tanks is conservatively neglected.
- With the CS system actuated, the reactor cavity is assumed flooded and the HVT full to a level that is just below the level at which water begins to return to the IRWST through the spillways.
- Spray water is being held up on surfaces throughout the containment. Locations for the accumulation of water inside the containment include water held up on horizontal surfaces, clogged floor drains, water held up in containment spray piping, water in the containment atmosphere, water film on vertical surfaces, puddles trapped on equipment, water soaked into insulation, and the containment free volume filled with steam.

The SI and CS pumps are located in the reactor building subsphere and are placed low enough below the minimum IRWST fluid level to assure adequate available NPSH. The minimum IRWST fluid volume after an accident has been determined to be 161,000 gallons. This corresponds to a water level elevation

of 75.5 feet.

The calculated available NPSH for the CS pumps ranges from 24 feet at the design flow rate of 5000 gpm to 21.2 feet at a pump runout flow of 5500 gpm. This exceeds the CS pump required NPSH of 20 feet at runout flow. *Insert A*

The calculated available NPSH for the SI pumps is 26.9 feet at a pump runout flow of 1235 gpm. This exceeds the SI pump required NPSH of 20 feet at runout flow.

During a LOCA, the reactor cavity will not be flooded as was assumed in determining the minimum water level of 75.5 feet. Because of this, an additional volume of water will be available to raise the minimum water level in the IRWST by approximately 2 feet, thereby increasing the available NPSH.

Prevention of the entrance of debris into the IRWST and SI and CS pump suction lines is described in Section 6.8.

Insert A: If the SC pump(s) is (are) aligned for containment spray, the minimum available NPSH to the SC pumps is 21.7 feet at 5000 gpm and 20.7 feet at 5500 gpm.

22.2

a typical

or MSLB

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193,000

23

5500

a typical

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System 80+ DCD - Potential Design Changes

Item Number: 3

Summary Description: Revise Safety and Seismic Classification of SDS Components from PSVs to IRWST

Affected DCD Sections: CDM: None
ADM: Tables 3.2-1 and 3.2-2

Description of Change:

This is a consistency change to correct errors, not a design change.

Reason for Change: The safety and seismic classification of the SDS spargers, vacuum breakers and piping from the SDS valves and PSVs to the IRWST were specified inconsistently. ABB-CE desires to change classifications so that all components in this portion of the SDS are classified consistently.

Description: This change revises the safety class and seismic category of the spargers and vacuum breakers, and upgrades the seismic category of the piping, so that they have the same safety classification and seismic category. For the spargers and vacuum breakers, the safety class will change from 2 to Non-Nuclear Safety (NNS) and the seismic category will change from I to II. For the piping, the seismic category will change from non-seismic (NS) to II. The quality class will change from 1 to 2 for all NNS components.

This change has no impact on safety: The spargers are not pressure retaining components, and do not perform a safety-related function. Changing the safety classification from 2 to NNS changes the design code from ASME B&PV Section III to ASME/ANSI B31.1, which has similar design rules for structural integrity. Changing the seismic category from I to II requires that the spargers still maintain sufficient integrity during seismic events, such that they would not damage safety-related equipment during the Safe Shutdown earthquake.

The vacuum breakers do not perform a safety-related function. Changing the seismic category from I to II requires that the vacuum breakers still maintain sufficient integrity during seismic events, such that they do not damage safety-related equipment.

This change provides consistent safety, seismic, and quality classification for these components. As currently specified, there would be several interfaces between safety and non-safety equipment, which would be complex to design and construct. The proposed revisions would provide consistent interfaces and higher assurance of correct construction.

DCD Markups Attached? Yes

Table 3.2-1 Classification of Structures, Systems, and Components

Component Identification	Safety Class	Seismic Category	Location ⁽²⁾	Quality Class ⁽²⁾
Reactor Coolant System				
Reactor Vessel	1	I	RC	1
Steam Generators (primary/secondary)	1/2 [1]*	I	RC	1
Pressurizer	1	I	RC	1
Reactor Coolant Pumps [2,3,9]*	1	I	RC	1
Piping within Reactor Coolant Pressure Boundary [5]	1/2 [4]	I	RC	1
Control Element Drive Mechanisms	[6]	[6]	RC	1
Core Support Structures and Internals Structures [7]	3	I	RC	1
Fuel Assemblies [8]	2	I	RC	1
Control Element Assemblies [8]	3	I	RC	1
Closure Head Lift Rig	NNS	II [10]	RC	2
Heated Junction Thermocouple Probe Assembly	1/3 [12]	I	RC	1
HJTC Pressure Housing	1	I	RC	1
ICI Cable Tray Support Frame	3	I	RC	1
ICI Holding Frame	NNS	NS	RC	3
ICI Guide Tubes	1	I	RC	1
ICI Guide Tube Supports	1	I	RC	1
ICI Seal Housing	1	I	RC	1
ICI Seal Table	1	I	RC	1
Piping [27]	1/2	I	RC	1
Valves [27]	1/2	I	RC	1
In-containment Water Storage System				
IRWST	3	I	RC	1
Holdup Volume Tank	3	I	RC	1
Pressure Relief Dampers	3	I	RC	1
Cavity Flooding System				
Piping	2	I	RC	1
Valves	2	I	RC	1
Safety Depressurization System				
Valves	1/2/NNS	1/II	RC	1/2
Piping	1/2/NNS	1/II	RC	1/2
Spargers	2/NNS	II	RC	2
Safety Injection System				
Safety Injection Pumps	2	I	RB	1
Safety Injection Tanks	2	I	RC	1
Piping [24,27]	1/2	I	RB/RC	1
Valves [27]	1/2	I	RB/RC	1

* Refer to Notes at end of table.

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location ^[26]	Quality Class ^[29]
Station Air System				
Air Compressors	NNS	NS	SB	3
Air Dryers/Filters	NNS	NS	SB	3
Air Receivers	NNS	NS	SB	3
Piping [27]	2/NNS	1/NS	All	1/3
Valves [27]	2/NNS	1/NS	All	1/3
Breathing Air System				
Air Compressors	NNS	NS	SB	3
Piping [27]	2/NNS	1/NS	All	1/3
Valves [27]	2/NNS	1/NS	All	1/3
Air Receivers	NNS	NS	SB	3
Air Dryer/Filters	NNS	NS	SB	3
Compressed Gas Systems				
High Pressure Gas Cylinders	NNS	NS	YA	3
Pressure Regulators	NNS	NS	YA	3
Leak Detection Systems	NNS	NS	All	3
Liquid Nitrogen Evaporators	NNS	NS	YA	3
Piping [26, 27]	2/NNS	1/NS	All	1/3
Valves [27]	2/NNS	1/NS	All	1/3
Fire Protection System				
Jockey Pump	NNS	NS	FP	2
Backup Storage Tank	NNS	I	NA	1
Fire Pumps	NNS	NS	FP	2
Backup Fire Pump	NNS	I	NA	1
Storage Tanks	NNS	NS	FB	2
Water Spray Systems (Deluge and Sprinkler) Piping, Valves [16, 27]	2/NNS	1/II/NS	TB/NA/RC/RB/DG/SB	1/2
Hose Systems/Standpipes [16, 27]	2/NNS	1/NS	All	1/2
Portable Fire Extinguishers [16]	NNS	NS	All	2
Exterior Distribution System				
Piping	NNS	NS	YA	2
Valves	NNS	NS	YA	2
Strainers	NNS	NS	YA	2
Alternate AC Source/Combustion Turbine-Generator	NNS	NS	YA	2
DG Engine Fuel Oil System [17]				
Fuel Oil Storage Tanks	3	I	DF	1
Recirculation Pumps	NNS	NS	DF	3
Booster Pumps	3	I	DG	1
Fuel Oil Day Tanks	3	I	DG	1
Emergency Diesel Generator System				
Diesel Generators	3	I	DG	1

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location ^[25]	Quality Class ^[29]
Main Steam Supply System Piping [21] Steam Generator to MSIV's Other	2 NNS	I NS	RC/MS MS/NA/TB	1 3
Main Steam Supply System Valves [21] Safety Valves MSIV's, MSIV Bypass Valves Atmospheric Dump Valves Valves	2 2 2 2/NNS	I I I I/NS	MS MS MS NA/MS/TB	1 1 1 1/3
Containment Hydrogen Recombiner System Hydrogen Recombiners Hydrogen Analyzers Hydrogen Recombiner Control Panel Piping [27] Valves [27]	2 2 3 2 2	I I I I I	NA NA NA NA/RC NA/RC	1 1 1 1 1
Steam Generator Blowdown System [22] Flash Tank Heat Exchanger Filter Demineralizers Piping [27] Valves [27]	NNS NNS NNS NNS 2/NNS 2/NSS	NS NS NS NS I/NS I/NS	TB TB TB TB RC/TB/MS RC/TB/MS	2 2 2 2 1/2 1/2
Steam Generator Wet Layup Recirculation System [22] Piping [27] Valves [27]	2/NNS 2/NSS	I/NS I/NS	RC/TB/MS RC/TB/MS	1/3 1/3
Hydrogen Mitigation System Hydrogen Igniters	NNS	I	RC	2
Potable and Sanitary Water Systems	NNS	NS	YA	3
Instrumentation and Control Systems Plant Protection System (PPS) The PPS includes the electrical and mechanical devices and circuitry (from sensors to actuation device input terminals) involved in generating the signals associated with the two protective functions defined below:				

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location ^[28]	Quality Class ^[29]
Nuclear Annex Structure				
Control Area	3	I	NA	1
EFW Tank/Main Steam Valve House Area	3	I	NA	1
Emergency Diesel Generator Areas	3	I	NA	1
CVCS/Maintenance Area	3	I	NA	1
Fuel Handling Area	3	H I	NA	1
Other Structures				
Unit Vent	NNS	II	NA/RB	2
Turbine Building	NNS	II	TB	2
Radwaste Building [28]	NNS	I II	RW	2
Station Service Water Pump/Intake Structure	3	I	SP	1
Component Cooling Water Heat Exchanger Structures and Pipe Tunnels	3	I	CX/YD	1
Diesel Fuel Storage Structure	3	I	DF	1
Station Services Building/Auxiliary Boiler Structure	NNS	NS	SB	3
Administration Building	NNS	NS	ADB	3
Warehouse	NNS	NS	WH	3
Fire Pump House	NNS	NS	FP	3
Alternate AC Source/Combustion Turbine-Generator Structure and Fuel Tank	NNS	NS	YA	2
Dikes				
Dike (Holdup, Boric Acid Storage and Reactor Makeup Water Tanks) [28]	NNS	II	YA	2
Dike (Condensate Storage Tank) [28]	NNS	II	YA	2
Cranes				
Polar Crane	NNS	II	RC	2
Cask Handling Hoist	NNS	II	NA	2
New Fuel Handling Hoist	NNS	II	NA	2
Component Supports [23]	1/2/3/NNS	I/NS	All	1/2/3

Ref. Change Item No. 3

System 80 +

Design Control Document

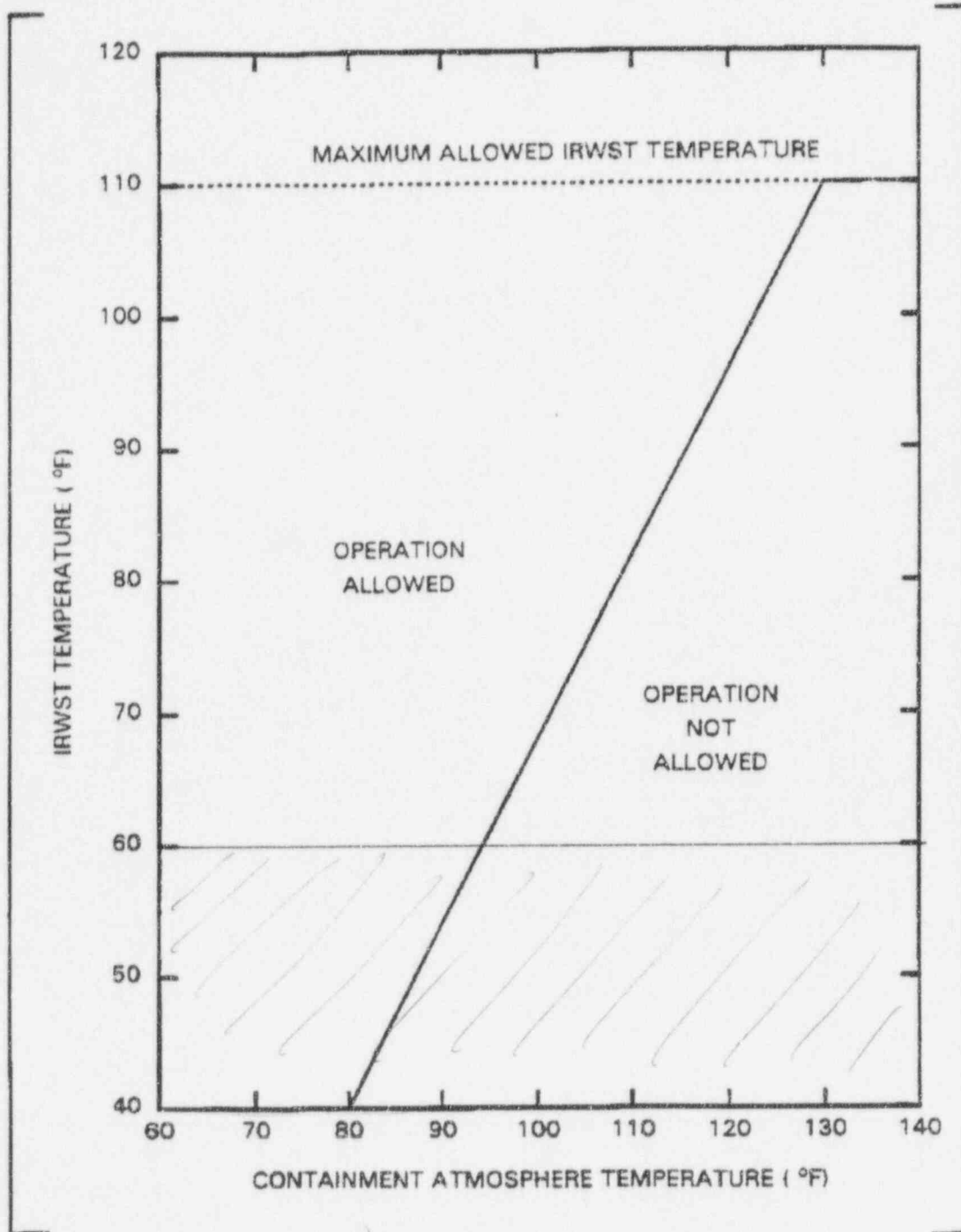
Table 3.2-2 Safety Class 1, 2 & 3 Valves (Cont'd.)

