Westinghouse

Westinghouse Electric Company
Nuclear Power Plants
P.O. Box 355
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USA

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

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Direct fax: 724-940-8505
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Your ref: Docket Number 52-006
Our ref: DCP_NRC_002874

May 21, 2010

References:

1. DCP_NRC_002744, Re-submittal of Proposed Changes for AP1000 Design Control Document Rev.18, January 20, 2010
2. DCP_NRC_002818, Supplementary Information to DCP_NRC_002744 – Re-Submittal of Proposed Changes for AP1000 Design Control Document Rev.18, March 12, 2010
3. DCP_NRC_002850, Information on Proposed Changes for the AP1000 Design Control Document Rev. 18, April 26, 2010

Subject: Final Information on Proposed Changes for the AP1000 Design Control Document Rev. 18

This letter is submitted in support of the AP1000 Design Certification Amendment Application (Docket No. 52-006). The information provided is generic and is expected to apply to all Combined License (COL) applicants referencing the AP1000 Design Certification and the AP1000 Design Certification Amendment Application.

This submittal contains proprietary information of Westinghouse Electric Company, LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal one copy of the Application for Withholding, AW-10-2817 (non-proprietary, Enclosure 1), and one copy of the associated Affidavit (non-proprietary, Enclosure 2) with Proprietary Information and Copyright Notices. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission. Pursuant to 10 CFR 50.30(b), "Description of Proposed Changes for AP1000 DCD Rev. 18 – Proprietary" and "Description of Proposed Changes for AP1000 DCD Rev. 18 – Non-Proprietary" are submitted as Enclosures 3 and 4. Correspondence with respect to the affidavit or Application for Withholding should include our reference number AW-10-2817 and should be addressed to James A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, LLC, P. 0. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Westinghouse provided preliminary information on changes which it proposed to include in Revision 18 of the AP1000 Design Control Document (DCD-18) in a January 20, 2010 letter (Reference 1). Supplementary information on some of those changes requested by the NRC was provided in a March 12, 2010 letter (Reference 2). Information was provided in an April 26, 2010 letter (Reference 3) for seven of the changes identified in the January 20, 2010 that were determined to meet one or more of the Interim Staff Guidance-11 (ISG-11) criteria for reporting to the NRC staff. Two of the remaining 50 "elective"

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items in the January 20 letter (Change Notice numbers 23 and 45) are addressed in RAI responses RAI-SRP11.3-CHPB-05 and RAI-SRP11.5-CHPB-05,, respectively. The purpose of this letter is to provide final information that we propose to include in DCD-18 for those 48 items, as supplemented by information in the March 12 letter and by responses to action items from our March 17-18 meeting with NRC staff. Final information for these 48 elective changes is provided in Enclosure 3, including the reasons for the changes and the sections of the DCD which are impacted. Enclosure 4 provides a non-proprietary version of the information in Enclosure 3. Enclosure 5 provides a copy of the revised DCD pages. Enclosure 6 contains sensitive, unclassified, non-safeguards information relative to the physical protection of an AP1000 Nuclear Power Plant that should be withheld from public disclosure pursuant to 10CFR2.390(d). This information is associated with CNs 53, 54, and 57. Enclosure 7 provides the redacted version of Enclosure 6 (public version). Enclosure 8 provides the responses to Action Items from the March 17, 2010 meeting. The information contained in this letter in addition to Reference 3 and the RAI responses noted above finalizes the information provided in References 1 and 2.

As noted previously, the changes described in this and the referenced letters do not constitute all of the changes which Westinghouse proposes to include in DCD-18. Rather, the changes in this letter are in addition to those which Westinghouse either has submitted or will submit to the NRC as responses to Requests for Additional Information or Safety Evaluation Report Open Items.

Westinghouse will work with the NRC staff to disposition the changes described in this letter as expeditiously as possible. Questions related to the content of this letter should be directed to Westinghouse. Please send copies of such questions to the prospective COL applicants referencing the AP1000 Design Certification. A representative for each applicant is included on the cc: list of this letter.

Very truly yours,

Robert Sisk

Robert Sisk, Manager
Licensing and Customer Interface
Regulatory Affairs and Strategy

/Enclosures

1. AW-10-2817, Application for Withholding Proprietary Information from Disclosure, dated May 21, 2010
2. AW-10-2817, Affidavit, Proprietary Information Notice, Copyright Notice dated May 21, 2010
3. Description of Proposed Changes for AP1000 DCD Rev. 18, Proprietary
4. Description of Proposed Changes for AP1000 DCD Rev. 18, Non-Proprietary
5. DCD Pages
6. DCD Pages for Change Numbers 53, 54, and 57 – Security Related Information Withheld from Public
7. DCD Pages for Change Numbers 53, 54, and 57 – Public Redacted Version
8. Responses to Action Items from March 17, 2010 Meeting, Non-Proprietary

cc:	D. Jaffe	-	U.S. NRC
	E. McKenna	-	U.S. NRC
	T. Spink	-	TVA
	P. Hastings	-	Duke Power
	R. Kitchen	-	Progress Energy
	A. Monroe	-	SCANA
	P. Jacobs	-	Florida Power & Light
	C. Pierce	-	Southern Company
	E. Schmiech	-	Westinghouse
	G. Zinke	-	NuStart/Entergy
	R. Grumbir	-	NuStart
	M. Melton	-	Westinghouse

ENCLOSURE 1

AW-10-2817

APPLICATION FOR WITHHOLDING
PROPRIETARY INFORMATION FROM DISCLOSURE

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
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U.S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
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Direct tel: 412-374-6206
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Your ref: Docket Number 52-006
Our ref: AW-10-2817

May 21, 2010

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Final Information on Proposed Changes for the AP1000 Design Control Document Rev. 18

The Application for Withholding is submitted by Westinghouse Electric Company, LLC (Westinghouse), pursuant to the provisions of Paragraph (b) (1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-10-2817 accompanies this Application for Withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectively requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this Application for Withholding or the accompanying affidavit should reference AW-10-2817 and should be addressed to James A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, LLC, P.O. Box 355, Pittsburgh, Pennsylvania, 15230-0355.

Very truly yours,

Robert Sisk, Manager
Licensing and Customer Interface
Regulatory Affairs and Strategy

cc: G. Bacuta - U.S. NRC

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ENCLOSURE 2

Affidavit

AW-10-2817
May 21, 2010

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared Robert Sisk, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Robert Sisk, Manager
Licensing and Customer Interface
Regulatory Affairs and Strategy

Sworn to and subscribed
before me this 21st day
of May 2010.

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Linda J. Bugle, Notary Public
City of Pittsburgh, Allegheny County
My Commission Expires June 18, 2013
Member, Pennsylvania Association of Notaries

Linda J. Bugle
Notary Public

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(1) I am Manager, Licensing and Customer Interface, Westinghouse Electric Company, LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Final Information on Proposed Changes for the AP1000 Design Control Document Rev. 18" in support of the AP1000 Design Certification Amendment Application, being transmitted by Westinghouse letter (DCP_NRC_002874) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse for the AP1000 Design Certification Amendment application is expected to be applicable in all licensee submittals referencing the AP1000 Design Certification and the AP1000 Design Certification Amendment Application in response to certain NRC requirements for justification of compliance of the safety system to regulations.

This information is part of that which will enable Westinghouse to:

(a) Manufacture and deliver products to utilities based on proprietary designs.

(b) Advance the AP1000 Design and reduce the licensing risk for the application of the AP1000 Design Certification

(c) Determine compliance with regulations and standards

(d) Establish design requirements and specifications for the system.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of plant construction and operation.

(b) Westinghouse can sell support and defense of safety systems based on the technology in the reports.

(c) The information requested to be withheld reveals the distinguishing aspects of an approach and schedule which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar digital technology safety systems and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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ENCLOSURE 4

Description of Proposed Changes for AP1000 DCD Rev. 18, Non-Proprietary

a,c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
1	Reported in DCP_NRC_002850 (4/26/2010)							
2	2	3	Table 3.2-3 (Sh 30/68)	AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment-add info on affected valves RNS-PL-V025/V026.				Medium
			Table 3.9-16 (Shs 14,15,23/23)	Add Containment Isolation Leak Test to IST Type and Frequency column. Delete Note 16.				
			Table 3-11-1 (Sh 41/50)	Environmentally Qualified Electrical and Mechanical Equipment-add info on affected valves RNS-PL-V025/V026				
			Table 3I.6-3 (Sh 21/31)	Add info on RNS-PL-V025/V026 to table				
		5	Figure 5.4-7	RNS P&ID-add detail of changes to affected valves RNS-PL-V025/V026 to figure				
		6	Table 6.2.3-1 (Sh 2, 4/4)	Containment Mechanical Penetrations and Isolation Valves – add info on valves RNS-PL-V002A/B. Delete Note 6.				
3	2	3	Table 3.9-16 (Sheet 10/23)	V0180A/B was changed from globe to ball valve.	The valve was changed from a globe valve to a ball valve.		ASME Boiler and Pressure Vessel Code, Section III - Class 1	Medium
		6	Section 6.3.2.1.2 Figure 6.3-2 Figure 6.3-4	See Enclosure 8, Action Item 9.				

a,c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
4	2	3	Table 3.2-3 (sh 22/68)	Six test valves added.	Remove the system requirements for remote position indication for valves PXS-PL-016A/B, PXS-PL-017A/B, PXS-PL-028A/B and PXS-PL-029A/B. Change valves PXS-PL-016A/B, PXS-PL-017A/B from swing check to in-line nozzle check valves. Add three test connections for each CMT outlet line to facilitate testing of the check valves. Add three safety related 1-inch manual valves to each CMT outlet line.		ASME Boiler and Pressure Vessel Code, Section III - Class 1	Medium
			Table 3.11-1 (sh 38/50)	Six test valves added.				
			Table 3.9-16 (sh 8,9/23), Notes	Four CMT check valves added; remote position function removed for accumulator check valves; Note 10 revised to remove quarterly open verification.				
			Table 3I.6-3 (Sh 18/31)	Six test valves added.				
		6	Section 6.3.2.2.8.1	Delete sentence describing check valves. This sentence had described all check valves as either piston type or swing type which is no longer true.				
			Section 6.3.7.6.1	Revise valve position indication description.				
			Section 6.3.7.6.2	Delete last sentence because there is no reference to Chapter 7 in Table 6.3-1.				
			Figure 6.3-1	Revise PXS P&ID				
			Table 6.3-3 (sh 1/4)	Revise entry for CMT discharge line check valves.				
5	Reported in DCP_NRC_002850 (4/26/2010)							
6	Reported in DCP_NRC_002850 (4/26/2010)							
7	2	6	Figure 6.3-2 (sh 2)	Change of position for the PRHR HX flow meter				Medium

a,c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
8	Reported in DCP_NRC_002850 (4/26/2010)							
9	Reported in DCP_NRC_002850 (4/26/2010)							
10A	1	2	Section 2.2.5	Section 2.2.5 - correct table number is 2.2.5-5				Low
10B	2	1	Reg Guide 1.93	Reg. Guide 1.93 - change judgement to judgment				Low
			Table 1.1-1 (Shs 1,2,3/4)	Table 1.1-1: Sh 1- Delete "n" from American Sh 2 - In-containment - lower case and hyphen Sh 2 - Add GRCA Sh 2 - Add Criterion Sh 3 - Add "s" to Motor operated valves				
			Table 1.6-1 (Sh 4/20)	Table 1.6-1: Revision 1 to Revision 2 for WCAP-15949.				
	2*	3	Section 3.7.3.13.4.3	Section 3.7.3.13.4.3 Third and fourth bullets should be combined into one bullet.				
	2	3	Section 3.8.3.5.7	Section 3.8.3.5.7- change judgement to judgment				
			Section 3.8.4.5.3	Section 3.8.4.5.3- change judgement to judgment				
			Section 3.8.5.4.3	Section 3.8.5.4.3- change judgement to judgment				
			Section 3.8.5.4.4	Section 3.8.5.4.4 - change judgement to judgment				
				Section 3.8.5.4 - grammatical error - change "do" to "does"				
			Section 3.10	Section 3.10.1.1 - reference renumbering				

a,c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
			Section 3B.2.6	Section 3B.2.6 - typo - "vavle" to "valves"				
		5	Section 5.4.7.2	Section 5.4.7.2 - in-containment				
			Table 5.4-17	Table 5.4-17 - add word "subsection" in Note				
		9	Table 9.2.1-1	Table 9.2.1-1 – "is" to "in"				
			Table 9A-3 (Shs 8, 24/24)	Table 9A-3: Sh 8 – change RNS Pump B heat value Sh24 - Table entries out of order and fire area zone 6030AF603214 should be 6030AF60324.				
			Section 9A.3.6.2	Section 9A.3.6.2 – Change "room" to "area				
		14	Section 14.2.9.4.9	Section 14.2.9.4.9 - move "and" to before blower				
		16	Section B 3.2.5	Section B 3.2.5 - RTD should be RTP				
			Section B 3.6.2 Section B 3.8.1 Section B 3.8.5 Section B 3.9.2 Section B 3.9.4	B 3.6.2, B 3.8.1, B 3.8.5, B 3.9.2, B 3.9.4 - change judgement to judgment See Enclosure 8, Action Item 12.				
		17	Table 17.4-1 (Sh 1/8)	Table 17.4-1 - type in title - delete "DCD"				
		18	Section 18.8	Section 18.8 - Reference 45: add document number and revision; correct title				
		19	Section 19.1.7	Reference 19.1-3 – provide document number and full title				

a,c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
			Section 19.59.6.1 Section 19.59.6.2	Section 19.59.6.1 - Change judgement to judgment Section 19.59.6.2 - Change judgement to judgment				
11	2	3	Section 3.4.1.2.2.1	Correct flood barrier and description of Rooms 11207 - 11209				Low
12	2	9	Section 9.1.5.2.1.2 Section 9.1.5.2.2.2	Change the word "shall" to "should" Change the word "shall" to "should"				Low
13	2	8	Table 8.3.1-2 (Sheet 4/4)	Spent fuel pump horsepower				Low
14	Reported in DCP_NRC_002850 (4/26/2010)							
15	2	3 5 16	Table 3.3.1-1 Table 3.3.2-1 Section 5.5.7 Section B 3.1.9 Section B 3.4.1 Section B 3.6.6 Section B 3.6.8	Table 3.3.1-1 - add other to header Table 3.3.2-1 - Page 9 header spacing Section 5.5.7 - "ensure" to "ensures" Section B 3.1.9 - "signalled" to "signaled" Section B 3.4.1 - "loadchanges" to "load changes" Section B 3.6.6 - "these" to "this" Section B 3.6.8 - "a accident" to "an accident"				Low

a,c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
			Section B 3.6.9 Section B 3.7.10 Section B 3.8.5 Table B.3.8.5-1	Section B 3.6.9 - "dash" to a "dot" Section B 3.7.10 - "Reference 6" to "Reference 3" Section B 3.8.5 - "Ref. 3" to "Ref. 4". Add Ref 4. Table B 3.8.5 - center Table Heading				
16	2	15	Tier 2 Master TOC Tier 2 Rev 17 - Change Roadmap Ch 15, p cxci - cxciv	Replace all DCP/NRC2209 with DCP/NRC2321				Low
17	2	3	Figure 3G.4-7Y Figure 3G.4-7Z	These figures should be changed to Elevation 116.50' to match the elevation on Figure 3G.4-7X				Low
18	2	9	Table 9.1-3 (Sheet 1/2)	Changed Spent Fuel Pool Heat Exchanger Design Pressure from 150 to 200 psig				Low

a, c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
19	2	5	Table 5.2-3	Replace Code Case 2142-1 with 2142-2. See Enclosure 8, Action Item 2.	Replace Code Case 2142-1 with 2142-2 to allow UNS N06054 to be used.		ASME Code Case 2142-2. F-Number Grouping for Ni-Cr-Fe Filler Metals Section IX	Low
20	2	6	Section 6.2.4.2.3 Table 6.2.4-3	Hydrogen Igniter Temperatures - the section and Table will be revised to reflect a minimum temperature of 1700F vs a range of 1600-1700F				Low
21	2	9	Section 9.3.5.2.2	Sumps and drain tanks-delete "into the VAS exhaust system" See Enclosure 8, Action Item 1.				Low
22	2	5	Table 5.2-1 (Sheet 4/6)	Table 5.2-1 of the DCD identifies the acceptable material for use on the CRDM latch housing and rod travel housing as SA-336 on sheet 4. This is changed to SA-182. See Enclosure 8, Action Item 3.	There is no actual change being made to the design. The material for the CRDM pressure boundary was always SA-182, which is consistent with sheet 1 of Table 5.2-1.		ASME Boiler and Pressure Vessel Code, Section II - Materials	Low
23	SEE RAI SRP-11.3-CHPB-05							