Component Identification	Location/Description	Safety Class	Seismic Category	Quality Class
Pool Cooling and Purification System (PCPS) [1] (Cont'd.)				
PC-208, 209	Cooling pump discharge isolation	3	I	1
PC-211, 212	Cooling HX inlet isolation	3	I	1
PC-213, 214	Cooling HX outlet isolation	3	I	1
PC-249	IRWST return line isolation	3	I	1
PC-257, 258	Refueling pool discharge isolation	2	I	1
PC-291, 292	Refueling pool inlet isolation	2	I	1
PC-300, 301, 302, 303	Cooling flow indication isolation	3	I	1
PC-320, 321	Cooling pump suction pressure	3	I	1
Safety Depressurization System (SDS)				
RC-406, 407, 408, 409	Rapid depressurization	1	I	1
RC-410, 411, 412, 413	Pressurizer vent	1	I	1
RC-414, 415, 416, 417	Reactor vessel vent	1	I	1
RC-418	RCGVS vent to RDT	2	I	1
RC-419	RCGVS vent to IRWST	2	I	1
RC-263, 264	RD pressure indication	2	I	1
RC-267	RCGVS pressure indication	2	I	1
RC-XXX	SDS/Safety valve sparger line-1 vacuum breaker	2 N/A	I N/A	1
RC-XXX	SDS/Safety valve sparger line-2 vacuum breaker	2 N/A	I N/A	1
Safety Injection System (SIS) [1]				
SI-100, 101	IRWST return check valve	2	I	1
SI-102, 103	IRWST isolation valve test	2	I	1
SI-104, 105	CS pump suction isolation	2	I	1
SI-106, 107	SCS pump suction isolation	2	I	1
SI-108, 109	SCS pump suction pressure indication isolation	2	I	1
SI-113, 123, 133, 143	Safety injection containment check	2	I	1
SI-115, 125, 135, 145	SI flow indication isolation	2	I	1
SI-116, 126, 136, 146	SI flow indication isolation	2	I	1
SI-117, 127, 137, 147	SIT pressure indication isolation	2	I	1
SI-119, 129, 139, 149	SIT pressure indication isolation	2	I	1

Delete
Delete

ITEM 4

IRWST
3.5.4



DELETE
FROM
FIGURE

Figure 3.5.4-1
Allowed IRWST Temperature vs. Containment Atmosphere Temperature

System 80+ DCD - Potential Design Changes

Item Number: #5

Summary Description: Additional CEA locations in Reactor Core

Affected DCD Sections: CDM: Yes, Figure 2.2.1.3; eliminate the maximum number of CEDMs in Section 2.2.2
ADM: Yes, Figure 4.2.11, Figure 4.3-46, Figure 4.3-47, Text in Section 4.2.2.4

Description of Change:

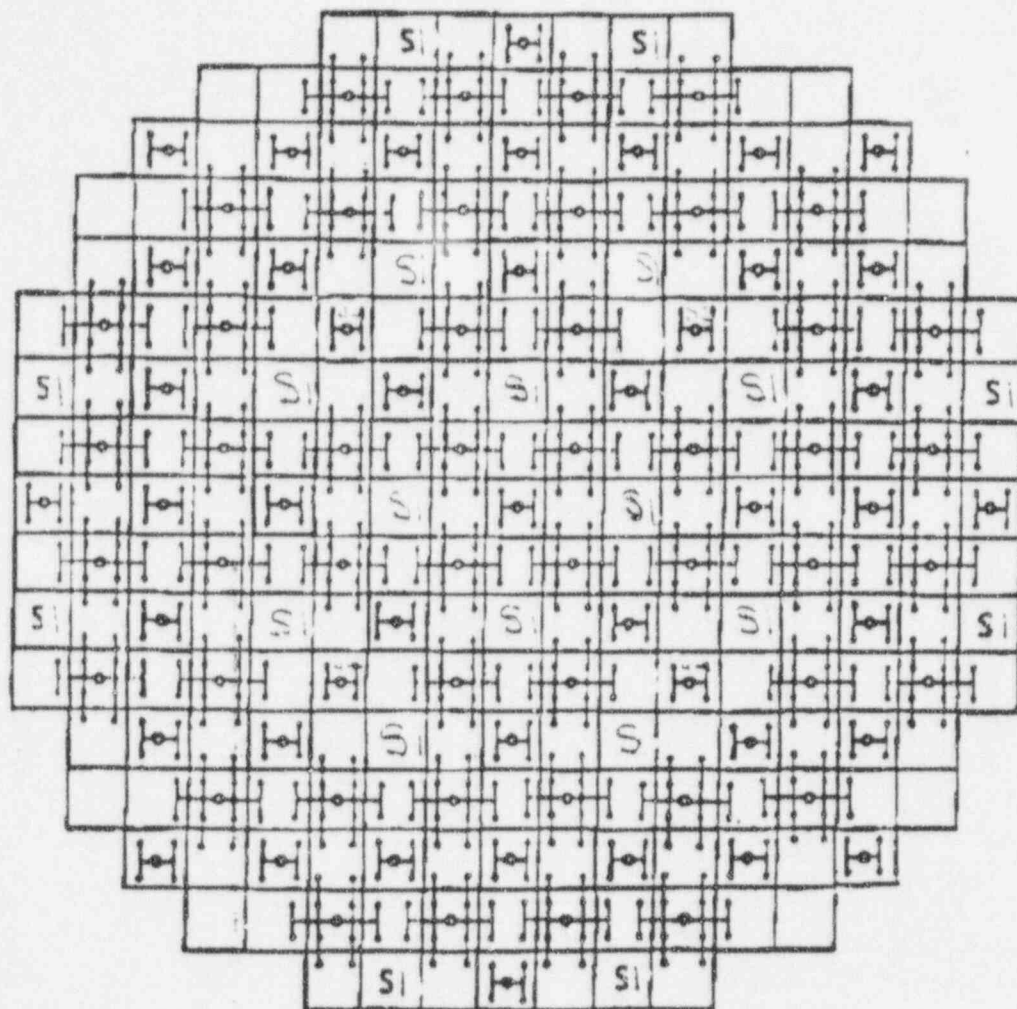
The objective of the change is twofold: (1) allow for the possibility of having 4 element CEAs at twelve specific core locations; and (2) allow for the possibility of replacing 4-element CEAs with 12-element CEAs at for specific core locations. Part (1) of the change is accomplished by identifying twelve core locations (in addition to the eight core locations previously identified) for spare locations which may contain 12 element CEAs instead of 4 element CEAs. Not all of the locations identified for spare 4 element CEAs would necessarily contain CEAs. The minimum number of CEAs would remain at 93.

The proposed change would allow flexibility in choosing locations for additional CEAs and/or CEA elements. The addition of CEAs in the central core region would permit enhanced maneuvering capability. The addition of CEAs or CEA elements could also improve shutdown margin. All locations identified for additional CEAs satisfy the minimum reactor vessel head ligament requirement.

The ADM (Section 4.1.1) currently allows changes to certain features and evaluated parameters for the fuel system design, nuclear design, and thermal and hydraulic design of the initial core without prior NRC review and approval provided these changes are within certain acceptance criteria. With the propose change, Section 4.1.1 would remain unchanged.

The CDM (Figure 2.3.1-3) and the ADM (Figure 5.3-7) show a minimum of 103 CEDM and instrumentation nozzles, and a minimum of 2 HJTC probes. The proposed change would maintain these minimum values. Since specific CEDM locations are not shown in the ADM, the change permits the selection of 8 different core locations for spare CEAs from those currently shown, as well as the addition of spare CEAs at other core locations.

DCD Markups Attached? Yes



12 ELEMENT CEAS



4 ELEMENT CEAS



LOCATIONS WHICH MAY CONTAIN 4 ELEMENT CEAS
~~DENOTES SPARE CEA LOCATIONS FOR 4 CEA ELEMENTS~~



LOCATIONS WHICH MAY CONTAIN 12 ELEMENT CEAS
~~SPARE LOCATIONS FOR 2 CEA ELEMENTS~~

NOTE:

The number of CEAs shown on this figure represents
 a minimum number of CEAs.

2.2.2 Control Element Drive Mechanism

Design Description

The control element drive mechanism is a magnetic jack device that positions and holds the control element assemblies relative to the fuel assemblies.

The primary safety-related function of the Control Element Drive Mechanism (CEDM) is to release the Control Element Assembly (CEA) upon termination of electrical power to the CEDM. A minimum of 93 CEDMs is required, ~~however, a maximum of one hundred one CEDM can be installed.~~

The CEDM also acts as a primary pressure boundary as part of the Reactor Coolant System. Refer to Section 2.3.1 for CEDM primary pressure boundary aspects.

Inspections, Tests, Analyses, and Acceptance Criteria

None

The initial test program addressed in Section 2.11 will test the ability of the CEDM to release the CEA upon termination of electrical power to the CEDM.

The Basic Configuration of the CEDM primary pressure boundary components will be verified as part of Section 2.3.1.

The CEDM pattern will be verified as part of Section 2.2.1.

a spider structure which couples to the control element drive mechanism (CEDM) drive shaft extension. The neutron absorber elements of a four-element CEA engage the four corner guide tubes in a single fuel assembly. The four-element CEAs are used for control of power distribution and core reactivity in the power operating range. The twelve-element CEAs engage the four corner guide tubes in one fuel assembly and the two nearest corner guide tubes in adjacent fuel assemblies. The twelve-element CEAs make up the balance of the control groups and provide the core with strong shutdown rods. The control element assemblies are shown in Figures 4.2-3 through 4.2-5 and Figure 4.2-14. The pattern of CEAs (total of 93) is shown in Figure 4.2-11. ~~Note that up to eight~~ Additional CEAs may be installed if desired for additional flexibility or future use. Twenty-five of the 93 CEAs are part-strength CEAs (PSCEAs).

Part-strength CEAs are differentiated from full-strength CEAs by using alphanumeric serialization instead of the numerical system used on the full-strength CEAs.

All control elements are sealed by welds which join the CEA cladding to an Inconel 625 nose cap at the bottom, and an Inconel 625 connector at the top which makes up part of the end fitting. The end fittings in turn, are threaded and crimped in place by a locking nut to the spider structure which provides rigid lateral and axial support for the control elements. The spider hub bore is specially machined to provide a point of attachment for the CEA extension shaft.

The control elements of a twelve-element full-strength CEA consist of an Inconel 625 tube loaded with a stack of cylindrical absorber pellets. The absorber material consists of 73% TD boron carbide (B_4C pellets, with the exception of the lower portion of the elements, which contain reduced diameter B_4C pellets wrapped in a sleeve of Type 347 stainless steel (felt metal).

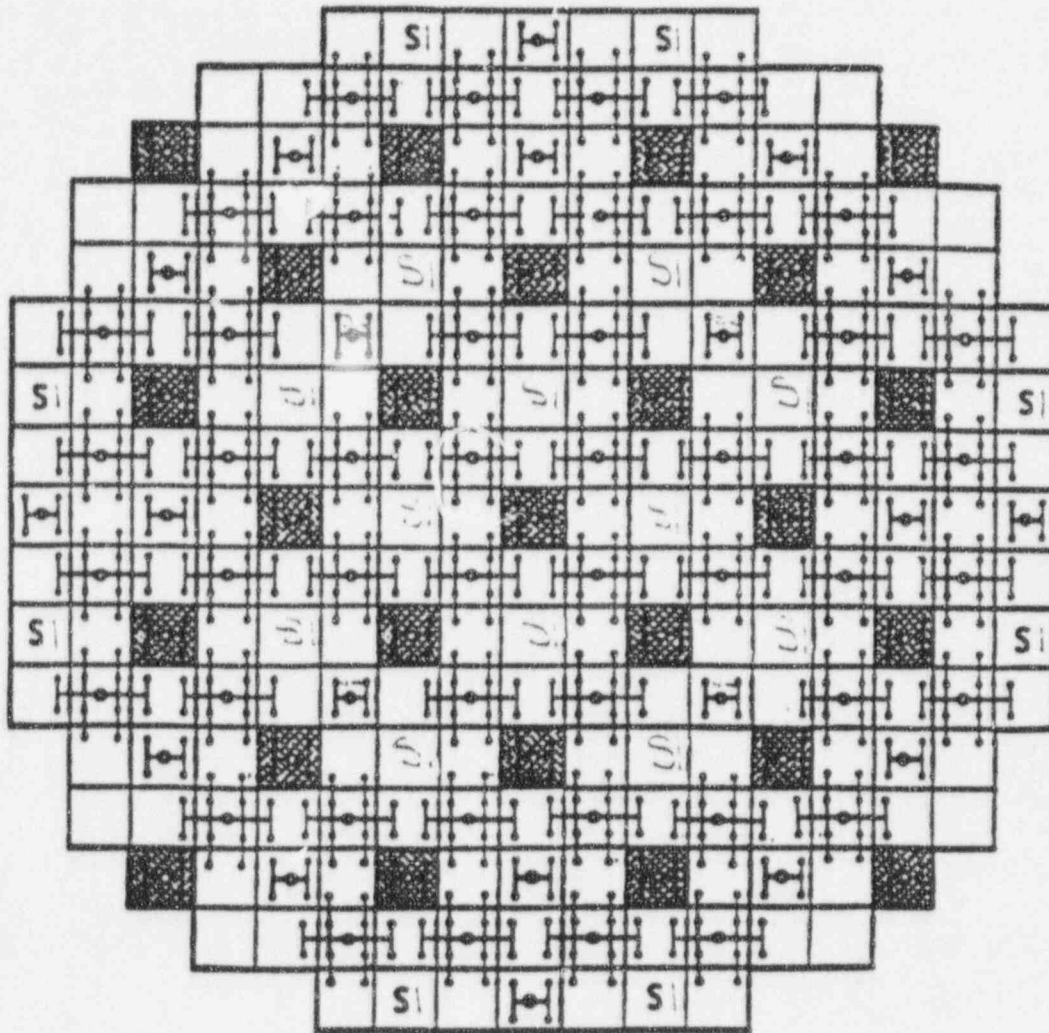
The design objective realized by the use of felt metal and reduced diameter B_4C pellets in the element tip zones is that as the B_4C pellets swell due to irradiation, the felt metal sleeve compresses as a result of the applied loading. This compression limits the amount of induced strain in the cladding. Therefore buffering of the CEA following scram, which occurs when the element tips enter the reduced diameter portion of the fuel assembly guide tubes, is not affected with long term exposure of the CEA to reactor operating conditions.

During normal power operation, all of the twelve-element CEAs are expected to be in the fully withdrawn position. Thus, the local B-10 burnup progresses at a lower rate, and CEA life is prolonged. Above the absorber column is a plenum which provides expansion volume for helium released from the B_4C . The plenum volume contains a Type 302 stainless steel holddown spring, which restrains the absorber material against longitudinal shifting with respect to the clad while allowing for differential expansion between the absorber and the clad. The spring develops a load sufficient to maintain the position of the absorber material during shipping and handling.

The control elements of a four-element full-strength CEA consist of an Inconel 625 tube loaded with a stack of cylindrical Ag-In-Cd absorber bars. This CEA design is used for the regulating banks. Two design objectives are realized by use of Ag-In-Cd absorber over the full active length:

- CEA Cladding Dimensional Stability

Because of its high ductility and low strength, the Ag-In-Cd will not deform the CEA cladding. Buffering of the CEA following scram, which occurs when the corner element tips enter the reduced diameter portion of the fuel assembly guide tubes, is not degraded with long-term exposure of the CEA to reactor operating conditions.



12 ELEMENT FULL STRENGTH CEA



4 ELEMENT FULL STRENGTH CEA



4 ELEMENT PART STRENGTH CEA



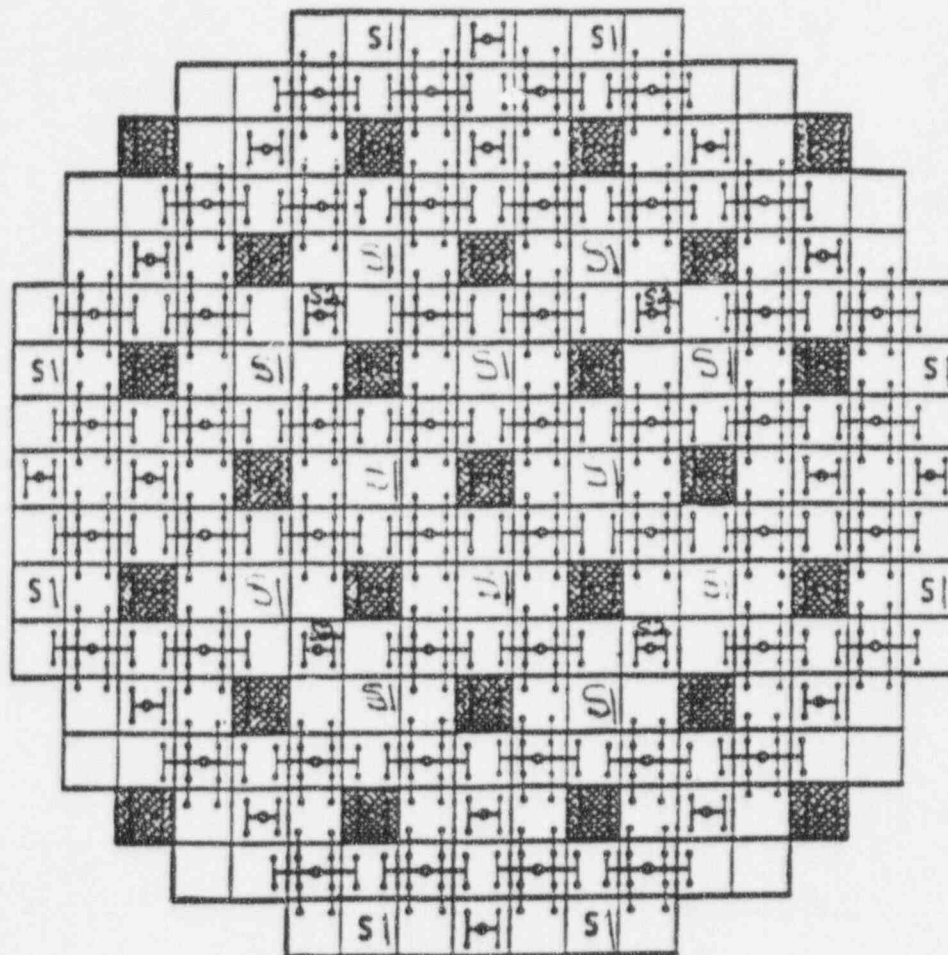
LOCATIONS WHICH MAY CONTAIN 4 ELEMENT CEAS
~~DENOTES SPARE CEA LOCATIONS FOR~~
~~FOR OPEN-MARKET PLUTONIUM~~
~~RECYCLE 4 CEA ELEMENTS~~



LOCATIONS WHICH MAY CONTAIN 12 ELEMENT CEAS
~~SPARE LOCATIONS FOR 2 CEA ELEMENTS~~

Control Element Assembly Locations

Figure 4.2-11



12 ELEMENT FULL STRENGTH CEA



4 ELEMENT FULL STRENGTH CEA



4 ELEMENT PART STRENGTH CEA



LOCATIONS WHICH MAY CONTAIN 4 ELEMENT CEAS
 DENOTES SPARE ORA LOCATIONS FOR
 RECYCLE CEA ELEMENTS

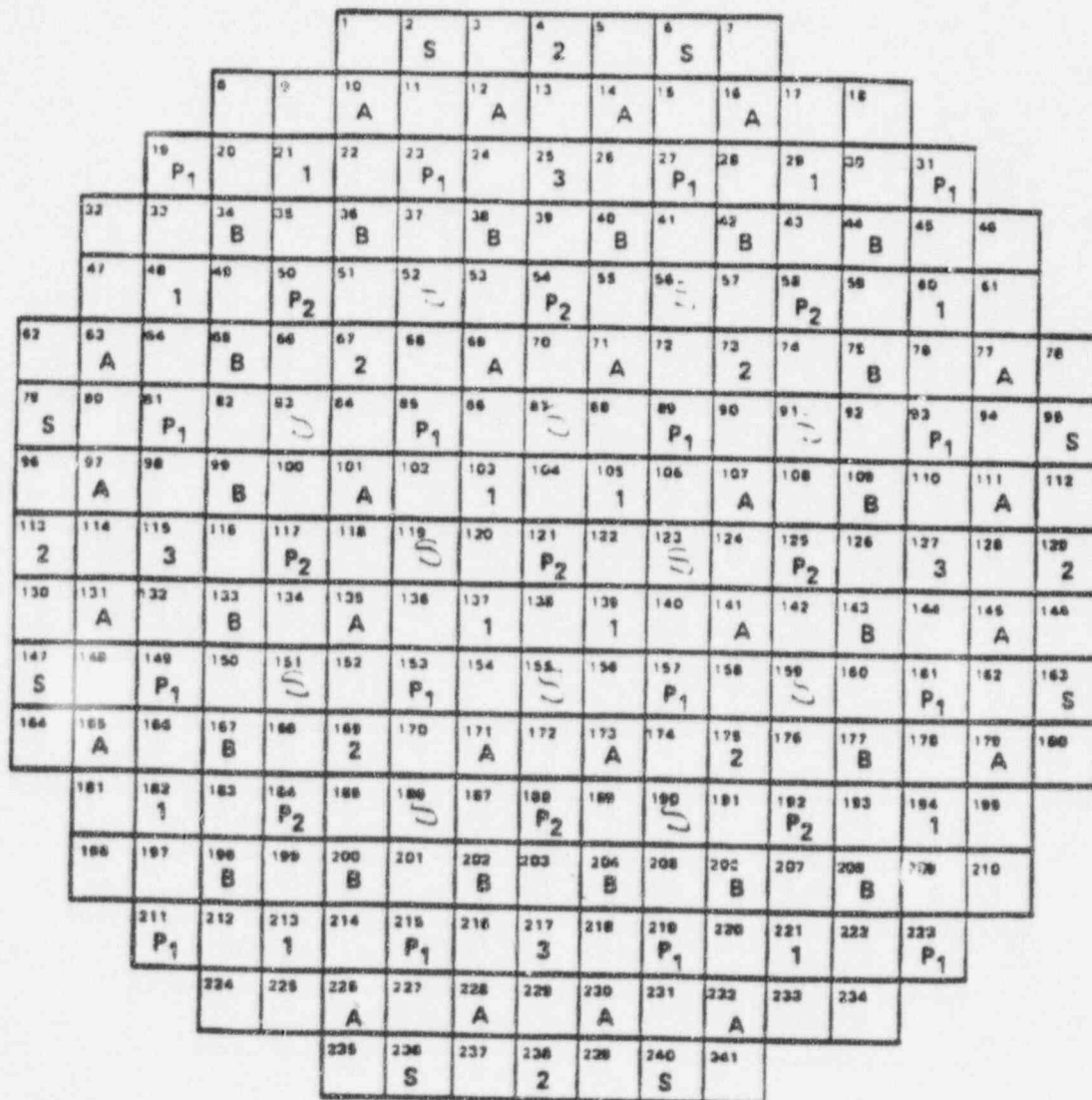


LOCATIONS WHICH MAY CONTAIN 12 ELEMENT CEAS
 SPARE LOCATIONS FOR CEA ELEMENTS

Control Element Assembly Locations

Figure 4.3-46

- P_2 - PART-STRENGTH GROUP 2 (LEAD)
 P_1 - PART-STRENGTH GROUP 1
 3 - FULL-STRENGTH REGULATING GROUP 3 (LEAD)
 2 - FULL-STRENGTH REGULATING GROUP 2
 1 - FULL-STRENGTH REGULATING GROUP 1
 B - SHUTDOWN GROUP B
 A - SHUTDOWN GROUP A
 S - SPARE CEA LOCATIONS



CEA Group Identification

Figure 4.3-47

System 80+ DCD - Potential Design Changes

Item Number:

86

Summary Description:

Add RCS Mid-loop Level Measurement Tank and Instrumentation Across the Hot Leg

Affected DCD Sections:

CDM: None
ADM: Chapter 5
Chapter 7
Appendix 19.8A

Description of Change:

A detailed engineering evaluation was performed to confirm the feasibility of adding tanks connected directly across each of the RCS hot leg pipes to measure the hot leg water level during Mid-Loop operations using HJTC instruments. The current DCD Chapter 5 includes the RCS P&ID and the cross section of the Reactor Vessel. The Shutdown Risk evaluation reported in DCD Chapter 19.8A provides a figure (Figure 2.8-4) which indicates that two additional sets of HJTC's are inserted through the reactor vessel head to measure the hot leg water level. The DCD text indicates this level should be measured to an accuracy of plus or minus 1 inch. In Chapter 19 the tank across the hot leg will be shown for measuring the hot leg water level during mid-loop operations. The water level will be measured with heated junction thermocouples that operate in the same manner as the ICCMS HJTC located in the reactor vessel.

The reasons of the proposed change are as follows:

1. the instruments are permanently installed and are not affected by refueling activities in the reactor vessel head area.
2. the reliability of the instruments is enhanced by relocating them to a more benign area.
3. there is improved flexibility for plant operations and maintenance for servicing the instruments and installing nozzle dams.
4. the water level measurements are taken closer to the Shutdown Cooling suction line which is the point of interest.

Locating the Mid-Loop HJTCs through the reactor vessel head have limitations that the proposed change eliminates. The cable to the head mounted instruments must be disconnected when the reactor vessel head is removed. The disconnection affects the ability of the instruments to be used. Having the instruments permanently connected reduces the potential for human error. Since the position of the reactor vessel head (and components attached to it) changes when the head is detensioned, the instruments should only be used with the head tensioned. The amount of position change is a function of the fuel burnup and is therefore not constant from fuel cycle to fuel cycle or from before refueling to after refueling. Relocating the instruments will reduce human error and increase operational flexibility. The current design reduces the flexibility in

System 80+ DCD - Potential Design Changes

when the nozzle dams can be installed or removed from the steam generator. The proposed change allows more flexibility and will result in the nozzle dams being installed for a shorter period.

The location of the instruments in the reactor vessel means that they are exposed to reactor pressure, temperature and radiation. These conditions will decrease the operating life of the instruments compared to locating the instrument in the tanks. Instrument replacement in the reactor vessel is more difficult and results in higher personnel exposure.

The Mid-Loop HJTC System uses the proven Heated Junction Thermocouples to measure the water level that are the same design as the ICCMS HJTCs. The HJTC heater controllers are the same as the ICCMS HJTC heater controllers and each controller is connected to a separate power supply. Placing the HJTC instruments in the tank allows a larger instrument diameter which results in higher resolution of the HJTC by using radial distance to accomplish thermal separation of the HJTCs. The instrument operating environment is at atmospheric pressure and temperatures of less than 212°F which will increase the operating life and increase reliability. The water level is indicated in the Control Room.

There is a fluid connection from the bottom of the hot leg to the bottom of the tank. An air connection from the top of the hot leg to the top of the tank causes an equalization of the water level in the hot leg and tank during mid-loop conditions. There are two (2) isolation valves on each of the connecting lines. The connecting lines up to and including the second isolation valve are designed RCS conditions using the ASME B & PV Code, Section III, Class 1. The remaining portions of the connecting lines and the tank up to and including the tank drain valve are designed for reactor operating temperature and pressure in accordance with ASME B & PV Code, Section VIII. The drain line from the tank connects to the liquid radwaste system.

DCD Markups Attached? Yes

5.0 Reactor Coolant System and Connected Systems

5.1 Summary Description

The reactor is a pressurized water reactor (PWR) with two coolant loops. The reactor coolant system (RCS) circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to a secondary system. The steam generators provide the interface between the reactor coolant (primary) system and the main steam (secondary) system. The steam generators are vertical U-tube heat exchangers with an integral economizer in which heat is transferred from the reactor coolant to the main steam system. Reactor coolant is prevented from mixing with the secondary steam by the steam generator tubes and the steam generator tube sheet, making the RCS a closed system thus forming a barrier to the release of radioactive materials from the core of the reactor to the secondary system and containment building.

The arrangement of the RCS is shown in Figures 5.1.3-1 and 5.1.3-2. The major components of the system are the reactor vessel; two parallel heat transfer loops, each containing one steam generator and two reactor coolant pumps; a pressurizer connected to one of the reactor vessel hot legs; and associated piping. All components are located inside the containment building.

Table 5.1.1-1 shows the principal pressures, temperatures, and design minimum flowrates of the RCS under normal steady-state, full-power operating conditions. Instrumentation provided for operation and control of the system is described in Chapter 7.

System pressure is controlled by the pressurizer, where steam and water are maintained in thermal equilibrium. Steam is formed by energizing immersion heaters in the pressurizer, or is condensed by the pressurizer spray to limit pressure variations caused by contraction or expansion of the reactor coolant.

The average temperature of the reactor coolant varies with power level and the fluid expands or contracts, changing the pressurizer water level.

The charging pumps and letdown control valves in the chemical and volume control system (CVCS) are used to maintain a programmed pressurizer water level. A continuous but variable letdown purification flow is maintained to keep the RCS chemistry within prescribed limits. A charging nozzle and a letdown nozzle are provided on the reactor coolant piping for this operation. The charging flow is also used to alter the boron concentration or correct the chemical content of the reactor coolant.

Other reactor coolant system penetrations are the pressurizer surge line in one hot leg; the four direct vessel injection nozzles in the reactor vessel for the safety injection system; two return nozzles to the shutdown cooling system, one in each hot leg; two pressurizer spray nozzles; vent and drain connections; and sample and instrument connections, including connections for the mid-loop tank.

Overpressure protection for the reactor coolant pressure boundary is provided by four spring-loaded ASME Code safety valves connected to the top of the pressurizer. These valves discharge to the in-containment refueling water storage tank, where the steam is released under water to be condensed and cooled. If the steam discharge exceeds the capacity of the in-containment refueling water storage tank, it is vented to the containment atmosphere.

Overpressure protection for the secondary side of the steam generators is provided by spring-loaded ASME Code safety valves located in the main steam system upstream of the steam line isolation valves.

range DP sensors measure RCS level in the hot leg region. The narrow range instrumentation includes low and low-low alarms which annunciate in the control room. The wide range DP sensors measure RCS level in the hot leg to the top of the pressurizer. The wide range instrumentation also includes low and low-low alarms which annunciate in the control room. The indication and alarms allow the operator to monitor RCS level from the control room during shutdown operations which require reduced RCS inventory.

Two redundant Refueling Water Level Probes (RWLP) provide independent level indication from the top of the vessel to the fuel alignment plate. The narrow range RWLP assemblies measure reactor coolant liquid inventory in the hot leg region during reduced inventory periods when the reactor head is installed. The wide range RWLP assemblies measure reactor coolant liquid inventory in the upper portion of the vessel to the fuel alignment plate. The basic principle of operation is the detection of a temperature difference between one of the heated thermocouples and the unheated thermocouple at the bottom of the RWLP assembly. Each RWLP assembly includes multiple HJTC sensors, an outer sheath, a seal plug and electrical connectors.

The RWLP thermal hydraulic operating environment is relatively uncomplicated. The narrow range RWLP is used only during non-power operation while the reactor vessel head is in place. The probe assembly is housed in a stainless steel structure that protects it from flow loads.

The narrow range RWLP heated junction thermocouple sensors are more closely spaced in the hot leg region to provide improved resolution. The RWLP's provide indication, high, high-high, low, and low-low alarms in the control room.

RCS temperature is measured using the existing CET temperatures, HJTC unheated sensor temperatures, and RCS Hot Leg RTD temperatures. The CETs and HJTC unheated sensors have high and high-high alarms to annunciate the approach to bulk boiling in the core. The RTDs have a high alarm annunciation. The HJTC unheated sensor temperature is not available when the head is off.

Each train of the SCS has a measurement of SCS flow. This measurement provides indication of return flow to the RCS when either the SCS pump or CS pump is being used for shutdown cooling. Low flow is annunciated in the control room.

To monitor the performance of the SCS and CS pumps, pump suction pressure, discharge pressure and motor current are monitored and annunciated in the control room.

The performance of the SCS heat exchanger is monitored and annunciated by measuring the temperature in the inlet and return lines. Valve position indication provides indication of the system lineup and provides the status of the available flowpaths.

7.7.1.1.16 Steam Generator Tube Rupture Detection Instrumentation

System 80+ incorporates N-16 gamma detection with a scintillation detector and microprocessor based signal conditioning on a header leaving each steam generator. A description and the applicability of using N-16 gammas in detecting steam Generator tube leaks is provided in Section 5.6 of Appendix 5F. The detection system will alert the operator to a SG tube leak condition originating at power and identify which steam generator is affected.

The addition of N-16 radiation detection and monitoring equipment further enhances the diagnosis of steam generator tube leaks or ruptures and provides the operator with more accurate information to assess

System 80+ DCD - Potential Design Changes

Insert in 7.7.1.1.15

The first HJTC system displays the output from the two (2) inadequate core cooling probes located inside the reactor vessel that are available when the reactor vessel head is installed. The inadequate core cooling probes are described in Section 7.5.1.1.7.

The second HJTC system measures the water level in the RCS hot leg pipes during Mode 5 reduced inventory conditions. This Mid-Loop HJTC system consists of an instrument installed in a tank attached to the RCS hot leg pipe. There is a separate tank and instrument attached to each hot leg in the vicinity of the SCS suction connection. There are two (2) connections each tank and corresponding hot leg. One connection is at the bottom of the hot leg to the bottom of the tank. The second connection is at the top of the hot leg to the top of the tank. When the isolation valves on the connections are open, the water level in the tank equalizes with the water level in the tank. The isolation valves are operated from the control room. The position of the valves is indicated in the control room.

The connecting pipe, up to and including the second isolation valve, is designed to the same conditions as the Reactor Coolant System and is ASME B & PV Code, Section III, Class 1. The tank and connecting pipe after the second isolation valve is designed for RCS operating pressure and temperature in accordance with ASME B & PC Code, Section VIII up to and including the tank drain valve.

During Mode 5 reduced inventory conditions, the Mid-Loop HJTC system is connected to the RCS by opening the isolation valves. The RCS mid-loop vent path described in Chapter 19.8A, Section 2.3.3.3 assures the system pressure at near to atmospheric and the fluid temperature at less than 212°F with no boiling.

The Mid-Loop HJTC instrument consists of a vertical array of heated junction and unheated junction thermocouples. The heated junction thermocouples are spaced to obtain the required measurement resolution and provide alarm points for high level (water level approaching the steam generator nozzles) and low level (water level approaching loss of SCS suction). The thermocouple design is described in Section 7.5.1.1.71.2. Since there is no two phase conditions in Mode 5 at the tank, the instrument does not include the phase separator tubes included in the ICCMS HJTC. There are separate heater controllers for each Mid-Loop HJTC instrument. Each controller is connected to a separate power supply to mitigate common mode failure.

The water level in the RCS hot legs is displayed in the control room.

- mitigation planning aimed at the reinitiation of shutdown cooling, delaying the onset of boiling, and delaying core uncover.

The design goals of the instrumentation package are to provide:

- Prevention - enhanced monitoring capabilities for prevention of a complete loss of SCS operation, and
- Mitigation - the timely response to a loss of SCS.

These goals have been achieved with the design features of the System 80+ instrumentation described in the following.

2.8.3.2 Instrumentation Description

Table 2.8-1 describes the instrumentation package for reduced inventory operations included in the System 80+ design. Additional details are provided below.

2.8.3.2.1 Level

Four unique sets of instruments are provided for the measurement of level during RCS draindown and reduced inventory operations. These instruments make up the refueling water level indication system.

The first set of instruments is a pair of wide-range, dP-based level sensors. These sensors are provided to measure level between the pressurizer and the junction of each SCS suction line with the RCS during draindown operations. Another pair of dP-based level sensors is utilized to determine RCS water level once it is within the reactor vessel. These narrow-range level sensors function to measure level between the direct vessel injection (DVI) nozzle and the junction of the SCS suction lines with the RCS.