a,c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
24	2	9	Section 9A.3.4.14 Table 9A-3 (Sheets 19, 20/24)	Fire Area 4034 AF 01 - change subdivisions Fire Protection Summary- entries in table modified as appropriate				Low
25	2	5 9A 19D	Section 5.3.1.1 Section 9A.3.4.3B Table 9A-3 (Sheet 5/24) Section 19D.8.2.2 Section 19D.8.2.13	Added flow skirt to list of components for which the reactor vessel provides support. Change description of Fire Area 4002 AF 03 Change combined load and equivalent duration for Fire Area 1244 AF 12451, Security Room. Revised description of equipment survivability assessment for thermocouples Discussion of use of float level sensors in severe accident management.				Low
26	2	9	Section 9.2.11.1	Change "compatibility" to "habitability"				Low
27	2	10	Table 10.1-1	Steam Generator Outlet Pressure - Revise value to 821 psig from 823 psig				Low
28	2	5	Section 5.4.4.3	Units of pressure differential - Change "psig" for the units of pressure differential to "psi"				Low
29	2	1	Appendix 1A, Reg. Guide 1.50	Criteria wording - Change "pressurized water heat transfer" to "post-weld heat treat"				Low

a,c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
30	2	3	Table 3.2-3 (sh 10/ 69) Table 3.11-1 (sh 33/51) Table 3I.6-3 (sh 13/32)	Adds a class C valve to table 3.2-3, 3.11-1 and 3I.6-3.	This change adds class C valve PCS-PL-V026 (Water Bucket Auxiliary Makeup Line Isolation Valve) to Tables 3.2-3, 3.11-1 and 3I.6-3		10 CFR 52 SECY-08-0152	Low
31	2	1	Table 1.8-1 (sh1/6)	Remove Item 1.1 of Table 1.8-1.	Delete Item 1.1 of Table 1.8-1 to be consistent with Section 1.9.3 of DCD.		Post-Accident Sampling, NUREG-0737	Low
32	1	2	Table 2.6.9-1	ITAAC number 3 in the Acceptance Criteria reads "see Tier 1 material, Table 3.3-6, Item 6". It should be "item 16"				Low
33	2	9	Table 9A-3 (sh12/24)	Fire Protection Summary-Fire Area 2030AF20300-change Heat Values for plastic and volatiles				Low
34	2	9	Table 9.3.3-2 (Sheet 3/4)	WGS inlet moisture indication - Remove WGS inlet moisture indication consistent with Ch 11 changes to remove moisture monitor.				Low

a,c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
35	2	9	Table 9.3.4-1 (Sheet 1, 2/2)	Secondary Sampling System - Clarify the type of specific conductivity measured.				Low
36	2	1 9	Section 1.8 Table 1.8-1 (Sheet 5/7) Section 9.2.5.3	Waste Water System - Remove retention basin and raw water system associations.				Low
37	No Technical Change – PDF Error							
38	2	1A	Conformance with Regulatory Guides - Criteria Section C.5 and C.6 of RG 1.133 Rev 1, Loose Part Detection Program for the Primary System of Light-Water-Cooled Reactors	Revised reason for criteria section C. 5 and C.6. See Enclosure 8, Action Item 6.				Low
39	2	17	Section 17.6	APP-GW-GL-022 is referenced as Rev. 0. That should be Rev. 8.				Low

a,c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
40	2	3 16	Table 3.3.3-1 B.3.3.3	Post-Accident Monitoring Instrumentation list of variables. Revised Function 19 requirements. PAM Instrumentation Bases. Revised Function 19 Bases. See Enclosure 8, Action Item 4.	Change the number of required instruments for Function 19 from 2 to 1 for the IRWST to RNS suction valve status variable, revise note (c), and revise the Bases, B 3.3.3, Function 19 for the IRWST to RNS Suction Valve Status. The position of the two motor-operated valves in the line from the IRWST to the RNS pump suction header is monitored to verify that the flow path is isolated following postulated events. The flow path must be isolated to prevent loss of IRWST inventory into the RNS.		Reg. Guide 1.97	Low
41	2*	3	Figure 3.8.3-8 (Sheet 3/3)	Modify Plant Module Bolted Connection				Low
42	2	1	Section 1.9.5.1.11	The normal operating pressure is shown as 2250 psig and it should be 2250 psia. See Enclosure 8, Action Item 7.				Low
43	2	1	Section 1.7	Text was provided on Interpretation of figures in Tier 2.				Low

a,c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
44	2	5	Table 5.2-1 (Sh 2/6)	Table 5.2-1 of the DCD identifies the acceptable safe end material for use on the pressurizer as SB-163 (NO6690). This is changed to SB-564. See Enclosure 8, Action Item 11.	The material designated for the nozzle safe ends of the pressurizer is changed from SB-163 to SB-564.		ASME Boiler and Pressure Vessel Code, Section II - Materials	Low
45	SEE RAI SRP-11.5-CHPB-05							
46	2	4	Section 4.5.2.1	Reactor Internal and Core Support Materials, Materials Specifications - added the material used for the flow skirt; added product forms that may be used by reactor internals structures; added components that were not stainless steel, i.e., locating and support pins, instrumentation adapter, instrument tube tip, guide stud, instrument stalk spring; changed instrument guide tube spring to instrument tube sleeve spring; added statement that reactor internals structures will use threaded structural fasteners of strain hardened Type 316 stainless steel; added internals structures materials as being addressed in the ASME code; clarified applicable section of ASME code.	This section describes the materials and product forms used for reactor vessel internals. There are additional materials and product forms that are being used beyond what is currently described in the DCD. These are related to internal structures and are provided in the revision. In addition the section of the ASME Code referenced is incorrect and is corrected in the revision. See Enclosure 8, Action Item 10.		ASME Code, Section II Part D, Subpart 1	Low

a,c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
47	2	7	Section 7.6.2.1	Availability of engineered safety features - Passive residual heat removal heat exchanger inlet isolation valve. Delete the following sentence: To prevent an inadvertent closure of the valve, redundant output cards are used in the protection and safety monitoring system cabinet. See Enclosure 8, Action Item 5.				Low
48	2	1	Table 3.2-3 sheet 21/69	AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment - the safety classification of valves PXS-PL-V128A/B and 129A/B is changed from Class B to Class A				Low
49	2	3	Section 17.4.7.4	Revise section to point to Table 3.9-16 for testing requirements.				Low
50	2	3	Table 3.9-16 sh 10, 11/23	Valve in-service test requirements - Revise the entry under Inservice Testing Type and Frequency for valves PXS-PL-V119A, PXS-PL-V119B, PXS-PL-				

a,c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
				V122A, PXS-PL-V122B, PXS-PL-V124A and PXSPLV124B as follows: (1) Remote Position Indication, Exercise/2 years (2) Check-Initial-Open Differential Pressure Refueling (3) Check Exercise/Refueling Shutdown				
51	2	5	Section 5.4.4.2	Main steam line flow restriction - design description				Low
52	2	16	B 3.6.9 B 3.5.1 B 3.5.2 B 3.7.2	Change number of baskets from two to four Consistency - spelling of "steam line"				Low
53A	1	3	Figure 3.3-11 Figure 3.3-12 Figure 3.3-13	Annex Building Plan View - delete column lines which are not mentioned in Table 3.3-1				Low
53B	2	3	Figure 3.7.2-19 (Sheets 1,2,3,5,6,7,8/10)	Annex Building Key Structural Dimensions - change Annex Bldg column line 10 to 10.05				Low
54	2 2* 2	1 6 9 12	Figure 1.2-8 Figure 1.2-9 Figure 6.4-1 Figure 9A-1(Sh 6/16) Figure 12.3-1(Sh 7/16) Figure 12.3-2(Sh 7/15) Figure 12.3-3(Sh 7/16)	Nuclear island fire area plan-relocate kitchen; decrease size of shift supervisor's office and add a second bathroom; add a basin in the raised floor below the toilet rooms and kitchen; relocate ancillary fans and self-contained breathing apparatus; add a fire door between operator break room and work area.				Low

a,c

Change Number	Tier	Chapter Number	Section/Table/ Figure Numbers	Description of DCD Change	Description of Design Change		Applicable Regulatory Standard that Design Meets	Assessment of NRC Review Resources
55A	Reported in DCP_NRC_002850 (4/26/2010)							
55B								
56A	1	2	List of Figures Fig 2.3.3-1	Delete "and Auxiliary Boiler" from figure title See Enclosure 8, Action Item 8.				Low
56B	2	14	Table 14.3-1	ITAAC Screening Summary- Delete "and Auxiliary Boiler" from Structure/System Description				Low
57	2	1 9	Fig 1.2-9 Section 9.1.4.3.1	Nuclear Island General Arrangement Safety Evaluation - Refueling Machine				Low

ENCLOSURE 5

DCD Pages

Change Number 2

Table 3.2-3 (Sheet 30 of 6865)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Normal Residual Heat Removal System (Continued)					
RNS-PL-V015B	RNS Discharge RCS Pressure Boundary	A	I	ASME III-1	
RNS-PL-V016	RNS Discharge Containment Penetration Isolation Valves Test	C	I	ASME III-3	
RNS-PL-V017A	RNS Discharge RCS Pressure Boundary	A	I	ASME III-1	
RNS-PL-V017B	RNS Discharge RCS Pressure Boundary	A	I	ASME III-1	
RNS-PL-V021	RNS HL Suction Pressure Relief	B	I	ASME III-2	
RNS-PL-V022	RNS Suction Header Containment Isolation - ORC	B	I	ASME III-2	
RNS-PL-V023	RNS Suction from IRWST - Containment Isolation	B	I	ASME III-2	
RNS-PL-V024	RNS Discharge to IRWST Isolation	C	I	ASME III-3	
RNS-PL-V025	RNS Suction from IRWST – Bonnet Relief Isolation	C	I	ASME III-3	
RNS-PL-V026	RNS Suction from IRWST – Containment Isolation Test	C	I	ASME III-3	
RNS-PL-V029	RNS Discharge to CVS	C	I	ASME III-3	
RNS-PL-V030A	RNS HX A Shell Drain	D	NS	ANSI B31.1	
RNS-PL-V030B	RNS HX B Shell Drain	D	NS	ANSI B31.1	
RNS-PL-V031A	RNS Train A Discharge Flow Instrument Isolation	C	I	ASME III-3	
RNS-PL-V031B	RNS Train B Discharge Flow Instrument Isolation	C	I	ASME III-3	
RNS-PL-V032A	RNS Train A Discharge Flow Instrument Isolation	C	I	ASME III-3	
RNS-PL-V032B	RNS Train B Discharge Flow Instrument Isolation	C	I	ASME III-3	
RNS-PL-V033A	RNS Pump A Suction Pressure Instrument Isolation	C	I	ASME III-3	
RNS-PL-V033B	RNS Pump B Suction Pressure Instrument Isolation	C	I	ASME III-3	

Table 3.9-16 (Sheet 14 of 23)

VALVE INSERVICE TEST REQUIREMENTS

Valve Tag Number	Description[1]	Valve/Actuator Type	Safety-Related Missions	Safety Functions[2]	ASME Class/ IST Category	Inservice Testing Type and Frequency	IST Notes
RCS-PL-V014B	Fourth Stage Automatic Depressurization System Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years	
RCS-PL-V014C	Fourth Stage Automatic Depressurization System Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years	
RCS-PL-V014D	Fourth Stage Automatic Depressurization System Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years	
RCS-PL-V150A	Reactor Vessel Head Vent	Remote SO GLOBE	Maintain Open Maintain Close Transfer Open	Active-to-Failed RCS Pressure Boundary Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Operability Test	4, 31
RCS-PL-V150B	Reactor Vessel Head Vent	Remote SO GLOBE	Maintain Open Maintain Close Transfer Open	Active-to-Failed RCS Pressure Boundary Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Operability Test	4, 31
RCS-PL-V150C	Reactor Vessel Head Vent	Remote SO GLOBE	Maintain Open Maintain Close Transfer Open	Active-to-Failed RCS Pressure Boundary Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Operability Test	4, 31
RCS-PL-V150D	Reactor Vessel Head Vent	Remote SO GLOBE	Maintain Open Maintain Close Transfer Open	Active-to-Failed RCS Pressure Boundary Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Operability Test	4, 31
RCS-K03	Safety Valve Discharge Chamber Rupture Disk	Relief	Transfer Open	Active	Class 3 Category BC	Inspect and Replace/5 Years	
RCS-K04	Safety Valve Discharge Chamber Rupture Disk	Relief	Transfer Open	Active	Class 3 Category BC	Inspect and Replace/5 Years	
RNS-PL-V001A	RNS Hot Leg Suction Isolation - Inner	Remote MO GATE	Maintain Close Transfer Close	Active RCS Pressure Boundary Safety Seat Leakage Remote Position	Class 1 Category A	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Pressure Isolation Leak Test/2 Years Operability Test	15, 31
RNS-PL-V001B	RNS Hot Leg Suction Isolation - Inner	Remote MO GATE	Maintain Close Transfer Close	Active RCS Pressure Boundary Safety Seat Leakage Remote Position	Class 1 Category A	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Pressure Isolation Leak Test/2 Years Operability Test	15, 31
RNS-PL-V002A	RNS Hot Leg Suction and Containment Isolation - Outer	Remote MO GATE	Maintain Close Transfer Close	Active RCS Pressure Boundary Containment Isolation Safety Seat Leakage Remote Position	Class 1 Category A	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Pressure Isolation Leak Test/2 Years Operability Test Containment Isolation Leak Test	15, ~~16,~~ 31

Table 3.9-16 (Sheet 15 of 23)

VALVE INSERVICE TEST REQUIREMENTS

Valve Tag Number	Description[1]	Valve/Actuator Type	Safety-Related Missions	Safety Functions[3]	ASME Class/ IST Category	Inservice Testing Type and Frequency	IST Notes
RNS-PL-V002B	RNS Hot Leg Suction and Containment Isolation - Outer	Remote MO GATE	Maintain Close Transfer Close	Active RCS Pressure Boundary Containment Isolation Safety Seat Leakage Remote Position	Class 1 Category A	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Pressure Isolation Leak Test/2 Years Operability Test Containment Isolation Leak Test	15, ~~16,~~ 31
RNS-PL-V003A	RCS Pressure Boundary Valve Thermal Relief	Check	Maintain Close Transfer Open Transfer Close	Active RCS Pressure Boundary	Class 2 Category BC	Check Exercise/Refueling Shutdown	23
RNS-PL-V003B	RCS Pressure Boundary Valve Thermal Relief	Check	Maintain Close Transfer Open Transfer Close	Active RCS Pressure Boundary	Class 2 Category BC	Check Exercise/Refueling Shutdown	23
RNS-PL-V011	RNS Discharge Containment Isolation Valve - ORC	Remote MO GATE	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	Class 2 Category A	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test Exercise Full Stroke/Quarterly Operability Test	27, 31
RNS-PL-V012	RNS Discharge Containment Isolation Test Connection Valve	Manual	Transfer Open Maintain Close	Active	Category B	Exercise Full Stroke/2 Years	
RNS-PL-V013	RNS Discharge Containment Isolation - IRC	Check	Maintain Close Transfer Open Transfer Close Maintain Open	Active Containment Isolation Safety Seat Leakage	Class 2 Category AC	Containment Isolation Leak Test Check Exercise/Quarterly	27
RNS-PL-V015A	RNS Discharge RCS Pressure Boundary	Check	Maintain Close Transfer Close Transfer Open Maintain Open	Active RCS Pressure Boundary Safety Seat Leakage	Class 1 Category AC	Check Exercise/Refueling Shutdown Pressure Isolation Leak Test/2 Years	24
RNS-PL-V015B	RNS Discharge RCS Pressure Boundary	Check	Maintain Close Transfer Close Transfer Open Maintain Open	Active RCS Pressure Boundary Safety Seat Leakage	Class 1 Category AC	Check Exercise/Refueling Shutdown Pressure Isolation Leak Test/2 Years	24
RNS-PL-V017A	RNS Discharge RCS Pressure Boundary	Check	Maintain Close Transfer Open Transfer Close Maintain Open	Active RCS Pressure Boundary Safety Seat Leakage	Class 1 Category AC	Check Exercise/Refueling Shutdown Pressure Isolation Leak Test/2 Years	24
RNS-PL-V017B	RNS Discharge RCS Pressure Boundary	Check	Maintain Close Transfer Open Transfer Close Maintain Open	Active RCS Pressure Boundary Safety Seat Leakage	Class 1 Category AC	Check Exercise/Refueling Shutdown Pressure Isolation Leak Test/2 Years	24

12. This note applies to the PXS IRWST injection check valves (PXS-V122A/B, V124A/B). To exercise these check valves a test cart must be moved into containment and temporary connections made to these check valves. In addition, the IRWST injection line isolation valves must have power restored and be closed. These valves are not exercised during power operations because closing the IRWST injection valve is not permitted by the Technical Specifications and the need to perform significant work inside containment. Testing is not performed during cold shutdown for the same reasons. These valves are exercised during refueling conditions when the IRWST injection lines are not required to be available by Technical Specifications and the radiation levels are reduced.

13. Deleted.

14. Component cooling water system containment isolation motor-operated valves CCS-V200, V207, V208 and check valve CCS-V201 are not exercised during power operation. Exercising these valves would stop cooling water flow to the reactor coolant pumps and letdown heat exchanger. Loss of cooling water may result in damage to equipment or reactor trip. These valves are exercised during cold shutdowns when these components do not require cooling water.

15. Normal residual heat removal system reactor coolant isolation motor-operated valves (RNS-V001A/B, V002A/B) are not exercised during power operation. These valves isolate the high pressure RCS from the low pressure RNS and passive core cooling system (PXS). Opening during normal operation may result in damage to equipment or reactor trip. These valves are exercised during cold shutdowns when the RNS is aligned to remove the core decay heat.