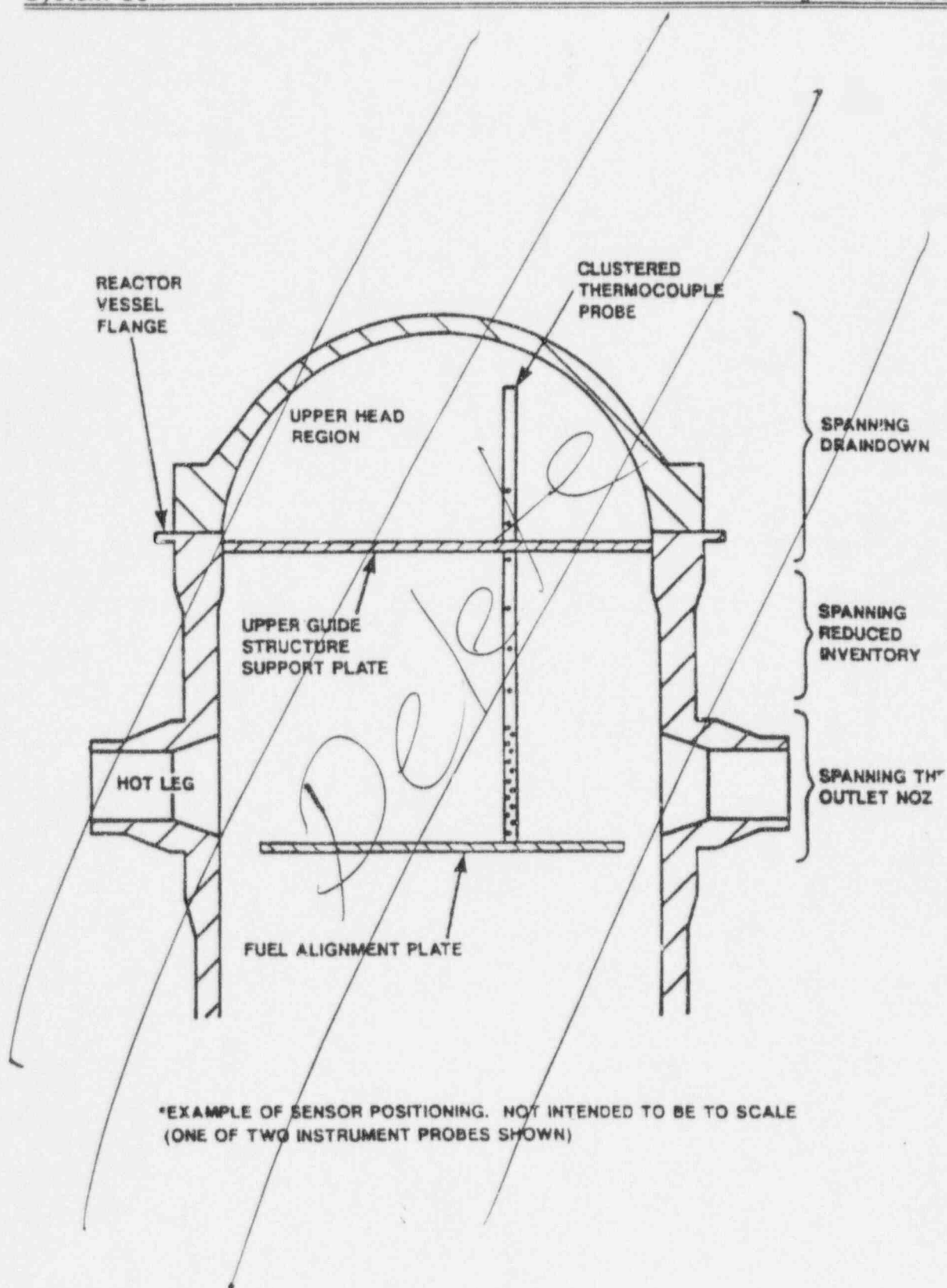
One wide-range and one narrow-range dP instrument are connected to each SCS suction line. Separate lower level taps are provided for each instrument. See Figure 2.8-2. Because of the location of the upper level taps, each of these dP instruments will operate with, or without, the reactor vessel head in place.

In addition to the dP-based instruments described above, two heated-junction thermocouple (HJTC) systems ~~will~~ ^{may} also be available for reactor vessel level measurement during Mode 5 reduced inventory operations. The first system displays the output from the two inadequate core cooling probes which are located inside the reactor vessel. The range of these probes extends from the reactor vessel head to the fuel alignment plate (See Figure 2.8-3). The measurement of RCS water level via these probes is limited only to those periods when the reactor vessel head is installed.

~~A second HJTC system provides narrow-range level indication for mid-loop operations via measurement of reactor vessel water level in the hot leg region.~~ ^{The} This system displays the output from two HJTC probes specifically designed with thermocouples clustered in the hot leg region (See Figure 2.8-4). The benefit of this design is that it permits very accurate measurement when the reactor vessel water level is in the hot legs. ^{is made}

The HJTC systems compensate for the flow gradient across the core associated with the operation of only one SCS suction line. The HJTC instruments are located in areas which minimize the effect of the core outlet nozzles. The HJTC sensors have an accuracy and response time consistent with the maximum

Handwritten note: The tanks are located to place the instruments at the same elevation as the hot legs. The connections to the hot legs allow operation of the water level in the hot legs.

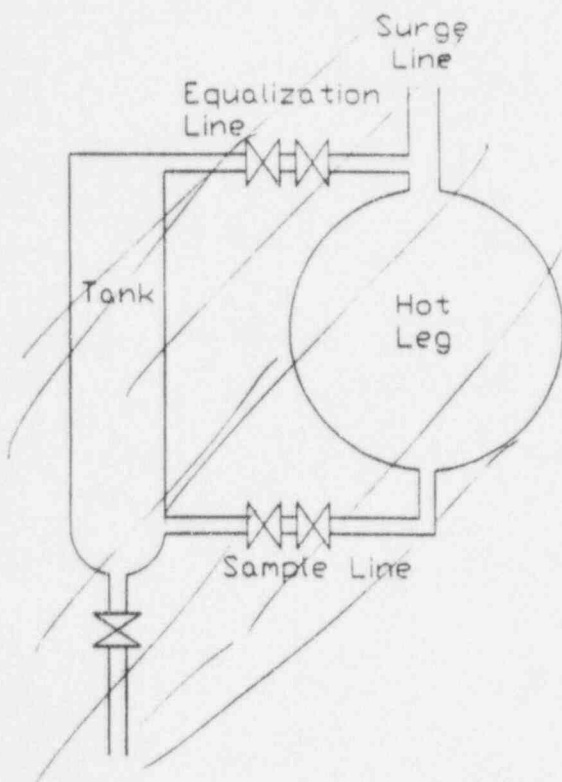


Schematic Representation of One Alternate Design for Narrow-Range Heated Junction Thermocouple Probes

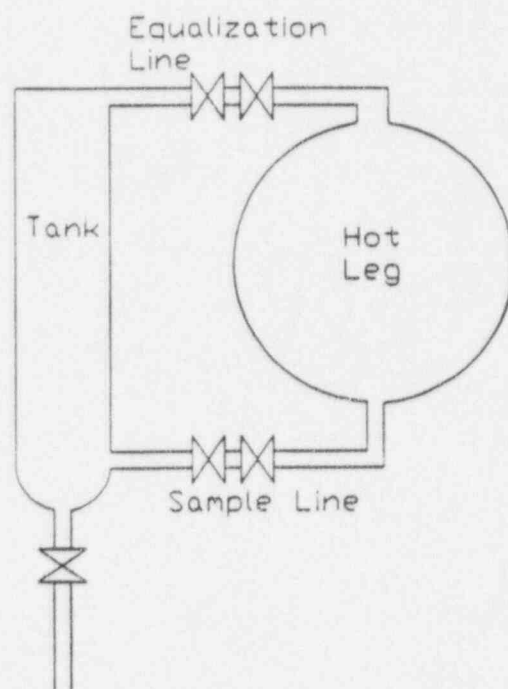
Figure 2.8-4

FIGURE 8
2.8-4

a) Tank Connection to RCS Loop with Surge Line



b) Tank Connection to RCS Loop without Surge Line



Schematic of Connection of Equalization Line to RCS
(Not to scale and some details omitted)

System 80+ DCD - Potential Design Changes

Item Number: 67

Summary Description: Reduce Safety Depressurization System (SDS) Line Size

Affected DCD Sections: CDM: None
 ADM: Figure 5.1.2-3
 Chapter 6.7
 Chapter 5.4

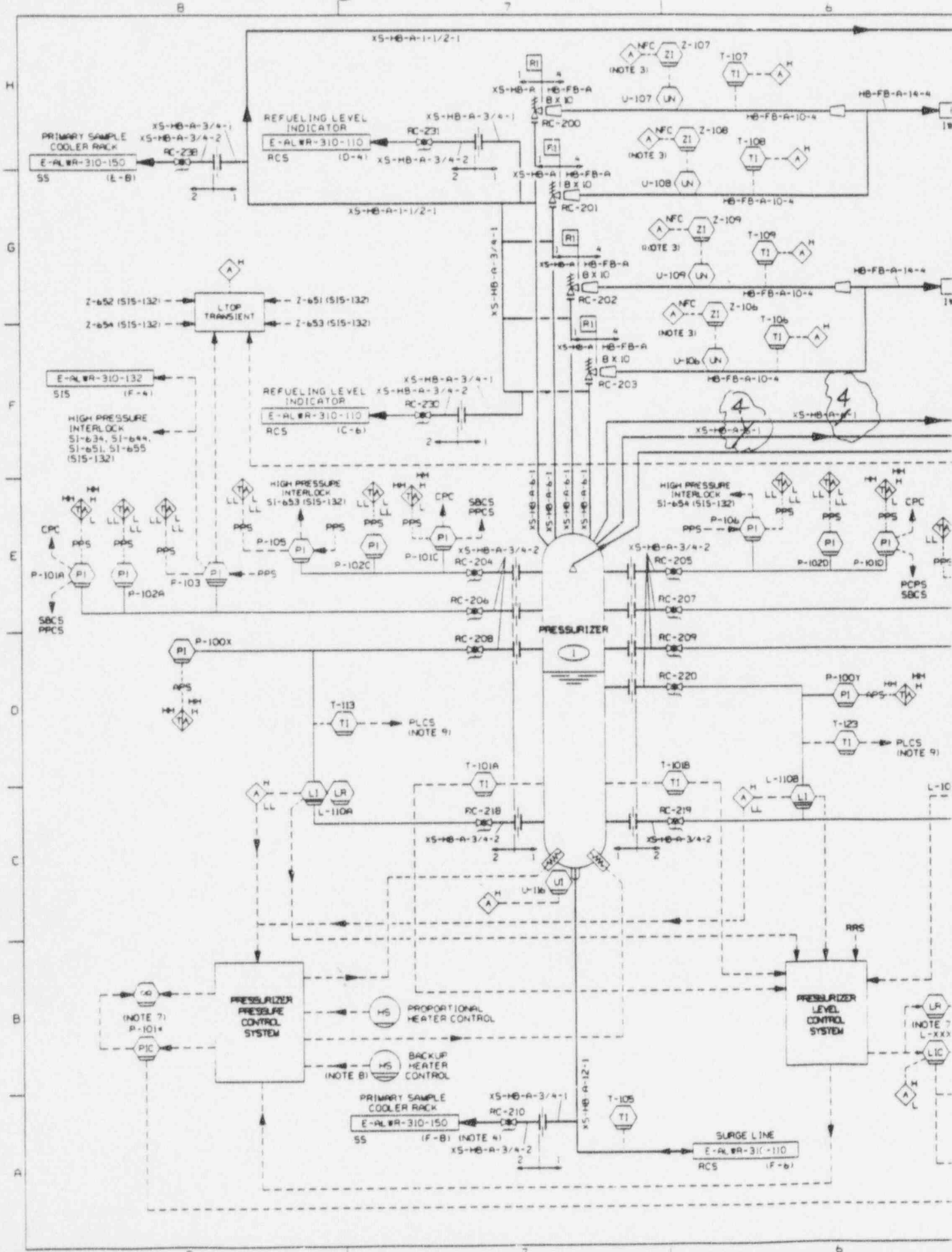
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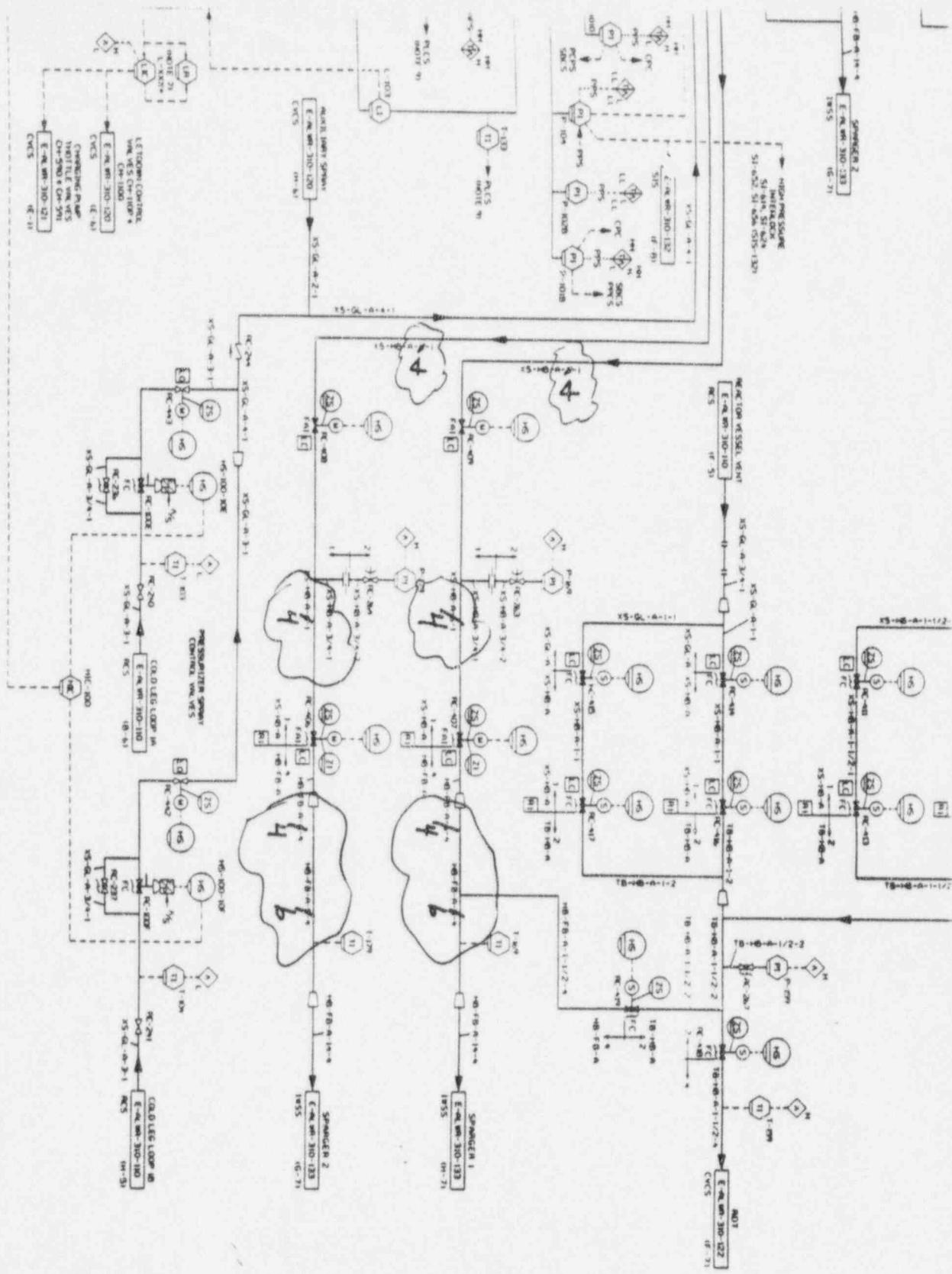
During the Design Certification engineering, a preliminary design analysis for the SDS line size was performed that showed 6 (six) inch piping and valves would pass more than the minimum flow to meet the EPRI requirements to mitigate a Total Loss of Feedwater (TLOFW) event. During detailed engineering, a more detailed system calculation was performed to determine if 4 (four) inch piping and valves would pass adequate flow. The new analyses confirms that a 4 (four) inch system passes more than enough flow to meet the requirements specified in the DCD (Section 6.7) and preserves the validity of the original TLOFW analysis. This change will revise the RCS/SDS P&ID (DCD Figure 5.1.2-3) to revise the SDS piping and valves from 6 to 4 inches. Modifications to the line and valve sizes are also required in Section 5.4 and 6.7.

Using a four inch SDS system for future System 80+ contracts will allow the use of the same gate and globe valves used for Yonggwang 3 & 4. These valves have been tested and qualified for this service, so proven components would be used and retesting would not be required.

DCD Markups Attached? Yes

S80+ DCD Figure 5.1.2-3





Pressurizer & Safety Depressurization System Piping and Instrumentation Diagram

Figure 5.1.2-3

- [illegible]

Table 5.4.10-1 Pressurizer Parameters

Property	Parameter
Design pressure, psia	2500
Design temperature, °F	700
Normal operating pressure, psia	2250
Normal operating temperature, °F	652.7
Internal free volume, ft ³	2400
Normal (full power) operating water volume, ft ³	1200
Normal (full power) steam volume, ft ³	1234
Installed heater capacity, kW	2400
Heater type	Immersion
Spray flow, minimum design capacity, gpm	375
Bypass Spray flow, continuous, gpm	1-6
Nozzles	
Surge, in. (nominal)	12, schedule 160
Spray, in. (nominal)	4, schedule 160
Safety valves, in. (nominal)	6, schedule 160
Safety depressurization, in. (nominal)	4 6, schedule 160
Instrument	
Level, in. (nominal)	3/4, schedule 160
Temperature, in. (nominal)	1, schedule 160
Pressure, in. (nominal)	3/4, schedule 160
Heater, O.D., in.	1-1/4

cylindrical in shape, about six inches in diameter and range in length from fourteen feet to sixteen feet from the junction with the horizontal header. Because of the asymmetry of the IRWST, the spargers on the side with the ICI chase are submerged about ten feet and about twelve feet on the opposite side.