16. ~~Normal residual heat removal system containment isolation motor-operated valves (RNS-V002A/B) are not containment isolation leak tested. The basis for the exception is:~~

~~The valve is submerged during post-accident operations which prevents the release of the containment atmosphere radiogas or aerosol.~~

~~The RNS is a closed, seismically designed safety class 3 system outside containment~~

~~The valves are closed when the plant is in modes above hot shutdown~~Not Used.

17. Not Used.

18. This note applies to the CAS instrument air containment isolation valves (CAS-V014, V015). It is not practical to exercise these valves during power operation or cold shutdowns. Exercising the valves during these conditions may result in some air-operated valves inadvertently opening or closing, resulting in plant or system transients. These valves are exercised during refueling conditions when system and plant transients would not occur.

19. Primary sampling system containment isolation check valve (PSS-V024) is located inside containment and considerable effort is required to install test equipment and cap the discharge line. Exercise testing is not performed during cold shutdown operations for the same reasons. These valves are exercised during refueling conditions when the radiation levels are reduced.

20. This note applies to the main steam isolation valves and main feedwater isolation valves (SGS-V040A/B, V057A/B). The valves are not full stroke tested quarterly at power since full valve stroking will result in a plant transient during normal power operation. Therefore, these valves will be full stroke tested on a cold shutdown frequency basis. The full stroke testing will be a full "slow" closure operation. The large size and fast stroking nature of the valve makes it advantageous to limit the number of fast closure operations which the valve experiences. The timed slow closure supports the continued operability status of the valves in the intervals between fast closure tests and ensures that the valve is not mechanically bound.

21. Post-72 hour check valves that require temporary connections for inservice-testing are exercised every refueling outage. These valves require transport and installation of temporary test equipment and pressure/fluid supplies. Since the valves are normally used very infrequently, constructed of stainless steel, maintained in controlled environments, and of a simple design, there is little benefit in testing them more frequently. For example, valve PCS-V039 is a simple valve that is opened to provide the addition of water to the PCS post-72 hour from a temporary water supply. To exercise the valve, a temporary pump and water supply is connected using temporary pipe and fittings, and the flow rate is observed using a temporary flow measuring device to confirm valve operation.

22. Exercise testing of the auxiliary spray isolation valve (CVS-V084, V085) will result in an undesirable temperature transient on the pressurizer due to the actuation of auxiliary spray flow. Therefore, quarterly exercise testing will not be performed. Exercise testing will be performed during cold shutdowns.

23. Thermal relief check valves in the normal residual heat removal suction line (RNS-V003A/B) and the Chemical and Volume Control System makeup line (CVS-V100) are located inside containment. To exercise test these valves, entry to the containment is required and temporary connections made to gas supplies. Because of the radiation exposure and effort required, this test is not conducted during power operation or during cold shutdowns. Exercise testing is performed during refueling shutdowns.

24. Normal residual heat removal system reactor coolant isolation check valves (RNS-V015A/B, V017A/B) are not exercise tested quarterly. During normal power operation these valves isolate the high pressure RCS from the low pressure RNS. Opening during normal operation would require a pressure greater than the RCS normal pressure, which is not available. It would also subject the RCS connection to undesirable transients. These valves will be exercised during cold shutdowns.

25. This note applies to the main feedwater control valves (SGS-V250A/B), moisture separator reheater 2nd stage steam isolation valve (MSS-V015A/B), turbine control valves (MTS-V002A/B, V004A/B). The valves are not quarterly stroke tested since full stroke testing would result in a plant transient during power operation. Normal feedwater and turbine control operation provides a partial stroke confirmation of valve operability. The valves will be full stroke tested during cold shutdowns.

26. This note applies to containment compartment drain line check valves (SFS-V071, SFS-V072, WLS-V071A/B/C, WLS-V072A/B/C). These check valves are located inside containment and require temporary connections for exercise testing. Because of the radiation exposure and effort required, these valves are not exercised during power operation or during cold shutdowns. The valves will be exercised during refuelings.

27. Containment isolation valves leakage test frequency will be conducted in accordance with the "Primary Containment Leakage Rate Test Program" in accordance with 10 CFR 50 Appendix J. Refer to SSAR subsection 6.2.5.

28. This note applies to the chilled water system containment isolation valves (VWS-V058, V062, V082 and V086). Closing any of these valves stops the water flow to the containment fan coolers. This water flow may be necessary to maintain the containment air temperature within Technical Specification limits. As a result, quarterly exercise testing will be deferred when plant operating conditions and site climatic conditions would cause the containment air temperature to exceed this limit during testing.

29. Exercise testing of the turbine bypass control valves (MSS-V001, V002, V003, V004, V005 and V006) will result in an undesirable temperature transient on the turbine, condenser and other portions of the turbine bypass due to the actuation of bypass flow. Therefore, quarterly exercise testing will not be performed. Exercise testing will be performed during cold shutdowns.

30. Deleted.

31 These valves are subject to operability testing per the requirements of 10 CFR 50.55a. The test frequencies are to be established in accordance with the results of the Joint Owners Group (JOG) program for periodic verification of design-basis capability of safety-related motor-operated valves (MOVs). Based on the composition of power-operated valves (POVs) in this table, the JOG approach shall be applied to all actuator types. POV risk ranking and functional margin are used to establish the recommended maximum periodic verification test (Operability) interval.

These POVs (motor-operated, air-operated, solenoid-operated, and hydraulically-operated) shall be addressed in the owner's POV respective program-specific documents. Attributes of these programs shall include lessons learned as delineated in the NRC's Regulatory Issue Summary (RIS) 2000-3, "Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions." See subsection 3.9.6.2.2 for the factors to be considered in the evaluation of operability testing and subsection 3.9.8.4 for the Combined License information item. ~~The test frequency is the longer of every 3 refueling cycles or 5 years until sufficient~~

Table 3.11-1 (Sheet 41 of 50)

ENVIRONMENTALLY QUALIFIED ELECTRICAL AND MECHANICAL EQUIPMENT

Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
RNS HX A Outlet Flow Control	RNS-PL-V006A	6	PB	1 yr	M **
RNS HX B Outlet Flow Control	RNS-PL-V006B	6	PB	1 yr	M **
RNS Pump A Discharge Isolation	RNS-PL-V007A	6	PB	1 yr	M **
RNS Pump B Discharge Isolation	RNS-PL-V007B	6	PB	1 yr	M **
RNS HX A Bypass Flow Control	RNS-PL-V008A	6	PB	1 yr	M **
RNS HX B Bypass Flow Control	RNS-PL-V008B	6	PB	1 yr	M **
RNS Discharge Containment Isolation Valve Test	RNS-PL-V010	6	PB	1 yr	M **
RNS Discharge Containment Isolation Valve Test Connection	RNS-PL-V014	1	PB	1 yr	M *
RNS Discharge Containment Penetration Isolation Valves Test	RNS-PL-V016	1	PB	1 yr	M *
RNS Discharge to IRWST Isolation	RNS-PL-V024	1	PB	1 yr	M *
IRWST to RNS Suction Valve Bonnet Relief Isolation	RNS-PL-V025	1	PB	1 yr.	M *
IRWST to RNS Suction Valve Containment Isolation Test	RNS-PL-V026	1	PB	1 yr.	M *
RNS Discharge to CVS	RNS-PL-V029	1	PB	1 yr	M *
RNS Train A Discharge Flow Instrument Isolation	RNS-PL-V031A	6	PB	1 yr	M **
RNS Train B Discharge Flow Instrument Isolation	RNS-PL-V031B	6	PB	1 yr	M **
RNS Train A Discharge Flow Instrument Isolation	RNS-PL-V032A	6	PB	1 yr	M **
RNS Train B Discharge Flow Instrument Isolation	RNS-PL-V032B	6	PB	1 yr	M **
RNS Pump A Suction Pressure Instrument Isolation	RNS-PL-V033A	6	PB	1 yr	M **
RNS Pump B Suction Pressure Instrument Isolation	RNS-PL-V033B	6	PB	1 yr	M **
RNS Pump A Discharge Pressure Instrument Isolation	RNS-PL-V034A	6	PB	1 yr	M **
RNS Pump B Discharge Pressure Instrument Isolation	RNS-PL-V034B	6	PB	1 yr	M **
RNS Pump A Suction Piping Drain Isolation	RNS-PL-V036A	6	PB	1 yr	M **

Table 3I.6-3 (Sheet 21 of 312)

LIST OF AP1000 SAFETY-RELATED ELECTRICAL AND MECHANICAL EQUIPMENT NOT HIGH FREQUENCY SENSITIVE

Description	AP1000 Tag Number	Comment
RCP 1A Drain	RCS-PL-V261A	2
RCP 1B Drain	RCS-PL-V261B	2
RCP 2A Drain	RCS-PL-V261C	2
RCP 2B Drain	RCS-PL-V261D	2
RCS Pressure Boundary Valve Thermal Relief Isolation	RNS-PL-V004A	2
RCS Pressure Boundary Valve Thermal Relief Isolation	RNS-PL-V004B	2
RNS Pump A Suction Isolation	RNS-PL-V005A	2
RNS Pump B Suction Isolation	RNS-PL-V005B	2
RNS HX A Outlet Flow Control	RNS-PL-V006A	2
RNS HX B Outlet Flow Control	RNS-PL-V006B	2
RNS Pump A Discharge Isolation	RNS-PL-V007A	2
RNS Pump B Discharge Isolation	RNS-PL-V007B	2
RNS HX A Bypass Flow Control	RNS-PL-V008A	2
RNS HX B Bypass Flow Control	RNS-PL-V008B	2
RNS Discharge Containment Isolation Valve Test	RNS-PL-V010	2
RNS Discharge Containment Isolation Valve Test Connection	RNS-PL-V014	2
RNS Discharge Containment Penetration Isolation Valves Test	RNS-PL-V016	2
RNS Discharge to IRWST Isolation	RNS-PL-V024	2
RNS Suction from IRWST – Bonnet Relief Isolation	RNS-PL-V025	2
RNS Suction from IRWST – Containment Isolation Test	RNS-PL-V026	2
RNS Discharge to CVS	RNS-PL-V029	2
RNS Train A Discharge Flow Instrument Isolation	RNS-PL-V031A	2
RNS Train B Discharge Flow Instrument Isolation	RNS-PL-V031B	2
RNS Train A Discharge Flow Instrument Isolation	RNS-PL-V032A	2
RNS Train B Discharge Flow Instrument Isolation	RNS-PL-V032B	2

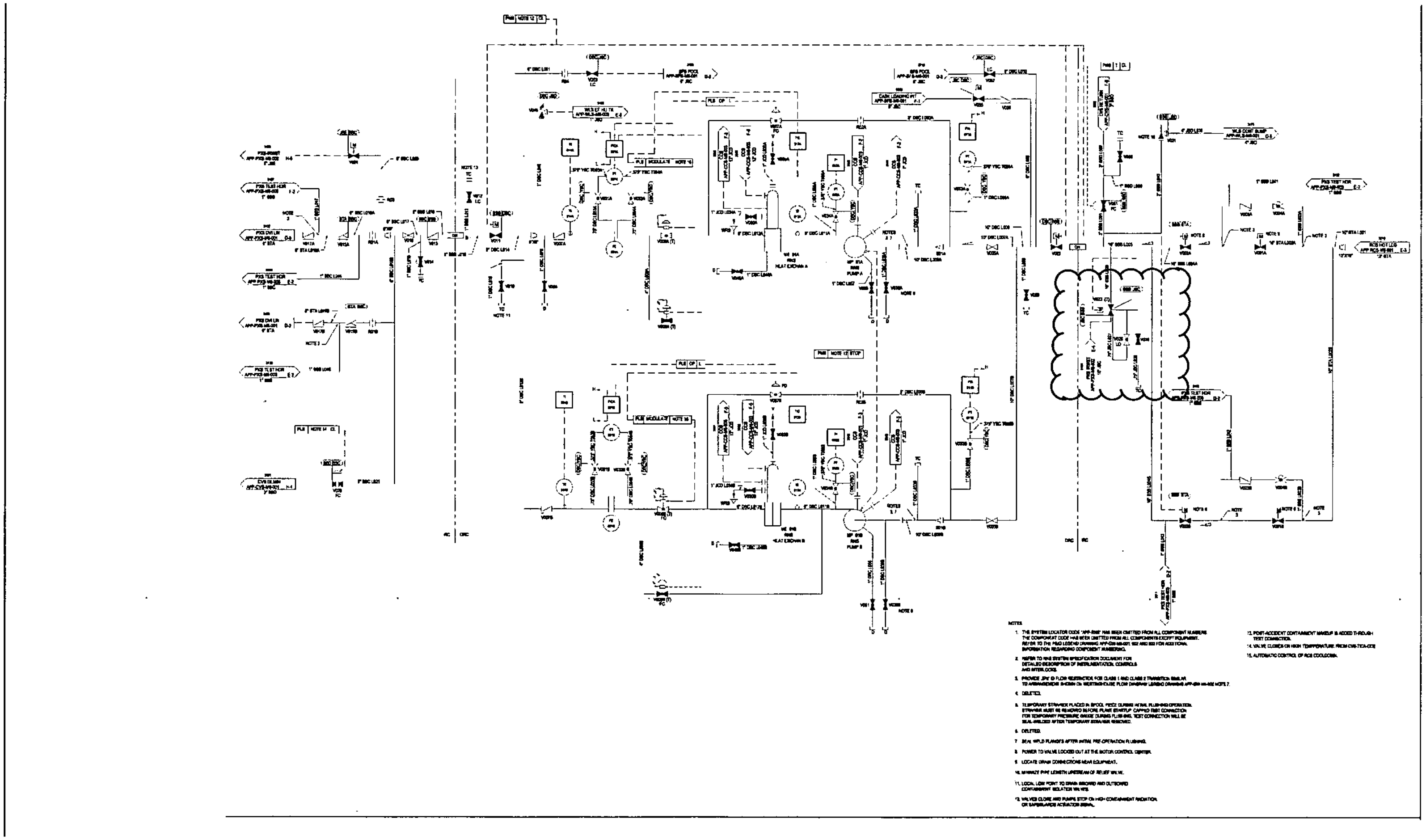


Figure 5.4-7

Normal Residual Heat Removal System Piping and Instrument Diagram

Table 6.2.3-1 (Sheet 2 of 4)

CONTAINMENT MECHANICAL PENETRATIONS AND ISOLATION VALVES

	Containment Penetration			Isolation Device						Test		
System	Line	Flow	Closed Sys IRC	Valve/Hatch Identification	Pipe Length	DCD Subsection	Position N-S-A	Signal	Closure Times	Type[1] & Note	Medium	Direction
PXS	N_2 to accumulators	In	No	PXS-PL-V042	2	6.3	O-O-C	T	std.	C	Air	Forward
				PXS-PL-V043	-		C-C-C	None	N/A			
RNS	RCS to RHR pump	Out	No	RNS-PL-V002A/B	1	5.4.7	C-O-C	HR, S	std.	C	Air	Forward
				RNS-PL-V023	-	5.4.7	C-O-C	HR, S	std.	C		Forward
				RNS-PL-V022	42	5.4.7	C-O-C	HR, S	std.	C,4		Forward
				RNS-PL-V021	-	5.4.7	C-C-C	None	N/A	C		Forward
				RNS-PL-V061	-	5.4.7	C-O-C	T	std.	C		Forward
				PXS-PL-V208A	-	6.3	C-C-C	None	N/A	C		Forward
	RHR pump to RCS	In	No	RNS-PL-V011	25	5.4.7	C-O-C	HR, S	std.	C,4	Air	Forward
				RNS-PL-V013	-		C-O-C	None	N/A	C,4		
SFS	IRWST/Ref. cav. SFP pump discharge	In	No	SFS-PL-V038	20	9.1.3	C-O-C	T	std.	C,5	Air	Forward
				SFS-PL-V037	-		C-O-C	None	N/A			
	IRWST/Ref. cav. purif. out	Out	No	SFS-PL-V035	31	9.1.3	C-O-C	T	std.	C,5	Air	Forward
				SFS-PL-V034	-		C-O-C	T	std.			
				SFS-PL-V067	-		C-C-C	None	N/A			
SGS	Main steam line 01	Out	Yes	SGS-PL-V040A	29	10.3	O-C-C	MS	5 sec	A,2	N2	Forward
				SGS-PL-V027A(7)	67		O-O-C	LSL	std.			
				SGS-PL-V030A,31A,32A,33A,34A,35A	11, 14, 18, 21, 23, 27		C-C-C	None	N/A			
				SGS-PL-V036A	39		O-O-C	MS	std.			
				SGS-PL-V240A	44		C-C-C	MS	std.			
	Main steam line 02	Out	Yes	SGS-PL-V040B	29	10.3	O-C-C	MS	5 sec	A,2	N2	Forward
				SGS-PL-V027B(7)	67		O-O-C	LSL	std.			
				SGS-PL-V030B,31B,32B,33B,34B,35B	11, 14, 18, 21, 23, 27		C-C-C	None	N/A			
				SGS-PL-V036B	39		O-O-C	MS	std.			
				SGS-PL-V240B	44		C-C-C	MS	std.			
	Main feedwater 01	In	Yes	SGS-PL-V057A	23	10.3	O-C-C	MF	5 sec	A,2	H2O	Forward
	Main feedwater 02	In	Yes	SGS-PL-V057B	23	10.3	O-C-C	MF	5 sec	A,2	H2O	Forward
	SG blowdown 01	Out	Yes	SGS-PL-V074A	14	10.3	O-O-C	PRHR	std.	A,2	H2O	Forward
	SG blowdown 02	Out	Yes	SGS-PL-V074B	13	10.3	O-O-C	PRHR	std.	A,2	H2O	Forward
	Startup feedwater 01	In	Yes	SGS-PL-V067A	28	10.3	C-O-C	LTC, SGL	std.	A,2	H2O	Forward
	Startup feedwater 02	In	Yes	SGS-PL-V067B	27	10.3	C-O-C	LTC, SGL	std.	A,2	H2O	Forward

Table 6.2.3-1 (Sheet 4 of 4)

CONTAINMENT MECHANICAL PENETRATIONS AND ISOLATION VALVES

Explanation of Heading and Acronyms for Table 6.2.3-1

System:	Fluid system penetrating containment	
Containment Penetration:	These fields refer to the penetration itself	
Line:	Fluid system line	
Flow:	Direction of flow in or out of containment	
Closed Sys IRC:	Closed system inside containment as defined in DCD Section 6.2.3.1.1	
Isolation Device:	These fields refer to the isolation devices for a given penetration	
Valve/Hatch ID:	Identification number on P&ID or system figure	
Pipe Length:	Nominal length of pipe to outboard containment isolation valve, feet	
Subsection Containing Figure:	Safety analysis report containing the system P&ID or figure	
Position N-S-A:	Device position for N (normal operation)	
	S	(shutdown)
	A	(post-accident)
Signal:	Device closure signal	
	MS:	Main steam line isolation
	LSL:	Low steam line pressure
	MF:	Main feedwater isolation
	LTC:	Low T_{cold}
	PRHR:	Passive residual heat removal actuation
	T:	Containment isolation
	S:	Safety injection signal
	HR:	High containment radiation
	DAS:	Diverse actuation system signal
	PL2:	High 2 pressurizer level signal
	S+PL1:	Safety injection signal plus high 1 pressurizer level
	SGL:	High steam generator level

Closure Time:		
Required valve closure stroke time		
std:	Industry standard for valve type (≤ 60 seconds)	
N/A:	Not Applicable	
Test:	These fields refer to the penetration testing requirements	
Type:	Required test type	
	A:	Integrated Leak Rate Test
	B:	Local Leak Rate Test -- penetration
	C:	Local Leak Rate Test -- fluid systems
Note:	See notes below	
Medium:	Test fluid on valve seat	
Direction:	Pressurization direction	
	Forward:	High pressure on containment side
	Reverse:	High pressure on outboard side

Notes:

1. Containment leak rate tests are designated Type A, B, or C according to 10CFR50, Appendix J.
2. The secondary side of the steam generator, including main steam, feedwater, startup feedwater, blowdown and sampling piping from the steam generators to the containment penetration, is considered an extension of the containment. These systems are not part of the reactor coolant pressure boundary and do not open directly to the containment atmosphere during post-accident conditions. During Type A tests, the secondary side of the steam generators is vented to the atmosphere outside containment to ensure that full test differential pressure is applied to this boundary.
3. The central chilled water system remains water-filled and operational during the Type A test in order to maintain stable containment atmospheric conditions.
4. The containment isolation valves for this penetration are open during the Type A test to facilitate testing. Their leak rates are measured separately.
5. The inboard valve flange is tested in the reverse direction.
6. ~~These valves are not subject to a Type C test. Upstream side of RNS hot leg suction isolation valves is not vented during local leak rate test to retain double isolation of RCS at elevated pressure. Valve is flooded during post accident operation.~~ Not used.
7. Refer to DCD Table 15.0-4b for PORV block valve closure time.

Change Number 3

Table 3.9-16 (Sheet 10 of 23)

VALVE INSERVICE TEST REQUIREMENTS

Valve Tag Number	Description(1)	Valve/Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME Class/ IST Category	Inservice Testing Type and Frequency	IST Notes
PXS-PL-V043	Nitrogen Supply Containment Isolation IRC	Check	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	Class 2 Category AC	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test Check Exercise/Quarterly	27
PXS-PL-V101	PRHR HX Inlet Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years	
PXS-PL-V108A	PRHR HX Control	Remote AO ~~GLOBE~~Ball	Maintain Open Transfer Open	Active-to-Failed Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test	31
PXS-PL-V108B	PRHR HX Control	Remote AO ~~GLOBE~~Ball	Maintain Open Transfer Open	Active-to-Failed Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test	31
PXS-PL-V117A	Containment Recirculation A Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 3 Category B	Remote Position Indication, Exercise/2 Years	31
PXS-PL-V117B	Containment Recirculation B Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 3 Category B	Remote Position Indication, Exercise/2-Years	
PXS-PL-V118A	Containment Recirculation A Isolation	Squib	Maintain Open Maintain Close Transfer Open	Active Remote Position	Class 3 Category D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5
PXS-PL-V118B	Containment Recirculation B Isolation	Squib	Maintain Open Maintain Close Transfer Open	Active Remote Position	Class 3 Category D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5
PXS-PL-V119A	Containment Recirculation A Check	Check	Maintain Open Maintain Close Transfer Open Transfer Close	Active Remote Position	Class 3 Category BC	Remote Position Indication, Exercise/2 Years Check-Initial Open Differential Pressure/Refueling~~2 Years~~ Check Exercise/Refueling Shutdown	11
PXS-PL-V119B	Containment Recirculation B Check	Check	Maintain Open Maintain Close Transfer Open Transfer Close	Active Remote Position	Class 3 Category BC	Remote Position Indication, Exercise/2 Years Check-Initial Open Differential Pressure/Refueling~~2 Years~~ Check Exercise/Refueling Shutdown	11
PXS-PL-V120A	Containment Recirculation A Isolation	Squib	Maintain Open Maintain Close Transfer Open	Active Remote Position	Class 3 Category D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5
PXS-PL-V120B	Containment Recirculation B Isolation	Squib	Maintain Open Maintain Close Transfer Open	Active Remote Position	Class 3 Category D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5

coolant loops. During normal operation, the core makeup tanks are completely full of cold, borated water. The boration capability of these tanks provides adequate core shutdown margin following a steam line break.

The core makeup tanks are connected to the reactor coolant system through a discharge injection line and an inlet pressure balance line connected to a cold leg. The discharge line is blocked by two normally closed, parallel air-operated isolation valves that open on a loss of air pressure or electrical power, or on control signal actuation. The core makeup tank discharge isolation valves are diverse from the passive residual heat removal heat exchanger outlet isolation valves discussed above. They use different ~~globe~~ valve body styles and different air operator types.

The pressure balance line from the cold leg is normally open to maintain the core makeup tanks at reactor coolant system pressure, which prevents water hammer upon initiation of core makeup tank injection.

The cold leg pressure balance line is connected to the top of the cold leg and is routed continuously upward to the high point near the core makeup tank inlet. The normal water temperature in this line will be hotter than the discharge line.

The outlet line from the bottom of each core makeup tank provides an injection path to one of the two direct vessel injection lines, which are connected to the reactor vessel downcomer annulus. Upon receipt of a safeguards actuation signal, the two parallel valves in each discharge line open to align the associated core makeup tank to the reactor coolant system.

There are two operating processes for the core makeup tanks, steam-compensated injection and water recirculation. During steam-compensated injection, steam is supplied to the core makeup tanks to displace the water that is injected into the reactor coolant system. This steam is provided to the core makeup tanks through the cold leg pressure balance line. The cold leg line only has steam flow if the cold legs are voided.

During water recirculation, hot water from the cold leg enters the core makeup tanks, and the cold water in the tank is discharged to the reactor coolant system. This results in reactor coolant system boration and a net increase in reactor coolant system mass.

The operating process for the core makeup tanks depends on conditions in the reactor coolant system, primarily voiding in the cold leg. When the cold leg is full of water, the cold leg pressure balance line remains full of water and the injection occurs via water recirculation. If reactor coolant system inventory decreases sufficiently to cause cold leg voiding, then steam flows through the cold leg balance lines to the core makeup tanks.

Following an event such as steam-line break, the reactor coolant system experiences a decrease in temperature and pressure due to an increase of energy removed by the secondary system as a consequence of the break. The cooldown results in a reduction of the core shutdown margin due to the negative moderator temperature coefficient. There is a potential return to power, assuming the most reactive rod cluster control assembly is stuck in its fully withdrawn position. The actuation of the core makeup tanks following this event provides injection of borated water via water recirculation to mitigate the reactivity transient and provide the required shutdown margin.

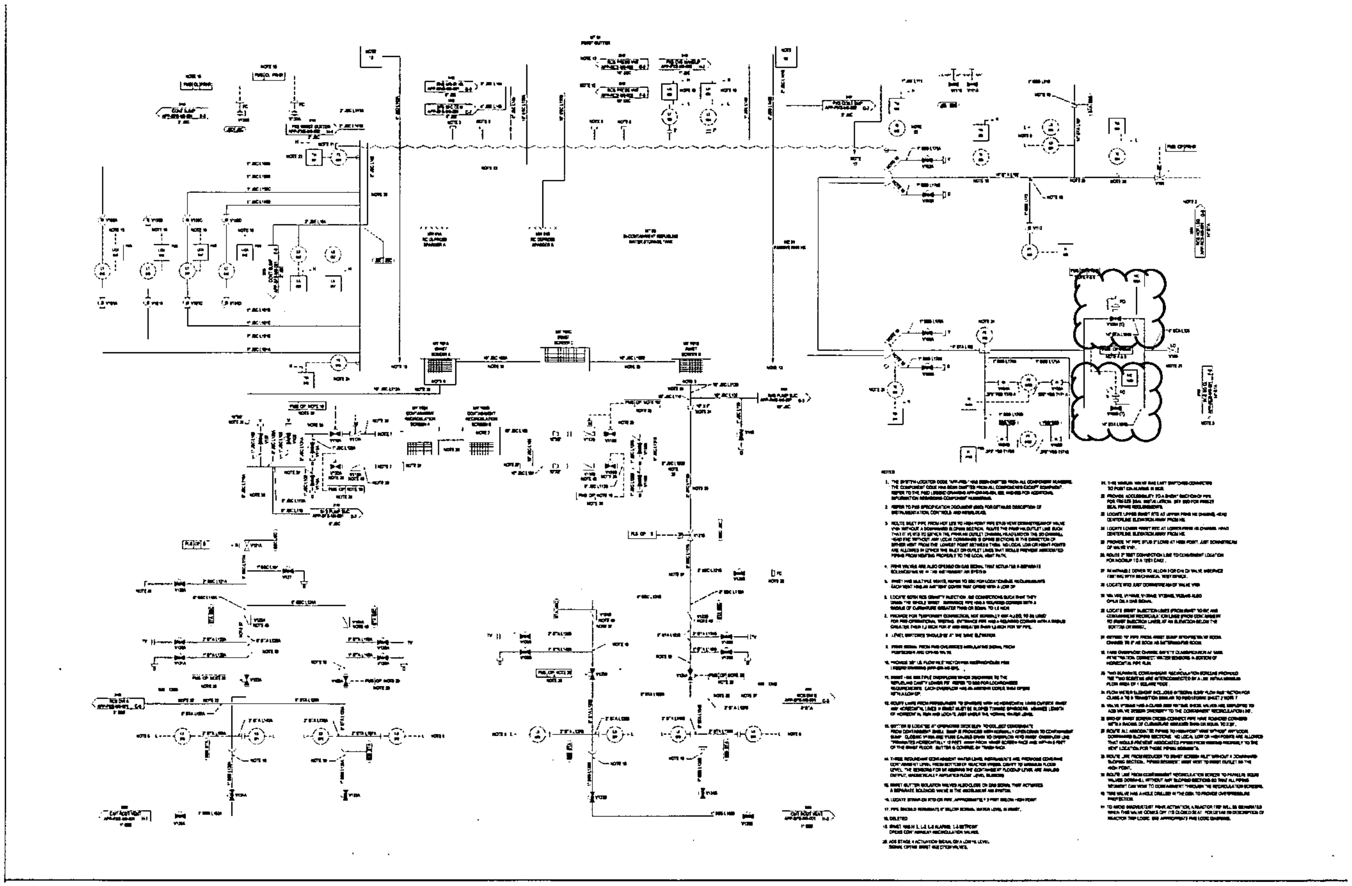
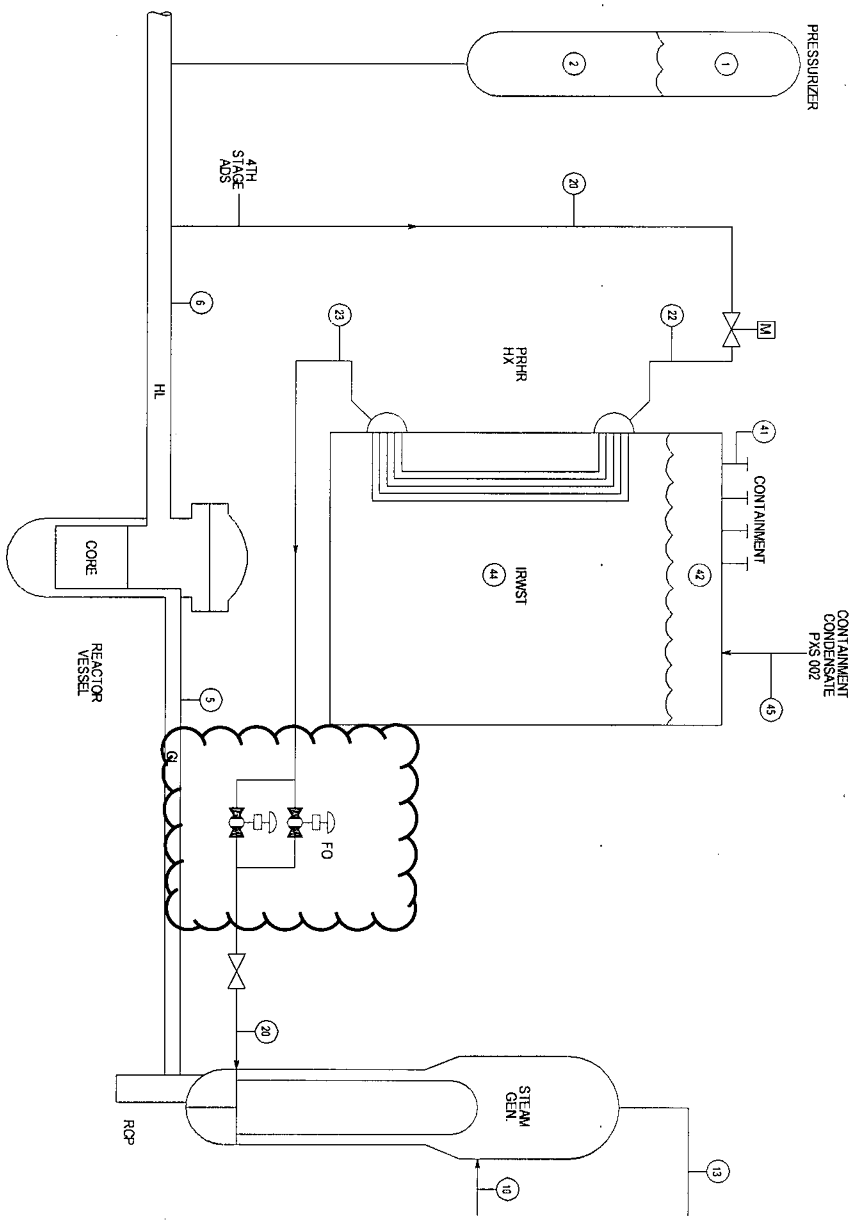


Figure 6.3-2

Passive Core Cooling System
Piping and Instrumentation Diagram (Sheet 2)



Inside Reactor Containment

Figure 6.3-4

Passive Decay Heat Removal (REF) RCS & PXS

Change Number 4

Table 3.2-3 (Sheet 22 of ~~65~~68)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V231A	CMT A Fill Check	B	I	ASME III-2	
PXS-PL-V231B	CMT B Fill Check	B	I	ASME III-2	
PXS-PL-V232A	Accumulator A Fill/Drain Isolation	C	I	ASME III-3	
PXS-PL-V232B	Accumulator B Fill/Drain Isolation	C	I	ASME III-3	
PXS-PL-V250A	CMT A Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V250B	CMT B Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V251A	CMT A Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V251B	CMT B Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V252A	CMT A Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V252B	CMT B Check Valve Test Valve	A	I	ASME III-1	
PXS-PY-C01	Nitrogen Makeup Containment Penetration	B	I	ASME III, 2	
Balance of system components are Class E					
Reactor Coolant System (RCS)					Location: Containment
RCS-MB-01	Steam Generator 1	A	I	ASME III-1	
RCS-MB-02	Steam Generator 2	A	I	ASME III-1	
RCS-MP-01A/B	SG 1A(B) Reactor Coolant Pump	A	I	ASME III-1	Pump Motor – Class D
n/a	Rotor Shaft	C	I	Manufacturer Std	
n/a	Impeller	C	I	Manufacturer Std	
n/a	Flywheel	C	I	Manufacturer Std	
n/a	RCP Heat Exchanger (Tube Side)	A	I	ASME III-1	Shellside – Class D, ASME VIII, Div. 1

Table 3.11-1 (Sheet 38 of 50)