The evaluation of the hydrodynamic loads was performed with methods and codes developed by ABB-Atom and verified through single cell and in-plant measurements. Analyses were completed based on the maximum mass rate of flow for four PSVs during rapid depressurization following the TLOFW event.

Figure 6.7-5 shows typical results (in this case for the header section) of the influence on the loads due to the gas dynamic shock, the water and air clearing phases and finally steam discharge.

The loads on the SDS piping are within the design capability of piping and supports for SRV piping. Thus, design of the SDS piping supports can utilize standard methods for piping analyses and support design.

Table 6.7-1 Safety Depressurization System - Active Valve List

Valve Number	Type	Line Size	Power Source 125V DC Bus	Actuator
Reactor Coolant Gas Vent Valves				
RC-410	Globe	1.5 inch	A	Solenoid
RC-411	Globe	1.5 inch	B	Solenoid
RC-412	Globe	1.5 inch	C	Solenoid
RC-413	Globe	1.5 inch	D	Solenoid
RC-414	Globe	1 inch	B	Solenoid
RC-415	Globe	1 inch	A	Solenoid
RC-416	Globe	1 inch	D	Solenoid
RC-417	Globe	1 inch	C	Solenoid
RC-418	Globe	1.5 inch	A	Solenoid
RC-419	Globe	1.5 inch	B	Solenoid
Rapid Depressurization Valves				
RC-408	Gate	4 8 inch	B	Motor ^[1]
RC-406	Globe	4 8 inch	D	Motor ^[1]
RC-409	Gate	4 8 inch	A	Motor ^[1]
RC-407	Globe	4 8 inch	C	Motor ^[1]

[1] 480 VAC motor operator supplied from Class 1E 125V DC through inverter and step-up transformer.

System 80+ DCD - Potential Design Changes

Item 8: Addition of Alternative LBB Evaluation Method

DCD Section: 3.6.3, 3.9A, Figure 3.9A-12

Description of Change:

The NRC has approved a change for determining leak-before-break (LBB) crack stability criterion on load. The System 80+ certified crack stability criterion on load was that a pipe with a leakage crack length subject to loads of $\sqrt{2} \times (\text{NOP} + \text{Maximum Design Load})$ was required to have significant margin between the material and loading curves. An alternative stability criterion on load is $1 \times (\text{NOP} + \text{Maximum Design Load})$ when the components of the NOP load as well as the NOP plus Maximum Design loads are combined by the absolute summation method. This criterion was approved by the NRC staff on another ALWR design. The System 80+ DCD will be changed to include this criterion as part of an acceptable alternative method for demonstrating LBB.

Related Changes to DCD

DCD Section	Insert #	Revision
3.6.3.7	3.6.3.7	¶ An acceptable alternative method for the margin on loads and margin on crack length evaluations is to combine each component of the NOP load and the Maximum Design Load absolutely. This method is referred to as 'the absolute summation of loads method'. If this alternative method is used, the margin on load for the leakage crack size is reduced from $\sqrt{2}$ to 1. The margin on crack length (2 time the leakage crack size) remains the same.
3.6.3.8, (2nd bullet)	3.6.3.8	<i>Alternatively, cracks of the length that leak at the rate given above can withstand the absolute combination of normal operation load components and maximum design load with a factor of 1.</i>
App 3.9A, ¶ 1.9.6.5.1	3.9A.1	¶ If the absolute summation of loads method is used to evaluate the margin on load and margin on crack length, the PED for the a_1 analysis is constructed using the formulas $M_c = (\text{NOP}_1 + \text{SSE}_1)$ and $M_c = \text{SF}_1$ for the points labeled "1" and $M_c = (\text{NOP}_2 + \text{SSE}_2)$ and $M_c = \text{SF}_2$ for the points labeled "2".
App 3.9A, ¶ 1.9.6.6	3.9A.2	¶ If the absolute summation of loads method is used, the PEDs will be reconstructed, and the piping design, evaluation and reconciliation will be based on the reconstructed PEDs.
Figure 3.9A-12	3.9A.3	⁽¹⁾ When the absolute summation of loads method is used to evaluate the margin on load and the margin on crack length, 'Max. Load/ $\sqrt{2}$ ' becomes 'Max. Load'.

a leak detection capability of 1.0 gpm, with a safety margin of 10. The LBB evaluation of the System 80+ main steam line inside containment is based on a leak detection capability of 1.0 gpm and a safety margin of 10.

See Appendix 3.9A for further discussion of flow rate correlation.

3.6.3.4 Material Properties

For the main coolant loop, the hot and cold leg piping material is SA516 Gr70 or SA508 CL1A. All hot- and cold-leg pipe-to-pipe welds and the pipe-to-reactor vessel, steam generator and reactor coolant pump welds are carbon steel. All main loop component nozzles are SA508 CL 1A, 2 or 3 or SA541 CL 1, 2 or 3. The surge line is SA312 Type 347 or Type 316 stainless steel, resulting in bimetallic safe end welds. The shutdown cooling line and the direct vessel safety injection line are Type 304 or 316 stainless steel. The main steam line is SA516 Gr70.

The stainless steel piping fabricated for the surge, shutdown cooling and direct vessel injection lines are seamless pipes. The detailed analysis of cracks in pipe welds requires consideration of the properties of the pipe and the weld materials. Previous work by C-E has shown that a conservative bounding analysis results when the material stress-strain properties of the base metal (lower yield) and the fracture properties of the weld (lower toughness) are used for the entire structure, (Reference 11). This material representation is applicable to all LBB analyses discussed in Section 1.9 of Appendix 3.9A. For both the final design and as-built configurations, material properties for piping systems subject to LBB which are listed in Section 3.6.3 will be reviewed. If either the base metal or the weld is found to have lower fracture toughness properties than those given in Appendix 3.9A, a LBB reanalysis using the material with the lower fracture toughness properties as the basis for the J-R curve will be performed. The tensile (stress-strain) curves and the J_D vs. Δa curves are required for each material type. Additional commitments with respect to review of final design and as-built configurations for piping systems subject to LBB are given in Section 1.9 of Appendix 3.9A.

3.6.3.5 Leakage Crack Length Determination

It is necessary that hypothesized through-wall cracks open significantly to allow detection by normal leakage monitoring under normal full power loadings.

The method for determining the appropriate leakage crack length is described in Section 1.9.6.2 of Appendix 3.9A.

3.6.3.6 Computation of J-Integral Values

3.6.3.6.1 Range of Crack Sizes

The range of crack lengths are calculated using a detailed stability analysis of the through-wall cracks in the piping evaluated. The finite-element analysis is performed for the leakage crack size and twice that length. This procedure, therefore, considers the stability of a range of crack lengths for all locations selected for the analysis.

3.6.3.6.2 J-Integral

The stability of through-wall cracks is evaluated using the J-integral technique. The J-integral is determined in the finite-element analysis for pressure, normal operation, and maximum design load,

which is the largest of the dynamic loads (due to safe shutdown earthquake, thermal stratified flow, rapid valve closure, or other load) included in the crack stability analysis. The J-integral is determined for two different crack lengths for each geometric model. For the margin on loads evaluation, the J-integral for the leakage crack size is evaluated for $\sqrt{2} \times$ (Pressure + NOP + Maximum Design) loads. For the margin on crack length evaluation, the J-integral for 2 times the leakage crack size is evaluated for (Pressure + NOP + Maximum Design) loads.

3.6.3.7 Stability Evaluation

The stability of the cracked pipes is assessed by comparing the J-integral value due to the applied loads on the pipe to the material crack resistance. The stability criterion for ductile crack extension employed is:

if $J_{\text{applied}} < J_{IC \text{ material}}$, and $(dJ/da)_{\text{applied}} < (dJ/da)_{\text{material}}$

then crack stability is assured.

The change in J-integral with crack length "a" is determined by analyzing several crack lengths in the region of interest. For a leakage crack of length "a", crack lengths "a", $a-\delta$, and $a+\delta$ are analyzed. Similarly, the change in J-integral with crack length in the region of length "2a" is determined by analyzing cracks with lengths 2a, $2a-\delta$, and $2a+\delta$. This method provides the derivative information in the two regions of interest. The variation of J with crack length in the region of "a" and "2a" is plotted along with the material curve. Evaluation of the plots allows for direct verification of the stability criteria.

The evaluations are performed for the locations chosen to envelop all limiting cases. The pipes with the leakage crack length subject to loads of $\sqrt{2} \times$ (P + NOP + Maximum Design Load) and the pipes with crack length twice the leakage crack length with loads of (P + NOP + Maximum Design Load) are demonstrated to have significant margin between the material curve and the loading curve, indicating that all pipe locations satisfy the LBB crack stability criteria.

See Appendix 3.9A, Sections 1.1.9.5.4 and 1.1.9.6 for a discussion of LBB design criteria development and a further discussion of analytical methods.

3.6.3.8 Results

The piping listed in Section 3.6.3 and evaluated by the methods described above are shown to meet all the *[[criteria for application of leak-before-break]]*² according to NUREG 1061, Volume 3. Specifically, these criteria *[[require the following:]]*²

- Cracks which are assumed to grow through the pipe wall leak significantly while remaining stable. The amount of *[[leakage is detectable with a safety margin of at least a factor of 10]]*² unless otherwise justified.
- *[[Cracks of the length that leak at the rate given above can withstand normal operation plus maximum design load loads with a safety factor of at least $\sqrt{2}$.]]*

INSERT 3.6.3.8

² NRC Staff approval is required prior to implementing change in this information; see DCD Introduction Section 3.5.

- *Cracks twice as long as those addressed above will remain stable when subjected to normal operation plus maximum design load.]]²*

Site specific evaluations will confirm that the bases for the LBB acceptance criteria are satisfied by the final as-built design and materials of the piping systems listed in Section 3.6.3 and will be documented in a LBB evaluation report.

References for Section 3.6

1. "Evaluation of Potential for Pipe Breaks," NUREG-1061, Vol. 3.
2. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, 2 or 3.
3. ASME Code for Pressure Piping, B31, Power Piping, ANSI/ASME B31.1.
4. USNRC Branch Technical Position MEB 3-1 Rev. 2 - Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment, attached to Standard Review Plan 3.6.2, June, 1987.
5. American National Standard Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture, ANSI/ANS 58.2-1988.
6. R. T. Lahey, Jr. and F. J. Moody, "Pipe Thrust and Jet Loads," The Thermal Hydraulics of a Boiling Water Nuclear Reactor, Section 9.2.3, pp. 375-409, Published by American Nuclear Society, Prepared by the Division of Technical Information, United States Energy Research and Development Administration, 1977.
7. RELAP 4/MOD 5, Computer Program User's Manual 098. 026-5.5.
8. USNRC Regulatory Guide 1.45 "Reactor Coolant Pressure Boundary Leakage Detection Systems."
9. Not used
10. NUREG/CR-2781, "Evaluation of Water Hammer Events in Light Water Reactor Plants," July 1982.
11. "Analysis of Cracked Pipe Weldments," EPRI NP-5057, February 1987.
12. USNRC Regulatory Guide 1.11 (Safety Guide 11), Instrument Lines Penetrating Primary Reactor Containment; including supplement, Backfitting Considerations.
13. Not used

² NRC Staff approval is required prior to implementing change in this information; see DCD Introduction Section 3.5.

In order to evaluate the derivative in the region of the leakage crack tip, three meshes are used. For a given leakage crack length "l" and model crack length a_1 , the three meshes have crack lengths a_1-d , a_1 , and a_1+d . The value d is a length appropriate to the anticipated amount of stable crack growth. This is indicated in Figure 3.9A-26. These three meshes are used in the analysis of the leakage crack. Similarly, three more meshes are generated for the analysis of twice the leakage crack lengths $2a_1-d$, $2a_1$, and $2a_1+d$.

For each load step in the analysis, the loading curve as a function of crack length is fit to a quadratic:

$$J(a) = C_1 a^2 + C_2 a + C_3$$

The values at a , $a \pm d$ provide the boundary conditions necessary to evaluate the constants C_1 , C_2 , and C_3 . At each loading point, the function is differentiated. This provides the dJ/da values for the loading curve. The material curves $J(a)$, $dJ(a)/da$ are evaluated at increasing crack extension. The loading functions $J(a)$, $dJ(a)/da$ are evaluated at either a_1 or $2a_1$, whichever crack length is being evaluated. Each point on the J vs. dJ/da loading curve corresponds to a different load state. As long as the loading curve stays below the material curve,

$$J_{LOAD} < J_{MAT} \quad \text{and}$$

$$dJ_{LOAD}/da < dJ_{MAT}/da$$

the crack growth is stable. For the case of increasing load, the loading curve will eventually intersect the material curve. At this point the crack will experience unstable crack growth. At this point of instability,

$$J_{LOAD} = J_{MAT} \quad \text{and}$$

$$dJ_{LOAD}/da = dJ_{MAT}/da$$

Development of the J versus dJ/da diagram for determining points of instability is shown in Figure 3.9A-27.

1.9.6.5 LBB Piping Evaluation Plots

1.9.6.5.1 Constructing an LBB Piping Evaluation Diagram

The method by which LBB Piping Evaluation Diagrams (PEDs) are constructed allows for the evaluation of the piping system in advance of the final piping analysis, incorporating LBB considerations into the piping design. The LBB PED for each specific pipe size, material and pressure is prepared prior to the piping design and analysis and is used to evaluate critical points in the pipeline. The PED is constructed to allow the maximum design load plus dead weight at the maximum design stress location to be plotted vs. the NOP load at the maximum design stress location.

The maximum design load at any time during the plant operation is the loading used in the stability analysis. Traditionally, this loading has been NOP+SSE. However, the combination of the NOP load and the largest of the design loads (i.e., the maximum design load) is used in the stability analysis (see Section 1.4 of this appendix). In the case of the surge line, for example, the line is evaluated for the larger of either NOP+SSE and for Stratified Flow (SF). For the discussion that follows, the maximum design load is considered to be the SSE load, and the loading combination is NOP+SSE.

The LBB piping evaluation plot requires performing two complete LBB evaluations. The evaluations are for two NOP loads which span the typical loadings for the line under consideration. A completed typical diagram is shown in Figure 3.9A-28. The procedure used for generating that figure is as follows:

1. Choose NOP = Pressure + NOP₁
2. Determine a₁
3. Increase the analysis moment until the critical moment is found for a₁ and 2a₁
4. Separate the critical analysis moment, M_c, into the correct addition of SSE and NOP₁ proportion for the a₁ and 2a₁ evaluations.

$$M_c = \sqrt{2} (NOP_1 + SSE_1) \quad (a_1 \text{ Analysis})$$

$$SSE_1 = \frac{M_c}{\sqrt{2}} - NOP_1 \quad \text{and}$$

$$M_c = (NOP_1 + SSE_1) \quad (2a_1 \text{ Analysis})$$

$$SSE_1 = M_c - NOP_1$$

5. Plot SSE values at NOP₁ for the a₁ and 2a₁ analyses, respectively. This corresponds to the points labeled "1" in Figure 3.9A-28.
6. Repeat steps (1) to (5) for NOP₂. The results are shown in Figure 3.9A-28, labeled "2".

Two stability evaluations are performed for each pipeline under consideration in order to complete the piping evaluation diagram.

When stratified flow (SF) is a critical thermal transient that must be considered in the stability analyses, the PED is constructed using the following relationships for the a₁ and 2a₁ evaluations in (4) above:

$$4. \quad M_c = \sqrt{2} SF_1 \quad (a_1 \text{ Analysis})$$

$$SF_1 = \frac{M_c}{\sqrt{2}} \quad \text{and}$$

$$M_c = SF_1 \quad (2a_1 \text{ Analysis})$$

$$SF_1 = M_c$$

As in the case for SSE, Step (4) is repeated to determine SF₂ for the a₁ and 2a₁ analyses, and SF is plotted vs. NOP.

INSERT
3.9.A.1

1.9.6.5.2 Using an LBB Piping Evaluation Diagram

Once the lines marking the acceptable areas of allowable piping loads are plotted as described in the previous section, normal operating piping loads and corresponding maximum design loads for the critical piping locations are plotted on the evaluation diagram. The critical locations are selected as the highest stressed point for each different type of material in the line. Figure 3.9A-29 shows how the plot is used for a hypothetical line. In this example, three points failed LBB and one point passed LBB. The reasons for each failure are given in the figure. The piping design can then be revised using the results (e.g., lowering the SSE response load by rerouting or by adding a snubber). Further review may result in other options for reducing the loads.

1.9.6.6 Results

Piping Evaluation Diagrams (PEDs) for piping systems listed in Section 3.6.3 are shown in Figures 3.9A-30 to 3.9A-36 and provide LBB acceptance criteria for these piping systems. These criteria are based on piping design parameters given in Table 3.9A-2. Analyses of preliminary design of these piping systems have demonstrated that the LBB criteria are met.