ENVIRONMENTALLY QUALIFIED ELECTRICAL AND MECHANICAL EQUIPMENT

Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Core Makeup Tank A Fill Isolation	PXS-PL-V230A	1	PB	1 yr	M *
Core Makeup Tank B Fill Isolation	PXS-PL-V230B	1	PB	1 yr	M *
Core Makeup Tank A Fill Check	PXS-PL-V231A	1	PB	1 yr	M *
Core Makeup Tank B Fill Check	PXS-PL-V231B	1	PB	1 yr	M *
Accumulator A Fill/Drain Isolation	PXS-PL-V232A	1	PB	1 yr	M *
Accumulator B Fill/Drain Isolation	PXS-PL-V232B	1	PB	1 yr	M *
CMT A Check Valve Test Valve	PXS-PL-V250A	1	PB	1 yr	M *
CMT B Check Valve Test Valve	PXS-PL-V250B	1	PB	1 yr	M *
CMT A Check Valve Test Valve	PXS-PL-V251A	1	PB	1 yr	M *
CMT B Check Valve Test Valve	PXS-PL-V251B	1	PB	1 yr	M *
CMT A Check Valve Test Valve	PXS-PL-V252A	1	PB	1 yr	M *
CMT B Check Valve Test Valve	PXS-PL-V252B	1	PB	1 yr	M *
ADS Test Valve	RCS-PL-V007A	1	PB	1 yr	M *
ADS Test Valve	RCS-PL-V007B	1	PB	1 yr	M *
Fourth Stage ADS Isolation Limit Switch Motor Operator	RCS-PL-V014A RCS-PL-V014A-L RCS-PL-V014A-M	1 1 1	PB PAMS ESF	1 yr 1 yr 24 hr	M * E * E *
Fourth Stage ADS Isolation Limit Switch Motor Operator	RCS-PL-V014B RCS-PL-V014B-L RCS-PL-V014B-M	1 1 1	PB PAMS ESF	1 yr 1 yr 24 hr	M * E * E *
Fourth Stage ADS Isolation Limit Switch Motor Operator	RCS-PL-V014C RCS-PL-V014C-L RCS-PL-V014C-M	1 1 1	PB PAMS ESF	1 yr 1 yr 24 hr	M * E * E *
Fourth Stage ADS Isolation Limit Switch Motor Operator	RCS-PL-V014D RCS-PL-V014D-L RCS-PL-V014D-M	1 1 1	PB PAMS ESF	1 yr 1 yr 24 hr	M * E * E *
Hot Leg 2 Level Instrument Root	RCS-PL-V095	1	PB	1 yr	M *
Hot Leg 2 Level Instrument Root	RCS-PL-V096	1	PB	1 yr	M *
Hot Leg 1 Level Instrument Root	RCS-PL-V097	1	PB	1 yr	M *
Hot Leg 1 Level Instrument Root	RCS-PL-V098	1	PB	1 yr	M *
Hot Leg 1 Flow Instrument Root	RCS-PL-V101A	1	PB	1 yr	M *
Hot Leg 1 Flow Instrument Root	RCS-PL-V101B	1	PB	1 yr	M *
Hot Leg 1 Flow Instrument Root	RCS-PL-V101C	1	PB	1 yr	M *

Table 3.9-16 (Sheet 8 of 23)

VALVE INSERVICE TEST REQUIREMENTS

Valve Tag Number	Description[1]	Valve/Actuator Type	Safety-Related Missions	Safety Functions[2]	ASME Class/ IST Category	Inservice Testing Type and Frequency	IST Notes
PSS-PL-V046	Air Sample Line Containment Isolation ORC	Remote AO GLOBE	Maintain Close Transfer Close	Active-to-Failed Containment Isolation Safety Seat Leakage Remote Position	Class 2 Category A	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test Exercise Full Stroke/Quarterly Operability Test	27, 31
PWS-PL-V418	PWS MCR Isolation Valve	Manual	Transfer Close Maintain Close	Active	Class 3 Category B	Exercise Full Stroke/2 Years	
PWS-PL-V420	PWS MCR Isolation Valve	Manual	Transfer Close Maintain Close	Active	Class 3 Category B	Exercise Full Stroke/2 Years	
PWS-PL-V498	PWS MCR Vacuum Breaker	Check	Transfer Open Transfer Close Maintain Close	Active	Class 3 Category C	Check Exercise/Quarterly	
~~PWS-PL-V420~~	~~PWS MCR Isolation Valve~~	~~Manual~~	~~Transfer Open~~ ~~Maintain Close~~	~~Active~~	~~Category B~~	~~Exercise Full Stroke/2 Years~~	
PXS-PL-V002A	Core Makeup Tank A Cold Leg Inlet Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years	
PXS-PL-V002B	Core Makeup Tank B Cold Leg Inlet Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years	
PXS-PL-V014A	Core Makeup Tank A Discharge Isolation	Remote AO GLOBE	Maintain Open Transfer Open	Active-to-Failed Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test	31
PXS-PL-V014B	Core Makeup Tank B Discharge Isolation	Remote AO GLOBE	Maintain Open Transfer Open	Active-to-Failed Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test	31
PXS-PL-V015A	Core Makeup Tank A Discharge Isolation	Remote AO GLOBE	Maintain Open Transfer Open	Active-to-Failed Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test	31
PXS-PL-V015B	Core Makeup Tank B Discharge Isolation	Remote AO GLOBE	Maintain Open Transfer Open	Active-to-Failed Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test	31
PXS-PL-V016A	Core Makeup Tank A Discharge Check	Check	Maintain Open Transfer Open Transfer Close	Active ~~Remote Position~~	Class 1 Category BC	~~Remote Position Indication, Exercise/2 Years~~ Check Exercise/Refueling Shutdown	10
PXS-PL-V016B	Core Makeup Tank B Discharge Check	Check	Maintain Open Transfer Open Transfer Close	Active ~~Remote Position~~	Class 1 Category BC	~~Remote Position Indication, Exercise/2 Years~~ Check Exercise/Refueling Shutdown	10

PXS-PL-V017A	Core Makeup Tank A Discharge Check	Check	Maintain Open Transfer Open Transfer Close	Active ~~Remote Position~~	Class 1 Category BC	~~Remote Position Indication, Exercise/2 Years~~ Check Exercise/Refueling Shutdown	10

Table 3.9-16 (Sheet 9 of 23)

VALVE INSERVICE TEST REQUIREMENTS

Valve Tag Number	Description[1]	Valve/Actuator Type	Safety-Related Missions	Safety Functions[2]	ASME Class/ IST Category	Inservice Testing Type and Frequency	IST Notes
PXS-PL-V017B	Core Makeup Tank B Discharge Check	Check	Maintain Open Transfer Open Transfer Close	Active ~~Remote Position~~	Class 1 Category BC	~~Remote Position Indication, Exercise/2 Years~~ Check Exercise/Refueling Shutdown	10
PXS-PL-V021A	Accumulator A Vent Isolation	Remote	Maintain Close	Remote Position	Category B	Remote Position Indication, Exercise/2 Years	
PXS-PL-V021B	Accumulator B Vent Isolation	Remote	Maintain Close	Remote Position	Category B	Remote Position Indication, Exercise/2 Years	
PXS-PL-V022A	Accumulator A Pressure Relief	Relief	Maintain Close Transfer Open Transfer Close	Active	Class 3 Category BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
PXS-PL-V022B	Accumulator B Pressure Relief	Relief	Maintain Close Transfer Open Transfer Close	Active	Class 3 Category BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
PXS-PL-V027A	Accumulator A Discharge Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years	
PXS-PL-V027B	Accumulator B Discharge Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years	
PXS-PL-V028A	Accumulator A Discharge Check	Check	Maintain Close Transfer Open Transfer Close	Active RCS Pressure Boundary ~~Remote Position~~ Safety Seat Leakage	Class 1 Category AC	~~Remote Position Indication, Exercise/2 Years~~ Check Exercise/Refueling Shutdown Pressure Isolation Leak Test/2 years	9
PXS-PL-V028B	Accumulator B Discharge Check	Check	Maintain Close Transfer Open Transfer Close	Active RCS Pressure Boundary ~~Remote Position~~ Safety Seat Leakage	Class 1 Category AC	~~Remote Position Indication, Exercise/2 Years~~ Check Exercise/Refueling Shutdown Pressure Isolation Leak Test/2 Years	9
PXS-PL-V029A	Accumulator A Discharge Check	Check	Maintain Close Transfer Open Transfer Close	Active RCS Pressure Boundary ~~Remote Position~~ Safety Seat Leakage	Class 1 Category AC	~~Remote Position Indication, Exercise/2 Years~~ Check Exercise/Refueling Shutdown Pressure Isolation Leak Test/Refueling Shutdown	9
PXS-PL-V029B	Accumulator B Discharge Check	Check	Maintain Close Transfer Open Transfer Close	Active RCS Pressure Boundary ~~Remote Position~~ Safety Seat Leakage	Class 1 Category AC	~~Remote Position Indication, Exercise/2 Years~~ Check Exercise/Refueling Shutdown Pressure Isolation Leak Test/Refueling Shutdown	9
PXS-PL-V042	Nitrogen Supply Containment Isolation ORC	Remote AO GLOBE	Maintain Close Transfer Close	Active-to-Failed Containment Isolation Safety Seat Leakage Remote Position	Class 2 Category A	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test Exercise Full Stroke/Quarterly Operability Test	27, 31

Notes:

1. Acronyms:

ADS	automatic depressurization system
CAS	compressed and instrument air system
CCS	component cooling water system
CVS	chemical and volume control system
DWS	demineralized water transfer and storage system
FPS	fire protection system
IRC	inside reactor containment
IRWST	in-containment refueling water storage tank
MSS	main steam system
MTS	main turbine system
ORC	outside reactor containment
PCCWST	passive containment cooling water storage tank
PCS	passive containment cooling system
PSS	primary sampling system
PWS	potable water system
PXS	passive core cooling system
RCS	reactor coolant system
RNS	normal residual heat removal system
SFS	spent fuel pool cooling system
SGS	steam generator system
VBS	nuclear island nonradioactive ventilation system
VES	main control room emergency habitability system
VFS	containment air filtration system
VWS	central chilled water system
WLS	liquid radwaste system
AO	air operated
MO	motor operated
SO	solenoid operated

2. Valves listed as having an active or an active-to-failed safety-related function provide the safety-related valve transfer capabilities identified in the safety-related mission column. Valves having an active-to-failed function will transfer to the position identified in the safety-related mission column on loss of motive power. Valves with an active-to-failed function shall be tested by observing the operation of the actuator upon loss of valve actuating power. This "fail-safe" requirement is not otherwise shown and is performed during exercise testing.

3. This note applies to the ADS stage 1/2/3 valves (RCS-V001A/B, V002A/B, V003A/B, V011A/B, V012A/B, V013A/B). These valves are normally closed to maintain the RCS pressure boundary. These valves have a safety-related function to open following LOCAs to allow safety injection from lower pressure water supplies (accumulators and IRWST). These valves also have beyond design basis functions to depressurize the RCS. These valves have the same design pressure as the RCS and are AP1000 equipment class A. Downstream of the second valve is a lower design pressure and is equipment class C. The discharge of these valves is open to the containment through the IRWST.

   Both ADS valves in each line are normally closed during normal reactor operation in accordance with 10 CFR 50.2 and ANS/ANSI 51.1. If one of these valves is opened, for example for testing, the RCS pressure boundary is not maintained in accordance with the criteria contained in these two documents. In addition, the ADS valve configuration is similar to the normal residual heat removal system suction valve configuration. Even though the RNS suction valve configuration includes a third valve in the high pressure portion of the line, and the first two RNS valves have safety-related functions to transfer closed, they are not stroke tested during normal reactor operation to avoid a plant configuration where the mispositioning of one valve would cause a LOCA. Note 15 describes the justification for testing the RNS valves during cold shutdown.

   These ADS valves are tested during cold shutdowns when the RCS pressure is reduced to atmospheric pressure so that mispositioning of a single valve during this IST will not cause a LOCA. ~~Testing these valves every cold shutdown is consistent with the AP1000 PRA which assumes more than 2 cold or refueling shutdowns per year.~~

4. This note applies to the reactor vessel head vent solenoid valves (RCS-V150A/B/C/D). Exercise testing of these valves at power represents a risk of loss of reactor coolant and depressurization of the RCS if the proper test sequence is not followed. Such testing may also result in the valves developing through seal leaks. Exercise testing of these valves will be performed at cold shutdown.

5. This note applies to squib valves in the RCS and the PXS. The squib valve charge is removed and test fired outside of valve. Squib valves are not exercised for inservice testing. Their position indication sensors will be tested by local inspection.

6. This note applies to the CVS isolation valves (CVS-V001, V002, V003, V080, V081, V082). Closing these valves at power will result in an undesirable temperature transient on the RCS due to the interruption of purification flow. Therefore, quarterly exercise testing will not be performed. Exercise testing will be performed at cold shutdown.

7. This note applies to the pressurizer safety valves (RCS-V005A/B) and to the main steam safety valves (SGS-V030A/B, V031A/B, V032A/B, V033A/B, V034A/B and V035A/B). Since these valves are not exercised for inservice testing, their position indication sensors are tested by local inspection without valve exercise.

8. This note applies to CVS valve (CVS-V081). The safety functions are satisfied by the check valve function of the valve.

9. This note applies to the PXS accumulator check valves (PXS-V028A/B, V029A/B). To exercise these valves, flow must be provided through these valves to the RCS. These valves are not exercised during power operations because the accumulators cannot provide flow to the RCS since they are at a lower pressure. In addition, providing flow to the RCS during power operation would cause undesirable thermal transients on the RCS. During cold shutdowns, a full flow stroke test is impractical because of the potential of adding significant water to the RCS, and lifting the RNS relief valve. There is also a risk of injecting nitrogen into the RCS. A partial stroke test is practical during longer cold shutdowns (≥48 hours in Mode 5). In this test, flow is provided from test connections, through the check valves and into the RCS. Sufficient flow is not available to provide a detectable obturator movement. Full stroke exercise testing of these valves is conducted during refueling shutdowns.

10. This note applies to the PXS CMT check valves (PXS-V016A/B, V017A/B). These check valves are biased open valves and are fully open during normal operation. ~~These valves will be verified to be open quarterly.~~ In order to exercise these check valves, significant reverse flow must be provided from the DVI line to the CMT. These valves are not tested during power operations because the test would cause undesirable thermal transients on the portion of the line at ambient temperatures and change the CMT boron concentration These valves are not exercised during cold shutdowns because of changes that would result in the CMT boron concentration. Because this parameter is controlled by Technical Specifications, this testing is impractical. These valves are exercised during refueling when the RCS boron concentration is nearly equal to the CMT concentration and the plant is in a mode where the CMTs are not required to be available by the Technical Specifications.

11. This note applies to the PXS containment recirculation check valves (PXS-V119A/B). Squib valves in line with the check valves prevent the use of IRWST water to test the valves. To exercise these check valves an operator must enter the containment, remove a cover from the recirculation screens, and insert a test device (a mechanical exerciser) into the recirculation pipe to push open the check valve. The test device is made to interface with the valve without causing valve damage. The test device incorporates loads measuring sensors to measure the initial opening and full open force. These valves are not exercised during power operations because of the need to enter highly radioactive areas and because during this test the recirculation screen is bypassed. These valves are not exercised during cold shutdown operations for the same reasons. These valves are exercised during refueling conditions when the recirculation lines are not required to be available by Technical Specifications LCOs 3.5.7 and 3.5.8 and the radiation levels are reduced.

Table 31.6-3 (Sheet 18 of 31~~2~~)

LIST OF AP1000 SAFETY-RELATED ELECTRICAL AND MECHANICAL EQUIPMENT NOT HIGH FREQUENCY SENSITIVE

Description	AP1000 Tag Number	Comment
Accumulator A Leak Test	PXS-PL-V201A	2
Accumulator B Leak Test	PXS-PL-V201B	2
Accumulator A Leak Test	PXS-PL-V202A	2
Accumulator B Leak Test	PXS-PL-V202B	2
RNS Discharge Leak Test	PXS-PL-V205A	2
RNS Discharge Leak Test	PXS-PL-V205B	2
RNS Discharge Leak Test	PXS-PL-V206	2
RNS Suction Leak Test	PXS-PL-V207A	2
RNS Suction Leak Test	PXS-PL-V207B	2
RNS Suction Leak Test	PXS-PL-V208A	2
Core Makeup Tank A Fill Isolation	PXS-PL-V230A	2
Core Makeup Tank B Fill Isolation	PXS-PL-V230B	2
Core Makeup Tank A Fill Check	PXS-PL-V231A	2
Core Makeup Tank B Fill Check	PXS-PL-V231B	2
Accumulator A Fill/Drain Isolation	PXS-PL-V232A	2
Accumulator B Fill/Drain Isolation	PXS-PL-V232B	2
CMT A Check Valve Test Valve	PXS-PL-V250A	2
CMT B Check Valve Test Valve	PXS-PL-V250B	2
CMT A Check Valve Test Valve	PXS-PL-V251A	2
CMT B Check Valve Test Valve	PXS-PL-V251B	2
CMT A Check Valve Test Valve	PXS-PL-V252A	2
CMT B Check Valve Test Valve	PXS-PL-V252B	2
ADS Test Valve	RCS-PL-V007A	2
ADS Test Valve	RCS-PL-V007B	2
Fourth Stage ADS Isolation	RCS-PL-V014A	2

A cleanliness program (refer to subsection 6.3.8.1) controls foreign debris introduced into the containment during maintenance and inspection operations. The Technical Specifications require visual inspections of the screens during every refueling outage.

The design of the containment recirculation screens reduces the chance of debris reaching the screens. The screens are orientated vertically such that debris settling out of the water will not fall on the screens. The protective plates described above provide additional protection to the screens from debris. A 2-foot-high debris curb is provided to prevent high density debris from being swept along the floor by water flow to the containment recirculation screens. The screen design provides the trash rack function. This is accomplished by the screens having a large surface area to prevent a single object from blocking a large portion of the screen and by the screens having a robust design to preclude an object from damaging the screen and causing by-pass. The screen prevents debris larger than 0.0625 inches~~125"~~ from being injected into the reactor coolant system and blocking fuel cooling passages. The screen is a type that has more surface area to accommodate debris that could be trapped on the screen. The design of the containment recirculation screens is further described in APP-GW-GLN-147 (Reference 4).

The screen flow area is conservatively designed, considering the operation of the normal residual heat removal system pumps, which produce a higher flow than the gravity driven IRWST injection/recirculation flows. As a result, when the normal residual heat removal system pumps are not operating there is even more margin in screen clogging.

6.3.2.2.8 Valves

Design features used to minimize leakage for valves in the passive core cooling system include:

- Packless valves are used for manual isolation valves that are 2 inches or smaller.
- Valves which are normally open, except check valves and those which perform control function, are provided with back seats to limit stem leakage.

6.3.2.2.8.1 Manual Globe, Gate, and Check Valves

Gate valves have backseats and external screw and yoke assemblies.

Globe valves, both "T" and "Y" styles, are full-ported with external screw and yoke construction.

~~Check valves are spring-loaded lift piston types for sizes 2 inches and smaller, and swing type for sizes 2.5 inches and larger.~~ Stainless steel check valves have no penetration welds other than the inlet, outlet, and bonnet. The check valve hinge is serviced through the bonnet.

The gasket of the stainless steel manual globe and gate valves is similar to those described in subsection 6.3.2.2.8.3 for motor-operated valves.

6.3.7.6 Valve Position Indication and Control

6.3.7.6.1 Valve Position Indication

Individual valve position is provided for the safety-related, remotely actuated valves listed in Table 6.3-1. In addition, valve position is provided for certain manually operated valves, as described in subsection 6.3.2.2.8.2, that can isolate redundant passive core cooling equipment, if mispositioned. The incontainment refueling water injection check valves~~,~~ and containment recirculation check valves~~, accumulator check valves, and the core makeup tank check valves~~ have nonintrusive position indication.

For certain passive core cooling system valves with position indication, alarms in the main control room are provided to alert the operators to valve mispositioning. For the passive residual heat removal heat exchanger discharge valves, valve position indication is used to initiate a reactor trip upon opening of these valves while the reactor is at power.

6.3.7.6.2 Valve Position Control

Valve controls are provided for remotely operated passive core cooling system valves. Table 6.3-1 provides a list of the passive core cooling system remotely operated valves. These remotely operated valves have controls in the main control room. ~~This table also provides references to specific sections in DCD Chapter 7 that provide additional descriptions of the valve controls.~~

6.3.7.6.2.1 Accumulator Motor-Operated Valve Controls

As part of the plant shutdown procedures, the operator is required to close the accumulator motor-operated valves. This prevents a loss of accumulator water inventory to the reactor coolant system when the reactor coolant system is depressurized. The valves are closed after the reactor coolant system has been depressurized to below the setpoint to block the safeguards actuation signal. The redundant pressure and level alarms on each accumulator function to alert the operator to close these valves, if any are inadvertently left open. Power is locked out after the valves are closed. During plant startup, the operator is directed by plant procedures to energize and open these valves prior to reaching the reactor coolant system pressure setpoint that unblocks the safeguards actuation signal. Redundant indication and alarms are available to alert the operator if a valve is inadvertently left closed once the reactor coolant system pressure increases beyond the setpoint. Power is also locked out after these valves are opened.

The accumulator isolation valves are not required to move during power operation. For a description of limiting conditions for operation and surveillance requirements of these valves, refer to the technical specifications. The accumulator isolation valves receive a safeguards actuation signal to confirm that they are open in the event of an accident. As a result of the power lock out, technical specifications, and the redundant position indication and alarms, the valve controls are non-safety-related.

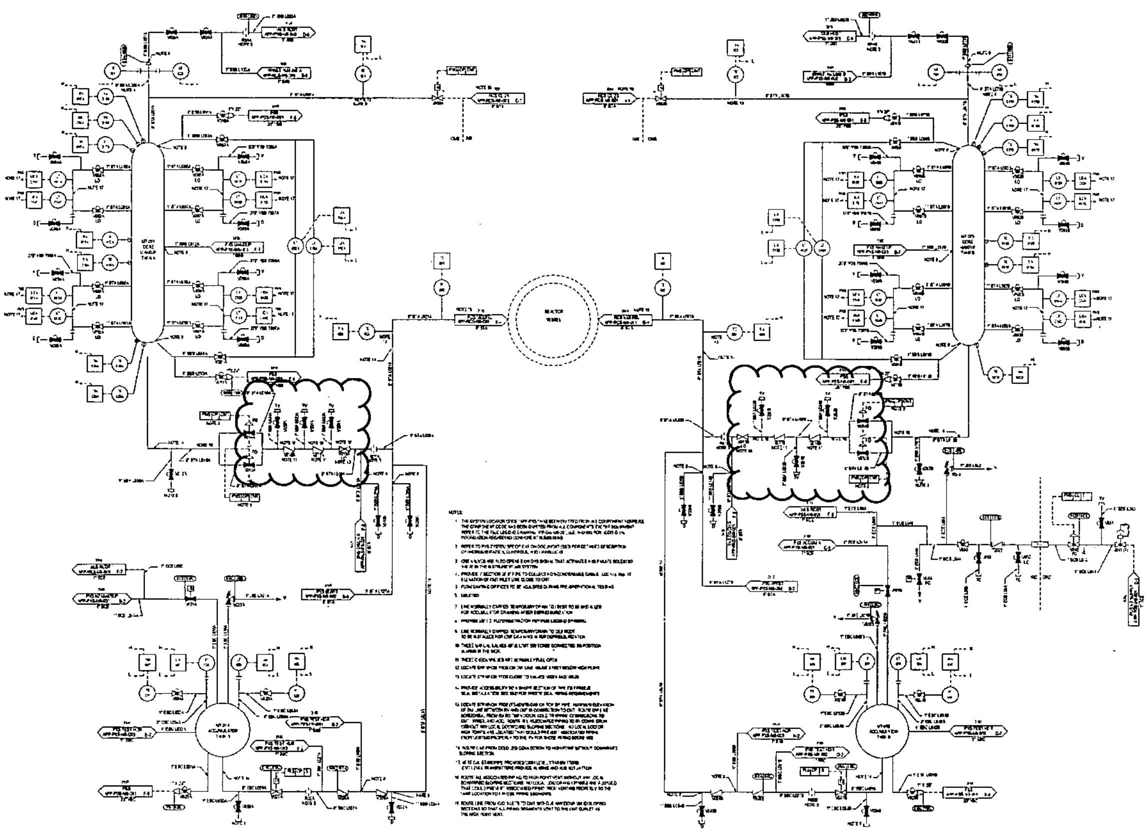


Figure 6.3-1

Passive Core Cooling System
Piping and Instrumentation Diagram (Sheet 1)

Table 6.3-3 (Sheet 1 of 4)

FAILURE MODE AND EFFECTS ANALYSIS - PASSIVE CORE COOLING SYSTEM COMPONENTS

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
CMT outlet isolation AOVs V014A/B, V015A/B Normally closed/ fail open	Failure to open on demand	All design basis events	No safety-related effect since each valve has a redundant, parallel isolation AOV, actuated by a separate division, which provides flow through a parallel branch line for the affected CMT. The other CMT is unaffected.	Valve position indication alarm in MCR and at RSW	
CMT discharge line check valves V016A/B, V017A/B Normally open	Failure to close on reverse flow	All design basis events	No safety-related effect since each valve has a redundant, series check valve which closes to prevent reverse flow, during a cold leg (large) LOCA or cold leg balance line break, preventing accumulator flow from bypassing the reactor vessel.	No ~~V~~valve position indication ~~alarm in MCR and at RSW~~	
Accumulator nitrogen supply/vent valves V021A/B, V045 Normally closed/ fail closed	Spurious opening	All design basis events	No safety-related effect since each valve has either a normally closed redundant, series isolation SOV or a check valve in each vent flow path, that prevents accumulator nitrogen from leaking out of the accumulator, which could degrade accumulator injection.	No valve position indication Accumulator low pressure alarm in MCR and at RSW	
Accumulator nitrogen supply containment isolation AOV V042 Normally open/ fail closed	Failure to close on demand	All design basis events	No safety-related effect since each valve has a redundant, series isolation check valve which independently closes on reverse flow in the line, preventing reactor coolant from leaking out of containment.	Valve position indication alarm in MCR and at RSW	
Accumulator nitrogen supply containment isolation check valve V043 Normally open	Failure to close on reverse flow	All design basis events	No safety-related effect since each valve has a redundant, series isolation AOV, actuated by a separate division, which closes to prevent reactor coolant from leaking out of containment.	No valve position indication	

Change Number 7

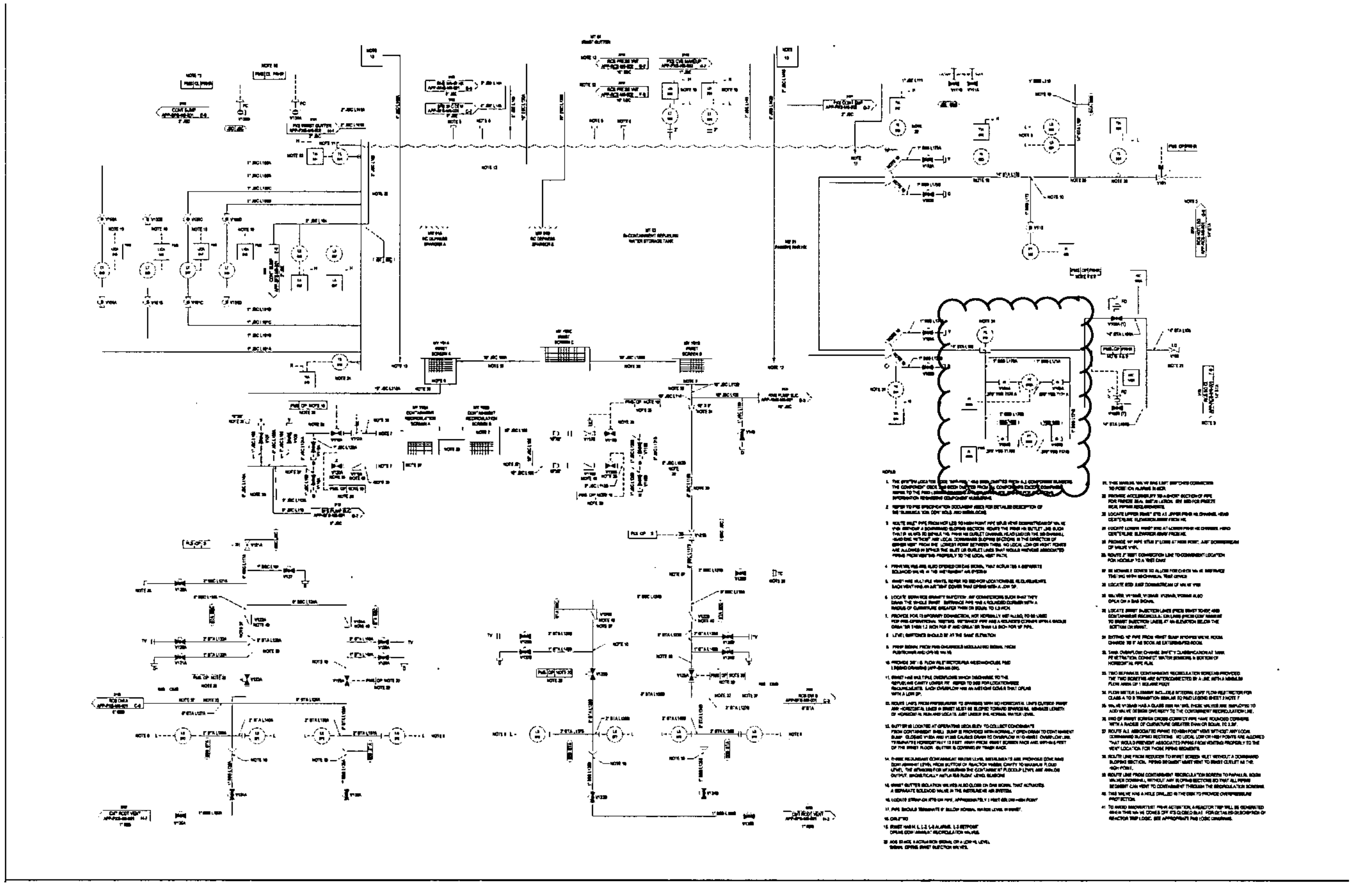


Figure 6.3-2

Passive Core Cooling System
Piping and Instrumentation Diagram (Sheet 2)

Change Number 10 A

7. The VES provides the following safety-related functions:

   a) The VES provides a 72-hour supply of breathable quality air for the occupants of the MCR.

   b) The VES maintains the MCR pressure boundary at a positive pressure with respect to the surrounding areas. There is a discharge of air through the MCR vestibule.

   c) The heat loads within the MCR, the I&C equipment rooms, and the Class 1E dc equipment rooms are within design basis assumptions to limit the heatup of the rooms identified in Table 2.2.5-4.

   d) The system provides a passive recirculation flow of MCR air to maintain main control room dose rates below an acceptable level during VES operation.

8. Safety-related displays identified in Table 2.2.5-1 can be retrieved in the MCR.

9. a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.2.5-1 to perform their active functions.

   b) The valves identified in Table 2.2.5-1 as having protection and safety monitoring system (PMS) control perform their active safety function after receiving a signal from the PMS.

10. After loss of motive power, the remotely operated valves identified in Table 2.2.5-1 assume the indicated loss of motive power position.

11. Displays of the parameters identified in Table 2.2.5-3 can be retrieved in the MCR.

12. The background noise level in the MCR does not exceed 65 dB(A) at the operator workstations when the VES is operating.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.2.5-54 specifies the inspections, tests, analyses, and associated acceptance criteria for the VES.

Change Number 10 B

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
			AP1000 uses a 6-hour completion time for the limiting conditions of operation associated with the loss of one dc power subsystem to be consistent with the guidance in C.1 for a conventional plant with the loss of one ac source. The 6-hour completion time is reasonable based on engineering judgment~~judgement~~ balancing the risks of operation without one dc subsystem against the risks of a forced shutdown. Additionally, the completion time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown, and if necessary, prepare and effect an orderly and safe shutdown.

Reg. Guide 1.94, Rev. 1, 4/76 – Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants

General	ANSI N45.2.5-1974	N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for the quality assurance program.

Reg. Guide 1.95 – Withdrawn

Reg. Guide 1.96, Rev. 1, 6/76 – Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants

General		N/A	Applies to boiling water reactors only.

Reg. Guide 1.97, Rev. 3, 5/83 – Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

General	ANS-4.5-1980	Conforms	The variables to be monitored are selected according to usage and need in the plant Emergency Response Guidelines. They are assigned design and qualification Category 1, 2, or 3 and classified as Type A, B, C, D, or E. Due to AP1000 specific design features, the selection of some plant-specific variables and their classifications and categories are different from those of this regulatory guide. For example, the use of the passive residual heat removal system as the safety grade heat sink allows steam generator wide range level to be category 2, not category 1 as specified in Regulatory Guide 1.97. The AP1000 has no Type A variables. See Section 7.5 for additional information. Since Category 3 instrumentation is not part of a safety-related system, it is not qualified to provide information when exposed to a post-accident adverse environment.