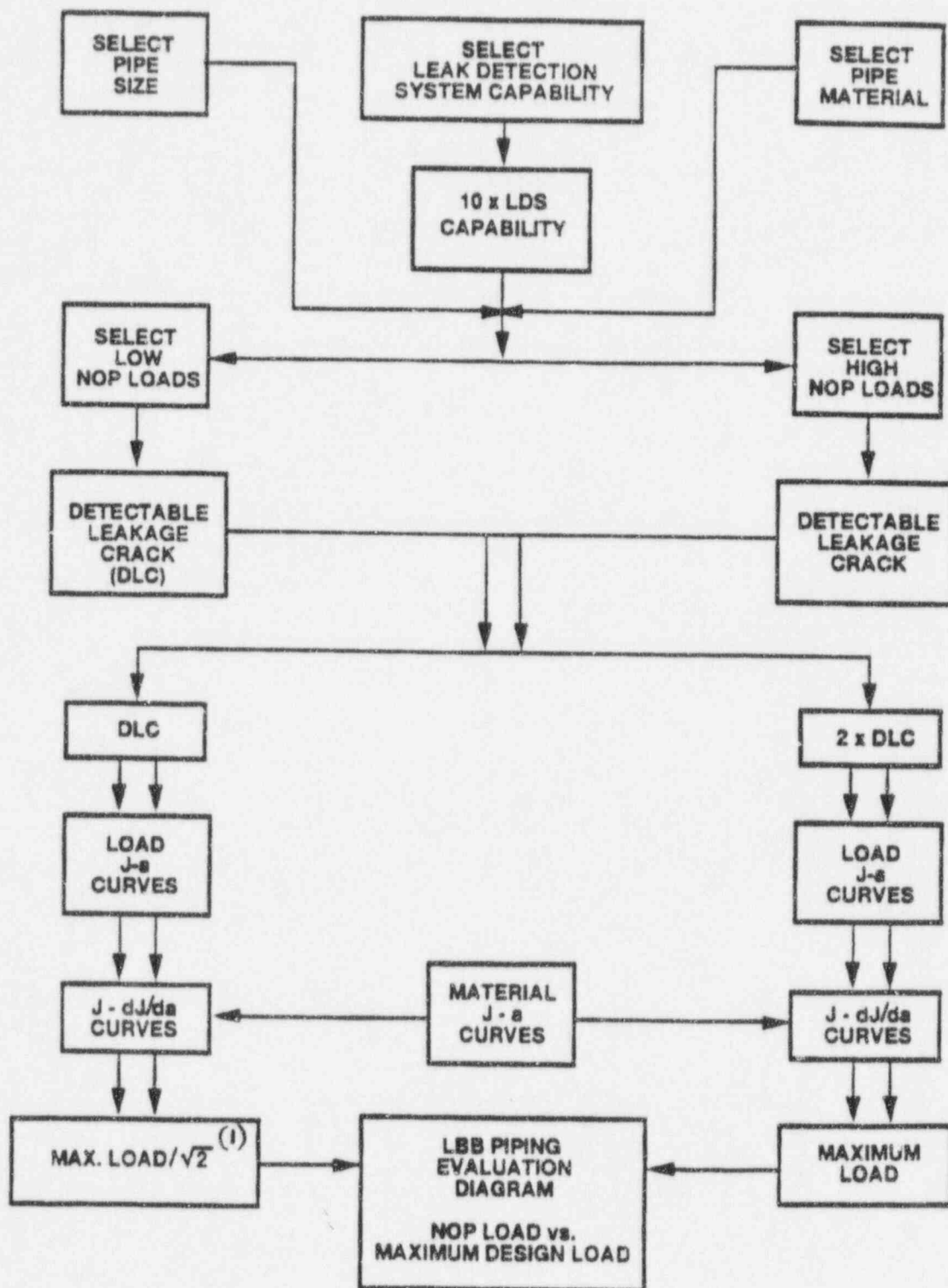
Site-specific information will demonstrate that the final detailed design parameters of each piping system are consistent with those given in Table 3.9A-2 and that the final detailed design meets the LBB criteria of Figures 3.9A-30 to 3.9A-36. If design parameters for a piping system are not enveloped by those in Table 3.9A-2, a new PED for that piping system will be constructed using the methodology given in this appendix and the piping design will be revised, as necessary, to meet the LBB criteria of the new PED. If a PED given in Figures 3.9A-30 to 3.9A-36 is applicable to the detailed design of a piping system but the detailed design does not meet the LBB criteria of the PED, the design will be revised to meet the LBB criteria of the PED.

Reconciliation of the as-built piping systems with the final design will be documented in a LBB Evaluation Report. The LBB Evaluation Report shall contain results of the LBB evaluations for as-built piping. The LBB evaluations shall employ methods described in Section 1.9 of this appendix. Reconciliation of each as-built piping system qualified for LBB will be made by demonstrating that:

1. the as-built piping system meets the screening criteria of Section 3.6.3,
2. the dimensional and material properties of the as-built piping system are consistent with the parameters used in the development of the final LBB PED(s) for that piping system, and
3. the as-built piping responses meet the ASME Code allowables and the final LBB PED criteria.

See also Sections 3.6 and 3.6.3.8.

INSERT
3.9.A.2



INSERT
3.9.A.3

LBB Design Criteria Development Diagram

Figure 3.9A-12

System 80+ DCD - Potential Design Changes

Item Number: 89

Summary Description: Eliminate Charging Pump Interlock (Add 160 gpm Flow Limit)

Affected DCD Sections: CDM: Section 2.7.16
ADM: Section 9.3.4

Change Description:

Reason for Change: The current System 80+ CVCS design includes an interlock in the charging pump controls so that both charging pumps cannot be operated at the same time. The interlock was added to limit the maximum charging flow within the upper bound assumed in boron dilution analyses.

During System 80+ detailed engineering, a concern was identified related to charging pump transfer from the operating to the standby pump. Normally, plant operations personnel start a standby pump, then shut down the running pump. This method of pump switching ensures continuity of charging flow, especially seal injection flow to the reactor coolant pumps. It also prevents thermal transients on the charging and letdown subsystems. Although the dedicated seal injection pump (DSIP) could be started during charging pump switching, the DSIP does not have sufficient capacity to provide charging flow also. During the transition period without charging flow, letdown flow may be reduced or isolated because the charging stream is not available to cool the letdown stream in the regenerative heat exchanger.

Description: The interlock that prohibits the operation of both charging pumps would be deleted. In its place, a commitment will be added to Tier 1 that the CVCS will limit charging flow to the RCS to 160 gpm. This flow limit will ensure that the maximum flow assumption in the boron dilution analyses is preserved and that the design flow for CVCS charging line components, typically 200 gpm, is not exceeded. The flow limit will most likely be implemented by adding a flow indicator controller and an isolation valve, in the combined charging pump discharge, which will close when a high flow rate is measured.

DCD Markups Attached? Yes

- All three tanks are located within a common dike structure designed to contain the maximum combined liquid inventory in the tanks.
- The dike structure will be designed to comply with applicable state and local regulations.

9.3.4.2 System Description

9.3.4.2.1 System

The normal reactor coolant flow path through the CVCS is indicated by the heavy lines on the flow diagrams (Figure 9.3.4-1, Sheets 1 through 4). Design parameters for the major components are shown in Table 9.3.4-4. Normal operating parameters for the CVCS are listed in Table 9.3.4-5. Process flow data is shown in Table 9.3.4-6.

Letdown flow from the RCS passes through the tube side of the regenerative heat exchanger where an initial temperature reduction takes place via heat transfer to cooler charging fluid on the shell side of the heat exchanger. The regenerative heat exchanger is designed to cool letdown flow to less than 450°F for all normal operations and to heat the charging flow by a minimum of 100°F. A final temperature reduction to the purification subsystem operating temperature is made by the letdown heat exchanger. The letdown heat exchanger is sized to cool inlet water from the maximum regenerative heat exchanger outlet temperature to 120°F (or lower) for most operating conditions. Both the letdown and the regenerative heat exchangers are designed for full RCS pressure and both are located inside containment.

Letdown fluid pressure is reduced from RCS pressure to the operating pressure of the purification subsystem in two stages. The first pressure reduction occurs at the letdown orifices and the second occurs at the letdown control valves located downstream of the orifices. The letdown orifices are located inside containment. The letdown orifices are sized to pass the maximum letdown flow at full RCS pressure with one control valve full open. The orifice provides the pressure reduction necessary to minimize erosion of the letdown control valve seating surfaces during normal RCS operations. A bypass valve around the orifices is provided for low pressure operations. The process flow is then filtered via the purification filter purified via a purification ion exchanger, and sprayed into the VCT. An excess hydrogen inventory is maintained in the RCS by keeping a hydrogen overpressure on the VCT contents.

The CVCS limits the maximum charging flow to the RCS to less than or equal to 160 gpm
The charging pumps normally take suction from the VCT and discharge to the RCS. During normal operations, one charging pump is running and the other is in standby. ~~An interlock is provided so that no more than one charging pump is operating at a time~~ during all modes of plant operation. One letdown and one charging pump flow control valve are normally selected for use. Seal injection water is supplied to the Reactor Coolant Pumps (RCPs) by diverting a portion of the charging flow just downstream of the charging pumps. This seal flow is then heated in the seal injection heat exchanger to approximately 125°F before filtering. Once the flow has been filtered, the seal injection fluid is distributed to the four RCPs. The undiverted charging fluid is sent to the regenerative heat exchanger where it is heated before injection into the RCS.

A chemical addition tank and a chemical addition metering pump are used to transfer chemical additives to the charging line downstream of the seal injection takeoff connection. Sufficient connections exist between the CVCS and the IRWST to allow for purification, inventory adjustments, and boron adjustments to the contents of that tank.

Analysis of inadvertent deboration events initiated during operational modes 2 through 6 (defined in the technical specifications) were performed. For Mode 1 operation, the reactivity addition due to a boron dilution event is less limiting than the CEA withdrawal events. These analyses show that Mode 5 (cold shutdown) in the drained down configuration results in the shortest available time for detection and termination of the event. Therefore, the initial conditions and analysis parameters are chosen for the cold shutdown operational mode to minimize the interval from initiation of dilution to the time at which criticality is reached. This results in the least amount of time between detection and criticality.

The following are the analysis assumptions for Mode 5:

1. The Technical Specification lower limit on shutdown margin for cold shutdown is assumed, 5.75% $\Delta\rho$.
2. The most adverse initial core condition would be for an initial K_{eff} corresponding to 5.75% $\Delta\rho$ subcritical and assuming subcriticality is maintained by boron concentration only.
3. The cold reactor coolant volume, including only the volumes for Mode 5 drained, is 3,961 ft³. A conservatively low reactor coolant mass was assumed by using the cold RCS internal volume. Assuming the coolant temperature of 210°F, the Technical Specification upper limit for cold shutdown, the resulting mass is 237,185 lbm.
4. ~~One charging pump is assumed to be running at its maximum rate, 150 gpm, and the other is isolated from the RCS. However, for conservatism the analysis used a charging pump flow rate of 160 gpm which corresponds to 22.26 lbm/sec.~~ ^{2 maximum}
to the RCS
5. The critical boron concentration with all rods in except the largest worth rod stuck out and the inverse boron worth are 814 ppm and 66 ppm/% $\Delta\rho$, respectively, including uncertainties. The initial boron concentration for the cold shutdown mode is found by adding the product of the inverse boron worth and the minimum shutdown margin (i.e. 5.75 percent) to the critical boron concentration. The resulting minimum initial boron concentration in Mode 5 is 1193 ppm. Thus, the change of boron concentration from 5.75% $\Delta\rho$ subcritical to critical is 379 ppm.

The parameters discussed above are summarized in Table 15.4.6-1.

● Results

Using the above conservative parameters in Equation (3), the minimum possible time interval to dilute from 5.75% $\Delta\rho$ subcritical to criticality is 67 minutes. Utilizing only the redundant, qualified neutron flux alarm, this time period will assure detection of a boron dilution event at least 30 minutes prior to criticality. Boron dilution will then be terminated before loss of shutdown margin by the operator actions discussed in Section 15.4.6.2.

15.4.6.4 Conclusions

The inadvertent deboration event will result in acceptable consequences. Sufficient time is available for the operator to detect and to terminate an inadvertent deboration event if it occurs. Fuel integrity is not challenged during this event.

Table 15.4.6-1 Assumptions for the Inadvertent Deboration Analysis

Parameter	Assumptions
Cold RCS Volume ⁽¹⁾ (mid-loop operation), ft ³	3,961
RCS Mass (mid-loop operation), lbm	237,185
Volumetric Charging Rate, gpm	160
Mass Charging Rate, lbm/sec	22.3
Dilution Time Constant, τ , sec ⁻¹	10650
Initial Boron Concentration - C_0 , ppm	1193
Critical Boron Concentration - C_{crit} , ppm	814

⁽¹⁾ Includes the reactor vessel up to the mid-plane of the hot legs, half of a single hot leg, half of two cold discharge legs and a shutdown cooling system.

Table 2.7.16-1 Chemical and Volume Control System (Continued)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Valves with response positions indicated on Figure 2.7.16-1 change position to that indicated on the Figure upon loss of motive power.	5. Testing of loss of motive power to these valves will be performed.	5. These valves change position to the position indicated on Figure 2.7.16-1 on loss of motive power.
6.a) The letdown line is isolated by a safety injection actuation signal (SIAS).	6.a) Testing will be performed using a signal simulating an SIAS.	6.a) The two CVCS letdown isolation valves inside containment close upon receipt of a signal simulating an SIAS.
6.b) The RCP seal controlled bleedoff line is isolated by a containment spray actuation signal (CSAS).	6.b) Testing will be performed using a signal simulating a CSAS.	6.b) The RCP seal controlled bleedoff line isolation valves close upon receipt of a signal simulating a CSAS.
7. An interlock is provided so that no more than one charging pump is operating at a time. <i>The CVCS limits the maximum charging flow to the RCS.</i>	7. Testing will be performed by attempting to start each charging pump from the MCR with the other pump running with the RCS at 0.15 gpm. <i>operating both</i>	7. The idle charging pump will not start when the other pump is running. <i>The CVCS maximum charging flow is less than or equal to 160 gpm.</i>
8. Motor operated valves (MOV) having an active safety function will open, or will close, or will open and also close, under differential pressure or fluid flow conditions and under temperature conditions.	8. Testing will be performed to open, or close, or open and also close, MOVs having an active safety function under preoperational differential pressure or fluid flow conditions and under temperature conditions.	8. Each MOV having an active safety function opens, or closes, or opens and also closes.
9. Check valves shown on Figure 2.7.16-1 will open, or will close, or will open and also close, under system pressure, fluid flow conditions, or temperature conditions.	9. Testing will be performed to open, or close, or open and also close, check valves shown on Figure 2.7.16-1 under system preoperational pressure, fluid flow conditions, or temperature conditions.	9. Each check valve shown on Figure 2.7.16-1, opens, or closes, or opens and also closes.
10. Flow limiting orifices are provided in the letdown line.	10. Inspection of the as-built letdown orifices will be performed.	10. Each letdown line flow limiting orifice has a cross-sectional area not greater than 0.01556 square feet.

The letdown line is isolated by a safety injection actuation signal (SIAS). The RCP controlled bleedoff line is isolated by a containment spray actuation signal (CSAS).

~~An interlock is provided so that no more than one charging pump is operating at a time.~~
The CVCS lim. is the maximum charging flow to the RCS.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.16-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Chemical and Volume Control System.

¹⁰
Item H: Increased Damping for Response Spectrum Piping Analyses

DCD Section: 3.7, Table 3.7-1, Figure 3.7-32

Description of Change:

The NRC has approved a change in the maximum allowable value of applied damping for piping analyses in which the uniform envelope response spectrum analysis method is used. The maximum allowable damping in the System 80+ DCD for this type of analysis is currently based on ASME Code Case N-411-1, which allows the damping to vary from 5% for modes of vibration up to and including 10 Hz to 2% for modes above 20 Hz. The NRC approved change allows 5% damping for all modes of vibration when the uniform envelope response spectrum analysis method is used on piping. This change was approved by the NRC staff on another ALWR design.

Damping values from Table 3.7-1, based on piping diameters ≤ 12 inches and >12 inches, continue to be used when piping is analyzed by time history or multiple support input methods.

Related Changes to DCD

DCD Section	Insert #	Revision
-------------	----------	----------

3.7.1.3		Revise as marked
---------	--	------------------

Figure 3.7-32		Delete
---------------	--	--------

Table 3.7-1		Add entry for 5% damping, delete Note [2]; renumber Note [3] to [2]. Last 3 entries to Table 3.7.-1 should read as follows:
-------------	--	---

<i>[[Piping (diameter ≤ 12 inches)</i>	<i>2.0</i>
<i>Piping (diameter > 12 inches)</i>	<i>3.0</i>
<i>Piping (uniform envelope response spectrum method of analysis)</i>	<i>5.0]]</i> ^[2]

^[2] NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

3.7.1.2 Design Time History

Since the System 80+ Standard Design is designed for generic site conditions, for the time history method of analysis, the generic free-field ground surface time histories are used as control motions in the analyses. In the soil-structure interaction analyses, for each generic site, the corresponding two horizontal and one vertical time histories at the free-field ground surface are used with the SSI model of that site. For the fixed-base analyses, the rock outcrop time histories are directly used as the control time histories.

The response spectra at 2, 5 and 7% damping of control motion CMS1, and 1, 2, 5 and 7% damping of control motions CMS2 and CMS3 and the corresponding spectral ordinates of the matching time histories are shown in Figures 3.7-1 to 3.7-12. The Power Spectral Densities of all time histories are included in Section 2.5.