Table 1.1-1 (Sheet 1 of 4)

AP1000 DCD ACRONYMS

ac	Alternating Current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ALARA	As-Low-As-Reasonably Achievable
ALWR	Advanced Light Water Reactor
AMCA	Air Movement and Control Association
ANS	American Nuclear Society
ANL	Argonne National Laboratory
ANSI	American National Standards Institute
API	American Petroleum Institute
ARI	Air Conditioning and Refrigeration Institute
ASCE	American Society of Civil Engineers
ASHRAE	American Society of Heating, Refrigeration and Air-Conditioning Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
AWS	American Welding Society
BEACON	Best Estimate Analyzer for Core Operations - Nuclear
BOL	Beginning of Life
BOP	Balance of Plant
BTP	Branch Technical Position
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CMAA	Crane Manufacturers Association of America
CMT	Core Makeup Tank
CRD	Control Rod Drive
COL	Combined Operating License/Combined License
CRDM	Control Rod Drive Mechanism
CSA	Control Support Area
CVS	Chemical and Volume Control System
DAC	Design Acceptance Criteria
dc	Direct Current
DBA	Design Basis Accident
DBE	Design Basis Event
DCD	Design Control Document

Table 1.1-1 (Sheet 2 of 4)

AP1000 DCD ACRONYMS

D-EHC	Digital Electrohydraulic Control
DEMA	Diesel Engine Manufacturers Association
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DOE	Department of Energy
DPU	Distributed Processing Unit
EFPD	Effective Full Power Days
EIS	Environmental Impact Statement
EMI	Electromagnetic Interference
EOF	Emergency Offsite Facility
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
ER	Environmental Report
ERF	Emergency Response Facility
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
FID	Fixed Incore Detector
FM	Factory Mutual Engineering and Research Corporation
FMEA	Failure Modes and Effects Analysis
FWPCA	Federal Water Pollution Control Act
GDC	General Design Criteria/Criterion
GRCA	Gray Rod Cluster Assembly
GSI	Generic Safety Issues
HEPA	High Efficiency Particulate Air
HFE	Human Factors Engineering
HVAC	Heating, Ventilation and Air Conditioning
I&C	Instrumentation and Control
ICEA	Insulated Cable Engineers Association
IDCOR	Industry Degraded Core Rulemaking
IEEE	Institute of Electrical and Electronics Engineers
IES	Illumination Engineering Society
ILRT	Integrated Leak Rate Test
INEL	Idaho National Engineering Laboratory
I/O	Input/Output
IRWST	In-c~~C~~ontainment Refueling Water Storage Tank
ISA	Instrument Society of America
ISI	Inservice Inspection
IST	Inservice Testing

Table 1.1-1 (Sheet 3 of 4)

AP1000 DCD ACRONYMS

ITAAC	Inspections, Tests, Analyses and Acceptance Criteria
LBB	Leak-Before-Break
LOCA	Loss of Coolant Accident
LOF	Loss-of-Flow with Failure to Scram
LOFT	Loss of Flow Test
LOOP	Loss of Offsite Power
LOSP	Loss of System Pressure with Degraded ECCS Operation
LPZ	Low Population Zone
LSB	Last Stage Blade
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Programs
MCC	Motor Control Center
MCR	Main Control Room
MCRHS	Main Control Room Habitability System
MFCV	Main Feedwater Control Valve
MFIV	Main Feedwater Isolation Valve
M-MIS	Man-Machine Interface System
MOV	Motor-operated Valves
MPC	Maximum Permissible Concentration
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MTBE(F)	Mean Time Between Event (Failure)
MW	Megawatt
MWe	Megawatt, electric
MWt	Megawatt, thermal
NAE	National Academy of Engineering
NAS	National Academy of Sciences
NBS	National Bureau of Standards
NEC	National Electrical Code
NEI	Nuclear Energy Institute
NEMA	National Electrical Manufacturers Association
NFPA	National Fire Protection Association
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resources Council (Superseded by NEI)
NUREG	Report designator for NRC reports

Table 1.6-1 (Sheet 4 of 20)

MATERIAL REFERENCED

DCD Section Number	Westinghouse Topical Report Number	Title
3.7	WCAP 7921-AR	Damping Values of Nuclear Power Plant Components, May 1974
	WCAP-9903 (P)	Justification of the Westinghouse Equivalent Static Analysis Method for Seismic Qualification of Nuclear Power Plant Auxiliary Mechanical Equipment, August 1980
3.8	WCAP-13891 (P) WCAP-14095	AP600 Automatic Depressurization System Phase A Test Data Report, May 1994
	WCAP-14324 (P) WCAP-14325	Final Data Report for ADS Phase B1 Tests, April 1995
	WCAP-15613 (P) WCAP-15706	AP1000 PIRT and Scaling Assessment, March 2001
3.9	WCAP-7765-AR	Westinghouse PWR Internals Vibrations Summary Three-Loop Internals Assurance, November 1973
	WCAP-8766 (P) WCAP-8780	Verification of Neutron Pad and 17x17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant, May 1976
	WCAP-8516-P (P) WCAP-8517	UHI Plant Internals Vibrations Measurement Program and Pre- and Post-Hot Functional Examinations, March 1975
	WCAP-10846 (P)	Doel 4 Reactor Internals Flow-Induced Vibration Measurement Program, March 1985
	WCAP-10865 (P) WCAP-10866	South Texas Plant (TGX) Reactor Internals Flow-Induced Vibration Assessment, February 1985
	WCAP-8708-P-A (P) Volumes 1 and 2 WCAP-8709-A Volumes 1 and 2	MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics, February 1976
	WCAP-8446 (P) WCAP-8449	17x17 Drive Line Components Tests – Phase 1B 11, 111 D-Loop Drop and Deflection, December 1974
	WCAP-9693 (P)	Investigation of Feedwater Line Cracking in Pressurized Water Reactor Plants, June 1980
	WCAP-15949-P (P) WCAP-15949-NP	AP1000 Reactor Internals Flow-Induced Vibration Assessment Program, Revision 2~~1~~, July 2003

(P) Denotes Document is Proprietary

evaluated for the safe shutdown earthquake loading using the rules of ASME III, Subsection NF.

- *Piping within the impact evaluation zone plus one transverse support in each transverse direction are evaluated to Equation 9 of ASME Code, Section III, Class 3, with a stress limit equal to the smaller of 4.5 S_h and 3.0 S_y. Outside the impact evaluation zone, the nonseismic piping meets ASME/ANSI B31.1 requirements.*

- *The nonseismic piping and seismic Category II supports are designed for loads from the nonseismic piping beyond the impact evaluation zone. This includes three plastic moment components (M_{p1}, M_{p2}, or M_{p3}) in each of three local coordinate directions applied at the first and last seismic Category II support. The responses to the three moments are evaluated independently. The response from the moments applied at the first seismic Category II support is combined with the response from the moments applied at the last seismic Category II support and with the responses to seismic anchor motions and equivalent static seismic inertia of the piping system by the absolute sum method. The support and anchor loads due to the plastic moments (M_{p1}, M_{p2}, or M_{p3}) of the seismically analyzed and supported section can be reduced if the elbow/bend resultant moments have exceeded the plastic limit moments of the elbow/bend. The value of the reduction factor RF is the same as the value for connected seismic Category II piping described above.*

- *The piping segment identified as the source has at least one effective axial support.*

- *Adequate free space between a source and a target is checked adding absolutely the piping safe shutdown earthquake deflections (defined following seismic Category II piping analysis methodology) and the safe shutdown earthquake target deflection. Also included are the displacements associated with the appropriate load cases.*

- *When the anchor is an equipment nozzle, the equipment is supported as seismic Category II as described in subsection 3.7.3.13.3.*]*

3.7.3.14 Seismic Analyses for Reactor Internals

See subsection 3.9.2 for the dynamic analyses of reactor internals.

3.7.3.15 Analysis Procedure for Damping

Damping values used in the seismic analyses of subsystems are presented in subsection 3.7.1.3. Safe shutdown earthquake damping values used for different types of analysis are provided in Table 3.7.1-1. For subsystems that are composed of different material types, the composite modal damping approach with the weighted stiffness method is used to determine the composite modal damping value. Alternately, the minimum damping value may be used for these systems. [*Composite modal damping for coupled building and piping systems is used for piping systems that are coupled to the primary coolant loop system and the interior concrete building. Composite modal damping is used for piping systems that are coupled to flexible equipment or flexible valves. Piping systems analyzed by the uniform envelope response spectra method with rigid*

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

The steel form modules are designed as steel structures according to the requirements of AISC-N690. This code is applicable since the form modules are constructed entirely out of structural steel plates and shapes and the applied loads are resisted by the steel elements.

3.8.3.5.7 Design Summary Report

A design summary report is prepared for containment internal structures documenting that the structures meet the acceptance criteria specified in subsection 3.8.3.5.

Deviations from the design due to as-procured or as-built conditions are acceptable based on an evaluation consistent with the methods and procedures of Section 3.7 and 3.8 provided the following acceptance criteria are met.

- The structural design meets the acceptance criteria specified in Section 3.8
- The seismic floor response spectra meet the acceptance criteria specified in subsection 3.7.5.4

Depending on the extent of the deviations, the evaluation may range from documentation of an engineering judgment~~judgement~~ to performance of a revised analysis and design. The results of the evaluation will be documented in an as-built summary report.

3.8.3.5.8 Design Summary of Critical Sections

3.8.3.5.8.1 Structural Wall Modules

[*This subsection summarizes the design of the following critical sections:*

- *South west wall of the refueling cavity (4′0″ thick)South wall of west steam generator cavity (2′6″ thick)*
- *North east wall of in-containment refueling water storage tank (2′6″ thick)*]*

[*The thicknesses and locations of these walls which are part of the boundary of the in-containment refueling water storage tank are shown in Figure 3.8.3-18. They are the portions of the structural wall modules experiencing the largest demand. The structural configuration and typical details are shown in Figures 3.8.3-1, 3.8.3-2, 3.8.3-8, 3.8.3-14, 3.8.3-15, and 3.8.3-17.*]* The structural analyses are described in subsection 3.8.3.4 summarized in Table 3.8.3-2. The design procedures are described in subsection 3.8.3.5.3.

[*The three walls extend from the floor of the in-containment refueling water storage tank at elevation 103′0″ to the operating floor at elevation 135′3″. The south west wall is also a boundary of the refueling cavity and has stainless steel plate on both faces. The other walls have stainless steel on one face and carbon steel on the other.*]* Design summaries are given in Tables 3.8.3-4, 3.8.3-5, and 3.8.3-6. See APP-GW-GLR-045 (Reference 56) for more details~~Appendix 3H for more detailed discussion~~.

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

3.8.4.5.2 Supplemental Requirements for Steel Structures

[*Supplemental requirements for use of AISC-N690 are as follows:*

- *In Section Q1.0.2, the definition of secondary stress applies to stresses developed by temperature loading only.*

- *In Section Q1.3, where the structural effects of differential settlement are present, they are included with the dead load, D.*

- *In Table Q1.5.7.1, the stress limit coefficients for compression are as follows:*

  1.3 instead of 1.5 in load combinations 2, 5, and 6.
  1.4 instead of 1.6 in load combinations 7, 8, and 9.
  1.6 instead of 1.7 in load combination 11.

- *In Section Q1.5.8, for constrained members (rotation and/or displacement constraint such that a thermal load causes significant stresses), supporting safety-related structures, systems, or components, the stresses under load combinations 9, 10, and 11 are limited to those allowed in Table Q1.5.7.1 as modified above.*

- *Sections Q1.24 and Q1.25.10 are supplemented as follows:*

  Shop painting is in accordance with Section M of the Manual of Steel Construction, Load and Resistance Factor Design, First Edition. Exposed areas after installation are field painted in accordance with the applicable portion of Chapter M of the Manual of Steel Construction, Load and Resistance Factor Design, First Edition.]* See subsection 6.1.2.1 for additional description of the protective coatings.

3.8.4.5.3 Design Summary Report

A design summary report is prepared for seismic Category I structures documenting that the structures meet the acceptance criteria specified in subsection 3.8.4.5.

Deviations from the design due to as-procured or as-built conditions are acceptable based on an evaluation consistent with the methods and procedures of Section 3.7 and 3.8 provided the following acceptance criteria are met.

- the structural design meets the acceptance criteria specified in Section 3.8
- the seismic floor response spectra meet the acceptance criteria specified in subsection 3.7.5.4

Depending on the extent of the deviations, the evaluation may range from documentation of an engineering judgment~~judgement~~ to performance of a revised analysis and design. The results of the evaluation will be documented in an as-built summary report.

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

The governing scenario is the case with a delay in the auxiliary building construction for the soft soil site with alternating layers of sand and clay. The delay is postulated to occur just prior to the stage where the auxiliary building walls are constructed. Member forces at the end of construction are calculated considering the effects of settlement during construction. The difference in these member forces from those calculated for dead load in the analyses on soil springs are added as additional dead loads in the critical safe shutdown earthquake load combination.

The member forces for the load combination of dead load plus safe shutdown earthquake, including the member forces locked-in during various stages of plant construction, are within the design capacity for the five critical locations. The evaluation demonstrates that the member forces including locked-in forces calculated by elastic analyses remain within the capacity of the section.

3.8.5.4.3 Design Summary Report

A design summary report is prepared for the basemat documenting that the structures meet the acceptance criteria specified in subsection 3.8.5.5.

Deviations from the design due to as-procured or as-built conditions are acceptable based on an evaluation consistent with the methods and procedures of Sections 3.7 and 3.8 provided the following acceptance criteria are met.

- The structural design meets the acceptance criteria specified in Section 3.8
- The seismic floor response spectra meet the acceptance criteria specified in subsection 3.7.5.4

Depending on the extent of the deviations, the evaluation may range from documentation of an engineering judgment~~judgement~~ to performance of a revised analysis and design. The results of the evaluation will be documented in an as-built summary report.

3.8.5.4.4 Design Summary of Critical Sections

The basemat is designed to meet the acceptance criteria specified in subsection 3.8.4.5. Two critical portions of the basemat are identified below together with a summary of their design. The boundaries are defined by the walls and column lines which are shown in Figure 3.7.2-12 (sheet 1 of 12). Table 3.8.5-3 shows the reinforcement required and the reinforcement provided for the critical sections.

[*Basemat between column lines 9.1 and 11 and column lines K and L*

This portion of the basemat is designed as a two way slab with the shorter directions spanning a distance of 23′6″ between the walls on column lines K and L. The slab is continuous with the adjacent slabs to the east and west. The critical loading is the bearing pressure on the underside of the slab due to dead and seismic loads. This establishes the demand for the top flexural reinforcement at mid span and for the bottom flexural and shear reinforcement at the walls. The basemat is designed for the member forces from the analyses]* described in subsection 3.8.5.4.1. *The top and bottom reinforcement in the east west direction of span are equal. The reinforcement*

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

provided is shown in sheets 1, 2 and 5 of Figure 3.8.5-3. Typical reinforcement details showing use of headed reinforcement for shear reinforcement are shown in Figure 3H.5-3.]*

[*Basemat between column lines 1 and 2 and column lines K-2 and N*

This portion of the basemat is designed as a two way slab with the shorter direction spanning a distance of 22′0″ between the walls on column lines 1 and 2. The slab is continuous with the adjacent slabs to the north and with the exterior wall to the south. The critical loading is the bearing pressure on the underside of the slab due to dead and seismic loads. This establishes the demand for the top flexural reinforcement at mid span and for the bottom flexural and shear reinforcement at wall 2. The basemat is designed for the member forces from the analyses on uniform soil springs]* described in subsection 3.8.5.4.1. [*The reinforcement provided is shown in sheets 1, 2 and 5 of Figure 3.8.5-3. Typical reinforcement details showing use of headed reinforcement for shear reinforcement are shown in Figure 3H.5-3.*]*

Deviations from the design due to as-procured or as-built conditions are acceptable based on an evaluation consistent with the methods and procedures of Sections 3.7 and 3.8 provided the following acceptance criteria are met.

- The structural design meets the acceptance criteria specified in Section 3.8.
- The amplitude of the seismic floor response spectra does not exceed the design basis floor response spectra by more than 10 percent.

Depending on the extent of the deviations, the evaluation may range from documentation of an engineering judgment~~judgement~~ to performance of a revised analysis and design.

3.8.5.5 Structural Criteria

The analysis and design of the foundation for the nuclear island structures are according to ACI-349 with margins of structural safety as specified within it. The limiting conditions for the foundation medium, together with a comparison of actual capacity and estimated structure loads, are described in Section 2.5. The minimum required factors of safety against sliding, overturning, and flotation for the nuclear island structures are given in Table 3.8.5-1.

[*The basemat below the auxiliary building is designed for shear in accordance with the provisions for continuous deep flexural members in paragraph 11.8.3 of ACI 349-01. As permitted by paragraph 11.5.5.1 of ACI 349-01, shear reinforcement is not provided when the factored shear force, V_u, is less than one half of the shear strength provided by the concrete, φV_c.*]*

3.8.5.5.1 Nuclear Island Maximum Bearing Pressures

The foundation will be demonstrated to be capable of withstanding the bearing demand from the nuclear island as described in subsection 2.5.4.5.6.

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

Central and Eastern United States rock sites show higher amplitude at high frequency than the CSDRS. Evaluations for high frequency exceedance at AP1000 plant rock sites have been performed as described in Appendix 3I. It is the conclusion of these evaluations that AP1000 plant systems, structures, and components are qualified for the high frequency seismic response based on the CSDRS with the exception of potential high frequency sensitive components (APP-GW-GLN-144, Reference 3~~5~~). Specific models of components are not identified as part of the AP1000 certified design and are evaluated for high frequency sensitivity as part of the equipment qualification. Appendix 3I provides the criteria for addressing potential high frequency sensitive components for plant locations where there is CSDRS exceedance in the high frequency region.

3.10.1 Seismic and Dynamic Qualification Criteria

3.10.1.1 Qualification Standards

The methods of meeting the general requirements for the seismic and dynamic qualification of seismic Category I mechanical and electrical equipment and instrumentation as described by General Design Criteria (GDC) 1, 2, 4, 14, 23, and 30 are described in Section 3.1. The general methods of implementing the requirements of Appendix B to 10CFR50 are described in Chapter 17.

The Nuclear Regulatory Commission (NRC) recommendations concerning the methods employed for seismic qualification of mechanical and electrical equipment are contained in Regulatory Guide 1.100, which endorses IEEE 344-1987 (Reference 1).

[*AP1000 meets IEEE 344-1987, as modified by Regulatory Guide 1.100, by either type testing or analysis or by an appropriate combination of these methods*]* employing the methodology described in Appendix 3D.