Each time history that is used in the SSI and rock analyses contains 20.48 seconds. For the SSI analyses, a time step of 0.005 seconds is used. For the Nuclear Island rock analyses, a time step of 0.0025 seconds is used.

For Category I structures not on the Nuclear Island a time step of 0.005 seconds is used for both SSI and rock analyses.

3.7.1.3 Critical Damping Values

alternative requirements for uniform envelope spectrum response piping analysis.

Damping values used for various nuclear safety-related structures systems and components are based upon Regulatory Guide 1.61 or ~~ASME Code Case N-411-1 (See Figure 3.7-22)~~. These values are expressed in percent of critical damping and are given in Table 3.7-1. ~~When the response spectra method of analysis is used for piping, damping values are based on Code Case N-411-1.~~

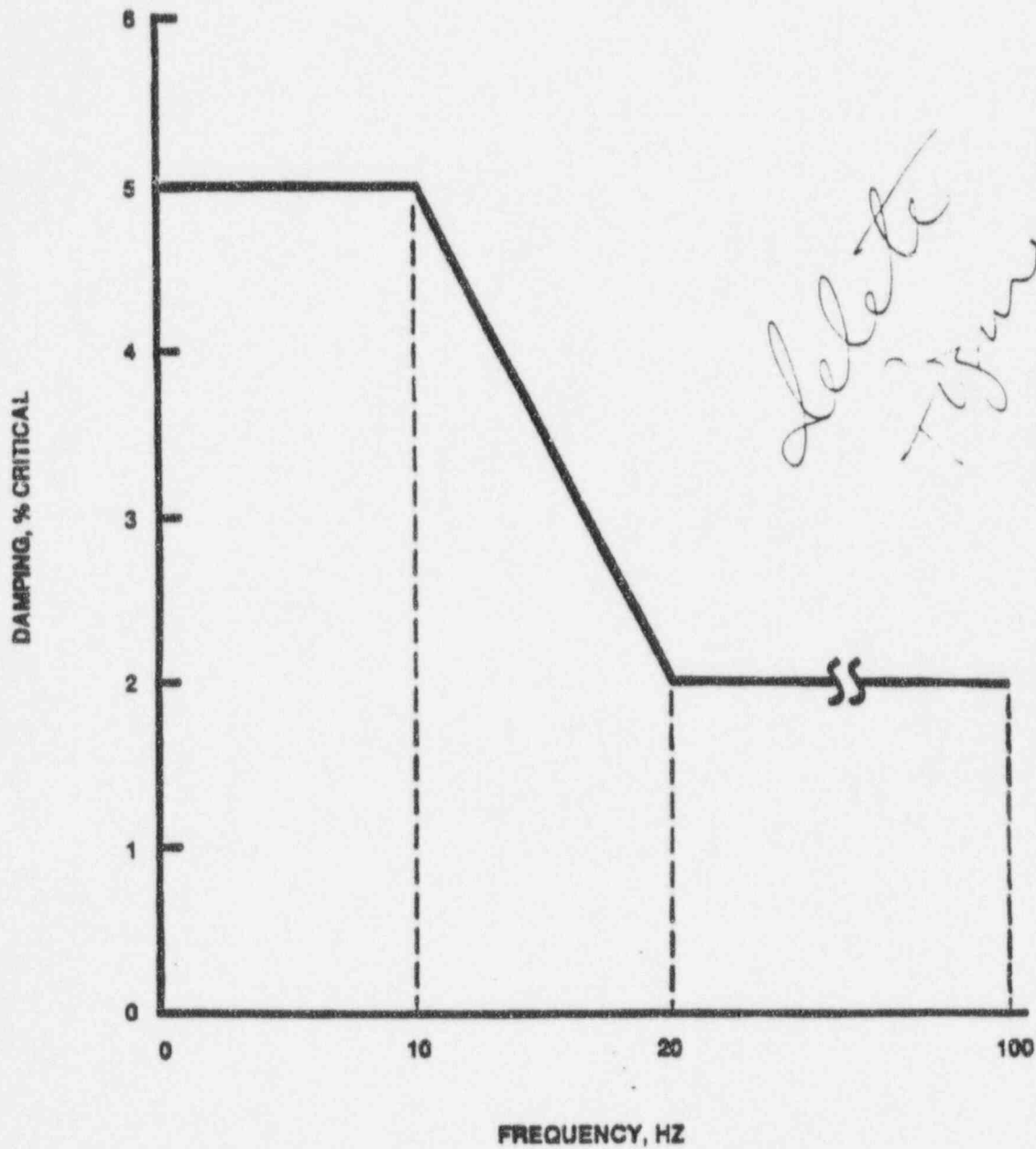
3.7.1.4 Supporting Media for Seismic Category I Structures

Category I structures are founded directly on rock or competent soil. For the Nuclear Island the foundation embedment depth for System 80+ standard plant is approximately 51 feet (Reference 7). The rock properties and the layering characteristics, including shear wave velocity, shear modulus, and density, are given in Section 2.5. The System 80+ Nuclear Island is designed for the range of soil conditions discussed in Section 2.5 and shown in Appendix 3.7B.

3.7.1.4.1 Soil Structure Interaction (SSI)

Two different types of analysis methodologies are used for the seismic analyses for the Nuclear Island. For the fixed-base cases, modal superposition time history analyses are performed using the three control motions (CMS1, CMS2, and CMS3) corresponding to rock site conditions. When a structure is supported on soil, the SSI is taken into account by coupling the structural model with the soil medium. To accomplish this, the methodology of the computer program SASSI (System for Analysis of Soil Structure Interaction, Reference 6) is used. Detailed methodology and results of the SSI analysis for the Nuclear Island are presented in Appendix 3.7B.

The methodology for the soil structure interaction for the non-Nuclear Island structures is presented in Appendix 3.7C.



GENERAL NOTE: DAMPING INDEPENDENT OF PIPE DIAMETER.

Table 3.6-3 High-Energy Lines Within Containment (Cont'd.)

Item No.	System	Line Functional Description	Operating Pressure (> 275 psig)	Operating Temperature (> 200°F)	Figure
38	Reactor Coolant	SG No. 2 RCS Loop 2B Drain Line to RDT (High-Energy to Isolation Valve RC-332)	Yes	Yes	5.1.2-1
39	Reactor Coolant	SG No. 2 RCS Hot Leg Drain Line to RDT (High-Energy to Isolation Valve RC-215)	Yes	Yes	5.1.2-1
40	Reactor Coolant	Pressurizer Relief Line #1 to Pressurizer Safety Valve RC-200	Yes	Yes	5.1.2-3
41	Reactor Coolant	Pressurizer Relief Line #2 to Pressurizer Safety Valve RC-201	Yes	Yes	5.1.2-3
42	Reactor Coolant	Pressurizer Relief Line #3 to Pressurizer Safety Valve RC-202	Yes	Yes	5.1.2-3
43	Reactor Coolant	Pressurizer Relief Line #4 to Pressurizer Safety Valve RC-203	Yes	Yes	5.1.2-3
44	Reactor Coolant	Pressurizer Spray Line from Cold Leg Loop 1A to 1A Spray Control Valve RC-100E	Yes	Yes	5.1.2-1 & 5.1.2-3
45	Reactor Coolant	Pressurizer Spray Line from Cold Leg Loop 1B to 1B Spray Control Valve RC-100F	Yes	Yes	5.1.2-1 & 5.1.2-3
46	Reactor Coolant	Pressurizer Spray Line from Loop 1A Spray Control Valve RC-100E to Pressurizer Spray Common Header	Yes	Yes	5.1.2-3

Since PSVs are mounted directly on P312.

Table 3.6-3 High-Energy Lines Within Containment (Cont'd.)

Item No.	System	Line Functional Description	Operating Pressure (> 275 psig)	Operating Temperature (> 200°F)	Figure
57	Reactor Coolant	Direct Vessel Injection Connection #4 to SIS Interior Check Valve SI-217	Yes	Yes	5.1.2-1 & 6.3.2-1C
58	Safety Depressurization System	Branch-Off of Pressurizer Safety Valve RC-201 Steam Line to RC-409 (Rapid Depress. Line).	Yes	Yes	5.1.2-3
59	Safety Depressurization System	Branch-Off of Pressurizer Safety Valve RC-203 Steam Line to RC-408 (Rapid Depress. Line).	Yes	Yes	5.1.2-3
60	CVCS	Letdown Line from Loop 2B to Regenerative Hx	Yes	Yes	9.3.4-1, 5.1.2-1
61	CVCS	Letdown Line from Regenerative Hx to Letdown Hx	Yes	Yes	9.3.4-1
62	CVCS	Letdown Line from Letdown Hx to Containment Penetration	Yes	No	9.3.4-1
63	CVCS	Charging Line from Containment Pen to Regenerative Hx	Yes	No	9.3.4-1
64	CVCS	Charging Line from Regenerative Hx to RCS Loop 2A	Yes	Yes	9.3.4-1
65	CVCS	Auxiliary Spray Line to Pressurizer Spray Common Header	Yes	Yes	9.3.4-1, 5.1.2-3
66	CVCS	SCS Hx Shutdown Purification Line Cont Pen Check Valve CH-304 to Letdown Hx	Yes	Yes	9.3.4-1

Table 3.9-2 Loading Combinations ^{for} ASME Code Class 1, 2, and 3 Components ^{and Component supports} ^[3]

Condition	Design Loading ^[1] Combination
Design	PD + DW
Level A (Normal) ^[2]	PO + DW
Level B (Upset) ^[2]	PO + DW
Level C (Emergency)	PO + DW + DE
Level D (Faulted)	PO + DW + SSE + DF

add ops, each side of " + "

[1] Legend:

PD = Design pressure

PO = Operating pressure

DW = Dead weight

SSE = Safe Shutdown Earthquake

DE = Dynamic system loadings associated with the emergency condition

DF = Dynamic system loadings associated with pipe breaks (not eliminated by leak before break analysis)

[2] As required by ASME Code Section III, other loads, such as thermal transient, and thermal gradient require consideration in addition to the primary stress producing loads listed. SSE is considered in equipment fatigue evaluations in accordance with Section 3.7.3.2.

[3] For piping, see Tables 3.9-10 and 3.9-11.

Table 4.2-3 C-E Poolside Fuel Inspection Program Summary (Reference 70) (Cont'd.)

Reactor	Shutdown Date/Cycle	Inspection Program Scope[1]
Yankee Rowe	1987/18	VE, UT, SRE
Millstone-2	1977/1	VE
	1982/4	VE
St. Lucie-2	1987/3	VE, UT
	1989/4	VE, UT, SRE
ANO-2	1981/1	VE, DM, SRE on C-E/EPRI Test Bundles
	1982/2	VE, DM
	1983/3	VE, DM, SRE on C-E/EPRI and DOE Test Bundles
	1985/4	VE, DM, SRE on DOE Test Bundles
	1986/5	VE, DM, UT
	1988/6	VE, DM, SRE on DOE Test Bundles
	1989/7	VE, UT, SRE
San Onofre-2	1984/1	VE, DM
	1985/2	VE, DM
	1987/3	VE, UT, GS, SRE
	1989/4	VE, UT, SRE, DM
San Onofre-3	1985/1	VE, UT
	1988/3	VE, UT, SRE
Palo Verde-1	1987/1	VE, DM
	1989/2	VE, DM
Palo Verde-2	1988/1	VE, DM
Waterford-3	1988/1-2	VE, UT, SRE

- (1) VE Visual Examination
 GS Gamma-Scanning
 CS Crud Sampling
 S Sipping
 UT Ultrasonic Testing
 SRE Disassembly and Single Rod Examinations
 DM Dimensional Measurements

Table 5.4.7-2 Shutdown Cooling System Failure Modes and Effects Analysis

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
1)	Shutdown Cooling Pump Suction Isolation Valve SI-106, SI-107	Fails Closed	Corrosion, mechanical binding, operator error	Effective loss of one shutdown cooling train cooling	Low flow indication F-302, F-305; periodic testing	Parallel redundant shutdown cooling path	Valve is normally locked open
		Fails Open	Same as 1a)	No effect on SCS operation	Periodic testing	None required	
2)	Shutdown Cooling Pump 1, 2	Fails to start	Mechanical failure, electrical failure	Effective loss of one SCS train	Low flow indication F-302, F-305; periodic testing	Parallel redundant shutdown cooling path	
3)	Shutdown Cooling Pump Discharge Isolation Valve SI-578, SI-579	Fails Open <i>closed</i>	Corrosion, mechanical binding, operator error	Effective loss of one shutdown cooling pump	Low flow indication F-302, F-305; periodic testing	Parallel redundant shutdown cooling path	Valves are locked open; min. flow line will provide the min. flow required to protect the pump
		Fails Open	Same as 3a)	No effect on SCS operation	Periodic testing	None required	
4)	Shutdown Cooling Heat Exchanger 1,2	Loss of Cooling	Insufficient component cooling water flow, excessive fouling	Diminished ability of subsystem to provide RCS heat removal	High temperature indication from T-302, T-305	Parallel redundant shutdown cooling path	
5)	Shutdown Cooling Heat Exchanger Bypass Valve SI-312, SI-313	Fails Closed	Corrosion, mechanical binding, electrical failure	Delays use of affected SCS train	Valve position indicator; periodic testing	Parallel redundant shutdown cooling path	Same as 3a)
		Fails Open	Mechanical failure, electrical failure	Effective loss of one shutdown cooling path	Valve position indicator; periodic testing	Parallel redundant shutdown cooling path	

Item 14

The boron recovery portion of the CVCS accepts letdown flow diverted from the VCT as a result of feed and bleed operations for shutdowns, startups, and boron dilution over core life. The diverted letdown flow, which has passed through a purification filter and ion exchanger, also passes through the pre-holdup ion exchanger. The pre-holdup ion exchanger retains cesium, lithium, and other ionic radionuclides with high efficiency. The process flow then passes through the gas stripper, where hydrogen and fission gases are removed with high efficiency; thus (1) precluding the buildup of explosive gas mixtures in the holdup tank and (2) minimizing the release of radioactive fission product gases in aerated vents or liquid discharges. The degassed liquid is automatically pumped from the gas stripper to the holdup tank.

Reactor coolant quality water from valve and equipment leakoffs, drains, and reliefs within the containment is collected in the Reactor Drain Tank (RDT) and scheduled for batch processing. Recoverable reactor coolant quality water outside the containment from various equipment and valve leakoffs, reliefs, and drains is collected in the Equipment Drain Tank (EDT) and scheduled for batch processing. Reactor coolant collected in either of these tanks is periodically discharged by the reactor drain pumps through the reactor drain filter and pre-holdup ion exchanger, and processed in the same manner as diverted VCT flow, as described above. This liquid is also pumped to the holdup tank.

When a sufficient volume accumulates in the holdup tank, it is pumped by a holdup pump to the boric acid concentrator, where the bottoms are concentrated to within the range of 4000 to 4400 ppm boron. The boric acid concentrator bottoms are continuously monitored for proper boron concentration, and normally pumped directly to the BAST. In the event that abnormal quantities of radionuclides are present, the bottoms are discharged to the LWMS. The boric acid concentrator distillate passes through a boric acid condensate ion exchanger, where boric acid carryover is removed. The distillate is collected in the RMWT for reuse in the plant. If recycle is not desired, the distillate is diverted to the LWMS.

When the SCS is operational, a flow path through the CVCS can be established for purification. This is accomplished by diverting a portion of the flow from the shutdown cooling heat exchanger to the letdown line upstream of the letdown heat exchanger. The flow then passes through the purification filter, purification ion exchanger, and letdown strainer, and is returned to the suction of the shutdown cooling pumps.

When continuous degasification of the RCS is desired, fluid is returned to the inlet of the VCT to the gas stripper, bypassing the pre-holdup ion exchanger. The flow then passes through the gas stripper and is then returned to the VCT via the letdown line. The overpressure can be used to replace the hydrogen charging pumps. The VCT hydrogen charging process. The VCT hydrogen charging process. The VCT hydrogen charging process.

A makeup subsystem of the CVCS provides for changes in RCS boron concentration. Boron is initially added to the CVCS using the boric acid batching tank (BABT). Reactor makeup water is added to the BABT via the makeup supply header, and the fluid is heated by immersion heaters. Boric acid powder is added to the heated fluid while a mixer agitates the fluid. A boric acid concentration of as high as 12 weight percent can be prepared. Electric immersion heaters maintain the temperature of the solution in the boric acid batching tank high enough to preclude precipitation. The concentrated boric acid solution in the BABT is drawn into the boric acid batching eductor and diluted by fluid being circulated from the BAST via the boric acid makeup pumps. The reactor makeup water pumps can also be used by taking suction from the reactor makeup water tank and pumping the water through the eductor to the BAST.

The resulting concentration of the refueling pool and the RCS is between the lower operating boron concentration limitation of the IRWST (4000 ppm) and the maximum operating boron concentration of the IRWST (4400 ppm). Thus, the contents of the refueling pool can be returned directly to the IRWST prior to plant startup without hindering plant operations.

During refueling shutdowns, the reactor makeup water supply piping is continuously monitored via flow switch F-250. An alarm is annunciated if flow is detected in order to prevent dilution of the refueling pool.

9.3.4.3 Design Evaluation

9.3.4.3.1 Availability and Reliability

A high degree of functional reliability is assured by providing standby components and by assuring fail-safe responses for the most probable modes of failure.

Redundancy is provided as follows:

<u>Component</u>	<u>Redundancy</u>
Purification and Deborating Ion Exchangers	Three identical components
Charging Pumps	One operating, one in standby
Charging Pump Flow Control Valves	One operating and one parallel, standby valve
Letdown Control Valves	One operating and one parallel, standby valve
Boric Acid Makeup Pumps	Two identical pumps in parallel, one operates on demand, one in standby
Gas Stripper Package	The gas stripper package includes redundant standby pumps
Seal Injection Filters	Two identical ^{filters} pumps in parallel, one ^{operational,} operates on demand, one in standby
Purification Filters	Two identical ^{filters} pumps in parallel, one ^{operational,} operates on demand, one in standby
Reactor Makeup Water Pumps	Two identical pumps in parallel, one operates on demand, one in standby
Boric Acid Concentrator	The concentrator package includes redundant standby pumps

In addition to component redundancy, it is possible to operate the CVCS in a manner such that some components are bypassed. It is possible to transfer boric acid to the charging pump suction header by bypassing the VCT. The letdown filter, and the purification and deborating ion exchangers can be bypassed. Controlled bleedoff flow can be routed to the RDT rather than the VCT.

Independent and redundant gravity feed lines from the BAST to the charging pump suction are provided to assure makeup. The charging pumps also have an alternate source of borated water from the spent fuel pool, which is maintained above 4000 ppm boron.

Table 9.3.4-4 Principal Component Data Summary (Cont'd.)

Reactor Makeup Water Pumps	
Quantity	2
Type	Centrifugal
Design pressure	200 psig
Design temperature	200°F
Rated head	300 ft.
Normal flow	180 gpm
Normal operating temperature	40-120°F
NPSH required	15 ft
Material in contact with pumped fluid	Austenitic stainless steel
Fluid	Demineralized water
Code	None
Holdup Pumps	
Quantity	2
Type	Centrifugal
Design pressure	100 psig
Design temperature	200°F
Rated head	145 ft
Normal flow	50 gpm
Normal operating temperature	40-120°F
NPSH required	10 ft
Materials in contact with pumped fluid	Austenitic stainless steel
Fluid	2.5 wt % boric acid, maximum
Code	None
Reactor Drain Pumps	
Quantity	2
Type	Centrifugal
Design pressure	200 psig
Design temperature	200°F
Rated head	145 ft
Normal flow	50 gpm
Normal operating temperature	120°F
NPSH required	10 ft
Materials in contact with pumped fluid	Austenitic stainless steel
Fluid	2.5 wt % boric acid, maximum
Code for fluid end	ASME III, Class 3
Volume Control Tank	
Quantity	1
Type	Vertical, cylindrical
Internal volume	5,800 gallons (approx)
Design pressure, internal	75 psig
Design pressure, external	15 psig
Normal operating temperature	120°F
Normal operating pressure	20 psig 20-50 psig

- Adequate clearances shall be provided for inservice inspection of the ASME Boiler and Pressure Vessel Code Section III, Class 2 portions of the main steam system piping, in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.
- Loop seals are not utilized in safety valve inlets.

10.3.2.3.2 Valves

10.3.2.3.2.1 Main Steam Isolation Valves (MSIVs) and MSIV Bypass Valves

- The valves are designed so that no damage due to excessive closure force is incurred during closure under design conditions.
- Backseating of valve stems is provided when the valve is in the full open position.
- Unrecovered pressure loss from valve inlet to valve outlet at rated flow with the valve full open does not exceed 3 psid.
- The Main Steam Isolation Valve (MSIV) in each main steam line is remotely operated and is capable of maintaining tight shutoff under the main steam line pressure, temperature and flow resulting from the transient conditions associated with a pipe break in either direction of the valves.
- The MSIV leak rate through a closed MSIV flow does not exceed 0.001% of nominal flow at 1200 psia in the forward direction and does not exceed 0.1% of nominal flow at 1200 psia in the reverse direction.
- The full open to close stroke time of the MSIV's under fully developed steam line break flow is 5 seconds or less upon receipt of a Main Steam Isolation Signal (MSIS). The full open to close stroke time of the MSIV's Bypass Valves is ~~10~~₅ seconds or less upon receipt of a MSIS.
- The MSIVs are supported such that the valve body and actuator will not be distorted to such a degree that the valve cannot close or be displaced as a result of pipe break thrust loadings.
- The MSIVs and the MSIV bypass valves are designed, fabricated and installed such that the requirements for In-service Testing and Inspection of ASME Section XI, Subsection IWV can be met.
- The provisions of General Design Criteria 57 for containment isolation valves are met.
- The MSIV is a fail close valve; upon receipt of a Main Steam Isolation Signal the MSIV closes automatically.
- The MSIV bypass valve is a fail-close valve; upon receipt of a Main Steam Isolation Signal the MSIV bypass valve closes automatically.
- The MSIVs and their supports and the MSIV bypass valves and their supports are designed to withstand loads arising from the various operating and design bases events as specified in Section 3.9.3.

(18)

- To permit testing for pH and the existence of foreign substances, sample connections are provided in the steam line piping between the steam generator nozzles and equalization header.
- During initial startup and during periods of unit shutdown, the tripping mechanisms for the main steam isolation valves are tested for proper operation in accordance with the technical specifications. The valves are periodically in-service tested for leakage and freedom of movement during plant operation in accordance with ASME Code Section XI, Subsection IWV.
- The main steam safety valves are tested during initial startup or during shutdown operation by checking the actual lift and closing pressures of the valves in comparison to the required design opening and closing pressures in accordance with ASME Code, Section XI, Subsection IWV.
- ASME Code Section XI, Subsection IWV requirements for in-service testing and inspection of nuclear safety-related valves apply to the atmospheric dump and atmospheric dump isolation valves.
- A test will be conducted to verify MSIV response to a simulated Main Steam Isolation Signal (MSIS).
 1. The objective of the test is to verify the function of the MSIVs and to confirm the 5 second closing time required by Section 10.3.2.3.2.1.
 2. Since steam pressure is normally required to operate the MSIVs, a supply of steam at conditions comparable to main steam is a prerequisite, in addition to the completion of construction activities on the MSIVs and required support systems.
 3. The test method consists of the application of a simulated MSIS to the controls of the MSIV under test, the recording of temperature and pressure parameters upstream and downstream of the valve seat, and the timing of the closure process from the receipt of signal to the instance of valve closure as indicated by the valve stem travel indicator.
 4. Acceptance criteria are that the MSIV operate to shut in 5 seconds or less, in accordance with Section 10.3.2.3. The test must be related by calculation and manufacturer's shop or type testing to the design basis conditions required by the safety function.
- A test will be conducted to verify MSIV bypass valve response to a simulated Main Steam Isolation Signal (MSIS):
 1. The objective of the test is to verify the function of the MSIV bypass valves and to confirm the ⁵10 second closing time required by Section 10.3.2.3.2.1.
 2. Construction activities on the MSIV bypass valves and their required support systems must be complete as a prerequisite.
 3. The test method consists of the application of a simulated MSIS to the controls of the MSIV bypass valve under test, the recording of temperature and pressure parameters upstream and downstream of the valve seat, and the timing of the closure process from receipt of signal to the instance of valve closure as indicated by the valve stem travel indicator.

4. Acceptance criteria are that the MSIV bypass valves operate to shut in 10 seconds or less, in accordance with Section 10.3.2.3. The test must be related by calculation and manufacturer's shop or type testing to the design basis conditions required by the safety function.

10.3.5 Secondary Water Chemistry

10.3.5.1 Chemistry Control Basis

Steam generator secondary side water chemistry control is accomplished by:

- Close control of the feedwater to limit the amount of impurities which can be introduced into the steam generator.
- Continuous blowdown of the steam generator to reduce the concentrating effects of the steam generator.
- Chemical addition to establish and maintain an environment which minimizes system corrosion.
- Pre-operational cleaning of the feedwater system.
- Minimizing feedwater oxygen content prior to entry into the steam generator.

Secondary water chemistry is based on the zero solids treatment method. This method employs the use of volatile additives to maintain system pH and to scavenge dissolved oxygen which may be present in the feedwater.

A neutralizing amine is added to establish and maintain alkaline conditions in the feedtrain. Neutralizing amines which can be used for pH control are ammonia and morpholine. Ammonia should be used in plants employing condensate polishing to avoid resin fouling. Although the amines are volatile and will not concentrate in the steam generator, they will reach an equilibrium level which will establish an alkaline condition in the steam generator.

Hydrazine is added to scavenge dissolved oxygen which may be present in the feedwater. Hydrazine also tends to promote the formation of a protective oxide layer on metal surfaces by keeping these layers in a reduced chemical state.

Both the pH agent and hydrazine can be injected continuously at the discharge headers of the condensate pumps or condensate demineralizer, if installed. These chemicals are added as necessary for chemistry control, and can also be added to the upper steam generator feed line when necessary.

Operating chemistry limits for secondary-side steam generator water, feedwater and condensate as extracted from Reference 2 EPRI Report NP-6239, are given in Tables 10.3.5-1, 10.3.5-2 and 10.3.5-3.

The limits stated are divided into three groups: normal, abnormal and immediate shutdown. The limits provide high quality chemistry control and yet permit operating flexibility. The normal chemistry conditions can be maintained by any plant operating with little or no condenser leakage. The abnormal steam generator limits are suggested to permit operations with minor system fault conditions until the affected component can be isolated and/or repaired. The immediate shutdown limits represent chemistry conditions at which continued operation could result in severe steam generator corrosion damage.

Table 19.7.5.1-1 Components in Seismic Fault Tree Models

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NAME	DESC	PROB	HCLPE
Initiator and Special Events			
EQ-MTC	Adverse Moderator Temperature Coefficient (MTC)	1.00E-02	
EQ-PSV	Primary Safety Valve (PSV) Fails to Reseat		1.35
EQ-PSVO	Primary Safety Valves (PSVs) Fail to Open		1.35
EQATWS	Seismically Induced ATWS		0.74
EQLOSP	Seismically Induced Loss of Site Power		0.12
EQMLOCA	Seismically Induced Medium LOCA		2.59
EQRVR	Seismically Induced LOCA in Excess of ECCS Capacity		0.73
EQSLOCA	Seismically Induced Small LOCA		0.90
EQTRANS	Seismically Induced Transient		0.05
Emergency Feedwater System (EFWS)			
ABDZEFWPBRKR	Seismically Induced Failure of EFW Motor Pump Circuit Breaker		0.95
AHFFASCSLOCA	Operator Fails to Perform Aggressive Secondary Cooldown for Small LOCA	6.40E-02	
AKPZEFWSCABLE	Seismically Induced Failure of EFWS Power Cable		1.80
APAJEFWP-102	Motor-Driven Emergency Pump EFWP-102 Fails to Start	3.00E-03	
APAJEFWP-104	Motor-Driven Emergency Pump EFWP-104 Fails to Start	3.00E-03	
APAKEFWP-102	Motor-Driven Emergency Pump EFWP-102 Fails to Run	3.60E-03	
APAKEFWP-104	Motor-Driven Emergency Pump EFWP-104 Fails to Run	3.60E-03	
APAVEFWP-102	Subtrain 1B Unavailable Due to Maintenance	2.00E-03	
APAVEFWP-104	Subtrain 2B Unavailable Due to Maintenance	2.00E-03	
APTJEFWP-101	EFW Turbine Driven Pump EFWP-101 Fails to Start	1.50E-02	
APTJEFWP-103	EFW Turbine Driven Pump EFWP-103 Fails to Start	1.50E-02	
APTKEFWP-101	EFW Turbine Driven Pump EFWP-101 Fails to Operate	7.20E-03	
APTKEFWP-103	EFW Turbine Driven Pump EFWP-103 Fails to Operate	7.20E-03	
APTVEFWP-101	Subtrain 1A Unavailable Due to Maintenance	5.00E-03	
APTVEFWP-103	Subtrain 2A Unavailable Due to Maintenance	5.00E-03	
APTXDP101-103	Common Cause Demand Failure of EFW Turbine Pumps	1.19E-03	
APTZEFWP	Seismically Induced Failure of Turbine-Driven EFW Pump		2.38
AQPZEFWSPiPE	Seismically Induced Failure of EFWS Piping		0.90
AQWZEFWDGROOM	Seismically Induced Failure of Wall Separating EFWS/DG Room		0.84

ITEM 19

Table 19.7.5.4-7 Summary of HCLPFs for Seismic Sequences for Soil Site B4

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Sequence	HCLPF	Dominant Cutset	Mixed HCLPF	Failure Prob.	Dominant Cutset
EQSTR	.73	-	1.25	2.50E-01	-
EQA-15	-	-	.74	1.00E-02	1
SEIS-SBO	.89	26	.12	2.10E-04	100
EQLP-8	.89	44	.89	5.00E-02	151
EQA-10	.89	6	.97	1.19E-03	42
EQLP-4	.90	61	-	-	-
EQLP-7	.90	4	-	-	-
EQSLO-3	.90	71	-	-	-
EQSLO-6	.90	7	-	-	-
EQA-3	.90	80	-	-	-
EQA-4	.90	76	-	-	-
EQSLO-9	.90	10	-	-	-
EQLP-3	.90	24	-	-	-
EQT-3	.90	80	-	-	-
EQT-7	.90	3	-	-	-
EQA-7	.90	2	-	3.25E-02	58
EQA-8	.90	67	-	3.25E-02	174
EQSLO-10	.90	17	-	3.30E-03	64
EQT-4	.90	100	-	5.60E-02	138
EQT-8	.90	76	-	5.60E-02	142
EQRVR	.91	-	-	-	-
EQLP-5	1.35	4	-	-	-
EQSLO-4	1.35	6	-	-	-
EQA-5	1.35	21	-	-	-
EQT-5	1.35	5	-	-	-
EQA-12	1.35	2	-	-	-
EQA-14	1.35	1	-	-	-
EQA-9	1.35	3	-	9.15E-03	79
EQLP-9	1.35	5	-	9.15E-03	82
EQT-9	1.35	6	-	9.15E-03	71
EQSLO-11	1.35	11	-	6.40E-02	80
EQSLO-7	1.35	3	-	9.15E-03	17
EQA-13	1.35	1	-	5.60E-02	27
EQLP-12	1.35	1	-	5.60E-02	53
EQLP-11	1.35	1	-	1.55E-03	15
EQML-3	2.59	1	-	5.60E-02	29
EQML-2	2.59	1	-	5.60E-02	17

Figure 19.11.5.4.5.1-6 shows that the cavity basemat erosion is insignificant (less than 1 inch) during this transient. This is due to adequate quenching of the core debris in the reactor cavity.

19.11.5.4.5.3 Fission Product Releases

A summary of fission product group concentrations in the containment atmosphere at 24 hours after vessel breach is provided in Table 19.11.5.4.5.1-3.

19.11.5.4.6 V Sequence

The dominant System 80+ V Sequence consists of an intersystem LOCA (ISL) initiated from a full shear break in the 16" diameter SCS line occurring within the containment building subsphere. This event is identified in the PRA as PDS 17.

In this event all ECCS systems are operable. The failure of the SCS pipe outside of containment results in a gradual transfer of ECCS inventory from the containment to the subsphere. This ultimately results in failure of the ECCS function due to the unavailability of a water source. Details of this transient are discussed below.

19.11.5.4.6.1 RCS Response Characteristics

The ISL represents a large LOCA initiated ^{begins at 7700} outside of containment. Consequently the RCS response is similar to that of the large LOCA discussed in Section 19.11.5.4.2. In this case the SCS line break is equivalent to 1.4 square feet. The larger failure area results in a more rapid RCS response. In this event the core ~~initially uncovers in 76 seconds~~ (See for example Figure 19.11.5.4.6.1-2 and Table 19.11.5.4.6-1). ~~SIT discharge rapidly temporarily recovers the core.~~

(no H) ^{a sustained} The ECCS maintains the RCS covered until the IRWST is depleted and suction is lost to ^{the} ECCS pumps. ~~A second sustained core uncover begins at 2 hours.~~ Support plate failure occurs at 13,800 seconds and RV failure is predicted to occur shortly thereafter.

The large failure area results in a rapid system depressurization to near atmospheric pressure which is sustained for the duration of the transient (Figure 19.11.5.4.6.1-1).

A summary of key transient parameters is provided in Table 19.11.5.4.6-2.

19.11.5.4.6.2 Containment Response Characteristics

The ISLOCA releases all the RCS and containment liquid inventory into the building subsphere. Once the RV fails the corium is assumed to fully drop into the dry reactor cavity. Core concrete attack begins immediately. Concrete erosion will ultimately lead to a basemat failure. However, the bypass pathway provides a more direct means for releasing fission products to the environment. These MAAP analyses do not credit the water accumulation expected in the subsphere ECCS rooms to scrub fission products leaving the RCS. Furthermore, detailed revolatilization models including the large length of SCS piping are likewise not considered in this demonstration.

19.11.5.4.6.3 Fission Product Releases

MAAP predicted fission product releases for the V sequence are summarized in Table 19.11.5.4.6-3.