The guidance provided in the ASME Code, Section III, is followed in the design of seismic Category I mechanical equipment to achieve the structural integrity of pressure boundary components. In addition, the AP1000 implements an operability program for active valves following Regulatory Guide 1.148, as addressed in subsection 1.9.1 and in Section 3.9.

3.10.1.2 Performance Requirements for Seismic Qualification

An equipment qualification data package (EQDP) is developed for the instrumentation and electrical equipment classified as seismic Category I. Table 3.11-1 of Section 3.11 identifies the seismic Category I electrical equipment and instrumentation supplied for the AP1000. Each equipment qualification data package contains a section entitled "Performance Requirements." This section establishes the safety-related functional requirements of the equipment to be demonstrated during and after a seismic event. The required response spectra employed by the AP1000 for generic seismic qualification are also identified in the section.

For active seismic Category I mechanical components, the performance requirements are defined in the appropriate design and equipment specifications. Requirements for active valves are discussed in subsection 3.10.2.2. The equipment qualification data packages are referenced in

*NRC Staff approval is required prior to implementing a change in this material; see DCD Introduction Section 3.5.

Direct vessel injection lines
Thermal stratification in the direct vessel injection lines, including leakage through the isolation valves~~vavles~~, is considered in the ASME Code pipe stress and fatigue analysis of these lines.

Main Steam Line

The steam lines are not subjected to thermal stratification by the nature of fluid transported.

Based on the above discussion, thermal stratification does not have an adverse effect on the integrity of AP1000 leak-before-break piping systems.

3B.2.7 Other Mechanisms

The pipe evaluated for leak-before-break does not operate at temperature for which creep fatigue must be considered. Creep fatigue is a concern for ferritic steel piping operation at temperatures above 700°F and for austenitic stainless steel operation above 800°F.

Pipe degradation or failure by indirect causes such as fires, missiles, and component support failures is precluded by criteria for design, fabrication, inspection, and separation of potential hazards in the vicinity of the safety-related piping. The structures, larger pipe, and components in the vicinity of pipe evaluated for leak-before-break are safety-related and seismically designed or are seismically supported if nonsafety-related.

Cleavage type failures are not a concern for systems operating temperature and material used in the stainless steel piping systems. The material used in the main steam line is highly ductile and resistant to cleavage type failure at operating temperatures. The resistance to failure have been demonstrated by material fracture toughness tests.

3B.3 Leak-Before-Break Bounding Analysis

The methodology used for performing the bounding analysis is consistent with that set forth in GDC-4, SRP 3.6.3 (Reference 1) and NUREG-1061, Volume 3 (Reference 2).

Bounding leak-before-break analysis for the applicable AP1000 piping systems is performed. The analysis criteria and development techniques of the bounding analysis curves (BAC) are described below. The bounding analysis curve allows for the evaluation of the piping system in advance of the final piping analysis, incorporating leak-before-break considerations early in the piping design process. The leak-before-break bounding analysis curve is used to evaluate critical points in the piping system. A minimum of two points are required to develop the bounding analysis curve. One point for the low normal stress case and the other point for the high normal stress case. If variations in pipe size, material, pressure or temperature occur for a specific piping system, an additional bounding analysis curve is generated. These points meet the following margins for leak-before-break analysis: (References 1 and 2).

- Margin of 10 on leak detection capability
- Margin of 2 on flaw size
- Establish margin of 1 on load by using absolute combination method of maximum loads

5.4.7.1.2.5 Low Temperature Overpressure Protection

The normal residual heat removal system provides a low temperature overpressure protection function for the reactor coolant system during refueling, startup, and shutdown operations. The system is designed to limit the reactor coolant system pressure to the lower of either the limits specified in 10 CFR 50, Appendix G, or 110 percent of the normal residual heat removal system design pressure.

5.4.7.1.2.6 Spent Fuel Pool Cooling

The normal residual heat removal system has the capability to supplement or take over the cooling of the spent fuel pool when it is not needed for normal shutdown cooling.

5.4.7.2 System Description

Figure 5.4-6 shows a simplified sketch of the normal residual heat removal system. Figure 5.4-7 shows the piping and instrumentation diagram for the normal residual heat removal system. Table 5.4-13 gives the important system design parameters.

The inside containment portions of the system from the reactor coolant system up to and including the containment isolation valves outside containment are designed for full reactor coolant system pressure. The portion of the system outside containment, including the pumps, valves and heat exchangers, has a design pressure and temperature such that full reactor coolant system pressure is below the ultimate rupture strength of the piping.

The normal residual heat removal system consists of two mechanical trains of equipment. Each train includes one residual heat removal pump and one residual heat removal heat exchanger. The two trains of equipment share a common suction line from the reactor coolant system and a common discharge header. The normal residual heat removal system includes the piping, valves and instrumentation necessary for system operation.

The normal residual heat removal system suction header is connected to a reactor coolant system hot leg with a single step-nozzle connection. The step-nozzle connection is employed to minimize the likelihood of air ingestion into the residual heat removal pumps during reactor coolant system mid-loop operations. The suction header then splits into lines with two parallel sets of two normally closed, motor-operated isolation valves in series. This arrangement allows for normal residual heat removal system operation following a single failure of an isolation valve to open and also allows for normal residual heat removal system isolation following a single failure of an isolation valve to close.

The lines join into a common suction line inside containment. A single line from the in~~-side-~~containment refueling water storage tank is connected to the suction header before it leaves containment.

Once outside containment, the suction header contains a single normally closed, motor-operated isolation valve. Downstream of the suction header isolation valve, the header branches into two separate lines, one to each pump. Each branch line has a normally open, manual isolation

Table 5.4-17

PRESSURIZER SAFETY VALVES - DESIGN PARAMETERS

Number	2
Minimum required relieving capacity per valve (lb/hr)	750,000 at 3% accumulation
Set pressure (psig)	2485 ±25 psi
Design temperature (°F)	680
Fluid	Saturated steam
Backpressure	
Normal (psig)	3 to 5
Expected maximum during discharge (psig)	500
Environmental conditions	
Ambient temperature (°F)	50 to 120
Relative humidity (percent)	0 to 100
Residual Heat Removal Relief Valve - Design Parameters	
Number	1
Nominal relieving capacity per valve, ASME flowrate (gpm)	850
Nominal set pressure (psig)	500*
Full-open pressure, with accumulation (psig)	550*
Design temperature (°F)	400
Fluid	Reactor coolant
Backpressure	
Normal (psig)	3 to 5
Expected maximum during discharge (psig)	21
Environmental conditions	
Ambient temperature (°F)	50 to 120
Relative humidity (percent)	0 to 100

Note:
* See text (subsection 5.4.9.3) for discussion of set pressure.

Table 9.2.1-1

NOMINAL SERVICE WATER FLOWS AND HEAT LOADS AT DIFFERENT OPERATING MODES

	CCS Pumps and Heat Exchangers	SWS Pumps and Cooling Tower Cells (Number Normally ins Service)	Flow (gpm)	Heat Transferred (Btu/hr)
Normal Operation (Full Load)	1	1	10,500	103×10^6
Cooldown	2	2	21,000	346×10^6 (173×10^6 per cell)
Refueling (Full Core Offload)	1	1	10,500	74.9×10^6
Plant Startup	2	2	21,000	75.8×10^6
Minimum to Support Shutdown Cooling and Spent Fuel Cooling	1	1	10,000	170×10^6

Table 9A-3 (Sheet 8 of 24)

FIRE PROTECTION SUMMARY

Fire Area/ Zone[1]	Safety Area?[2]	Floor Area Sq Ft	Combust. Material[3]	Fire Sev. Cat.	Amount	Heat Value (Btu)	Comb. Load, Btu/ Sq Ft	Equiv. Dur. (Min)	Boundary Fire Res.[4] (Hours)	Detect. Cap.	Fixed Suppression Capability[5]
1202 AF 03	YES								3	SMOKE	HOSE STATION
1212 AF 12102			BATTERIES	A	120	2.4E+07					
DIVISION C BATTERY			CABLE INS	C	1000	1.0E+07					
ROOM 1		560	NET CAT.	C	TOTAL:	3.4E+07	61000	50			
1222 AF 12202			BATTERIES	A	120	2.4E+07					
DIVISION C BATTERY			CABLE INS	C	1000	1.0E+07					
ROOM 2		560	NET CAT.	C	TOTAL:	3.4E+07	61000	50			
1222 AF 12203			CABLE INS	C	2500	2.6E+07					
DIVISION C DC		395	NET CAT.	C	TOTAL:	2.6E+07	65000	54			
EQUIPMENT ROOM											
1232 AF 12302			CABLE INS	C	3500	3.6E+07					
DIVISION C I&C ROOM		550	NET CAT.	C	TOTAL:	3.6E+07	65000	55			
1232 AF 12312			CABLE INS	C	1500	1.5E+07					
DIVISION C RCP TRIP		395	NET CAT.	C	TOTAL:	1.5E+07	39000	29			
SWITCHGEAR ROOM											
1232 AF 12313			CABLE INS	C	2500	2.6E+07					
I&C/DIVISION C		555	NET CAT.	C	TOTAL:	2.6E+07	46000	35			
PENETRATION ROOM											
FIRE AREA TOTAL:		3015	NET CAT.	C	TOTAL:	1.7E+08	57000	45			
1202 AF 04	YES								3	SMOKE	HOSE STATION
1212 AF 12101			BATTERIES	A	120	2.4E+07					
DIVISION A BATTERY			CABLE INS	C	1000	1.0E+07					
ROOM		525	NET CAT.	C	TOTAL:	3.4E+07	65000	55			
1222 AF 12201			CABLE INS	C	3500	3.6E+07					
DIVISION A DC		525	NET CAT.	C	TOTAL:	3.6E+07	68000	58			
EQUIPMENT ROOM											
1232 AF 12301			CABLE INS	C	3500	3.6E+07					
DIVISION A I&C ROOM		550	NET CAT.	C	TOTAL:	3.6E+07	65000	55			
FIRE AREA TOTAL:		1600	NET CAT.	C	TOTAL:	1.1E+08	66000	56			
1202 AF 05	NO								3	SMOKE	NONE
STAIRWELL S05			NEGLIGIBLE								
1204 AF 01	NO								3	SMOKE	HOSE STATION
			CABLE INS	C	500	5.0E+06					
RNS PUMP ROOM B			LUBE OIL	E	5	7.6E+05					
		205	NET CAT.	D	TOTAL:	5.8E+06	28000	23			
FIRE AREA TOTAL:		205	NET CAT.	D	TOTAL:	5.8E+06	28000	23			

Table 9A-3 (Sheet 24 of 24)

FIRE PROTECTION SUMMARY

Fire Area/ Zone[1]	Safety Area?[2]	Floor Area Sq Ft	Combust. Material[3]	Fire Sev. Cat.	Amount	Heat Value (Btu)	Comb. Load, Btu/ Sq Ft	Equiv. Dur. (Min)	Boundary Fire Res.[4] (Hours)	Detect. Cap.	Fixed Suppression Capability[5]
6030 AF 60324			CABLE INS	C	100	1.0E+06				NONE	HOSE STATION
TOOL STORAGE AREA			PAPER	C	50	3.9E+05					
			PLASTIC	D	50	6.6E+05					
			WOOD	C	100	8.4E+05					
		144	NET CAT.	C	TOTAL:	2.9E+06	20070	15			
6030 AF 60330			CABLE INS	C	2000	2.0E+06				NONE	HOSE STATION
SECURITY ROOM			TRASH	B	100	1.9E+05					
			PAPER	C	1000	7.7E+05					
			PLASTIC	D	1000	3.3E+05					
		144	NET CAT.	C	TOTAL:	3.6E+06	25200	19			
FIRE AREA TOTAL:		2328	NET CAT.	E	TOTAL:	1.4E+08	60140	45			
6030 AF 03	NO								3	NONE	DRY PIPE SPRINKLERS
DIESEL FUEL DAY			FUEL	E	1500	2.2E+08					HOSE STATION
TANK VAULT A			OIL								
FIRE AREA TOTAL:		100	NET CAT.	E	TOTAL:	2.2E+08	2160000	1620			
6030 AF 04	NO								3	NONE	DRY PIPE SPRINKLERS
DIESEL FUEL DAY			FUEL	E	1500	2.2E+08					HOSE STATION
TANK VAULT B			OIL								
FIRE AREA TOTAL:		100	NET CAT.	E	TOTAL:	2.2E+08	2160000	1620			
6001 AF01	NO								2	NONE	NONE
STAIRWELL			NEGLIGIBLE								

Notes:

1. The first four digits of the fire area and fire zone numbers indicate the building, level and building area in which the fire area/zone is located. When the third or fourth digit is a zero, the fire area/zone spans more than one level or building area. The last two digits in a fire area number are a sequence number only. The last five digits in a fire zone number coincide with the room number of a prominent room in the fire zone.
2. A YES indication in the Safety Area column means that one or more safety-related components are located in the fire area.
3. Estimated quantities of combustible materials are shown. Where the presence of transient combustibles is anticipated, their presence is indicated by the listing of volatiles or trash. The units and heat of combustion values for the combustible materials are shown in Table 9A-1.
4. The boundary fire resistance for each fire area represents the minimum resistance, in hours, for the surrounding walls, floor, and ceiling, except that:
   - A non-rated barrier capable of qualifying as a three-hour barrier is considered to have a resistance of three hours, provided that penetrations are adequately sealed.
   - Stairwells, elevator shafts and the like, which are enclosed by two-hour (minimum) fire barrier walls, may comprise a portion of the boundary of a fire area having a three hour resistance.
   - Building exterior walls below grade (soil on the outside) are considered to have a fire resistance of at least three hours even though they are not fire-rated.

9A.3.6.2 Fire Area 6030 AF 02

This fire area contains the diesel generator and supporting equipment for one train of the onsite standby ac power system, tool storage area and security room. The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 6030 AF 60320	60320	Diesel generator room B
• 6030 AF 60321	60321	Service module B
• 6030 AF 60323	60323	Combustion air cleaner area B
• 6030 AF 60324	60324	Tool storage area~~room~~
• 6030 AF 60330	60330	Security room

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors in the service module
- Dry pipe sprinklers in the diesel generator room
- Hose station(s)
- Portable fire extinguishers (including carbon dioxide)

Smoke Control Features

The diesel generator building ventilation system (VZS) serves this fire area by means of the engine room air handling unit, the service module air handling unit, and the standby exhaust fans. The engine room air handling unit stops upon actuation of the fire suppression system in the fire area or if smoke is detected in the supply air duct from the air handling unit. The service module air handling unit stops upon actuation of the fire suppression system in the fire area or if smoke is detected in the supply air duct from the air handling unit. The standby exhaust fans stop upon actuation of the fire suppression system in the fire area. These actions control the spread of fire and smoke. After the fire, smoke is removed from the fire area by manually turning on the ventilation exhaust fans mounted on the roof over the fire area, or by opening the roll-up door and personnel doors and utilizing portable exhaust fans.

Fire Protection Adequacy Evaluation

A fire in the diesel generator room is detected through the operation of the dry pipe sprinkler system which produces an audible alarm locally, and both visual and audible alarms in the main control room and security central alarm station. A fire in the service module is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and the security central alarm station. A fire in the diesel generator room is extinguished by the automatic fire suppression system or manually, using hose streams or portable extinguishers. A fire in the service module is extinguished manually using hose streams or portable extinguishers.

General Test Method and Acceptance Criteria

Demineralized water transfer and storage system performance is observed and recorded during a series of individual component and integrated system testing. The following defense-in-depth testing verifies that the system functions as described in subsection 9.2.4 and appropriate design specifications:

a) Proper operation of the system pumps, valves, and blower~~, and~~ is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

14.2.9.4.10 Compressed and Instrument Air System Testing

Purpose

The purpose of the compressed and instrument air system testing is to verify that the as-installed components properly perform the functions of providing compressed air at the required pressures to various plant users, as described in the Compressed and Instrument Air portion of Section 9.3.

Prerequisites

The construction testing of the compressed and instrument air system has been completed. The component cooling water system is operational and providing cooling for the compressor units. Required support systems, electrical power supplies and control circuits are operational.

General Test Method and Acceptance Criteria

Compressed and instrument air system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system and its plant users, where applicable, function as described in subsection 9.3.1.4 and appropriate design specifications:

a) Proper operation of the system compressors, receivers, prefilters, air dryers, afterfilters, purifiers, and valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

c) Integral testing is performed to verify that the instrument air subsystem can provide sufficient air pressure to accommodate the maximum number of air-operated valves expected to operate simultaneously.

d) Testing is performed to verify the fail-safe positioning of safety-related air-operated valves for sudden loss of instrument air or gradual loss of pressure as described in subsection 9.3.1.4.

BASES

LCO (continued)

Peak kw/ft limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA. The highest calculated linear power densities in the core at specific core elevations are displayed for operator visual verification relative to the COLR values.

The determination of $F^N_{\Delta H}$ identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for DNB. Should $F^N_{\Delta H}$ exceed the limit given in the COLR, the possibility exists for DNBR to exceed the value used as a base condition for the safety analysis.

Two levels of alarms on power distribution parameters are provided to the operator. One serves as a warning before the three parameters (kw/ft(Z), $F^N_{\Delta H}$, DNBR) exceed their values used as a base condition for the safety analysis. The other alarm indicates when the parameters have reached their limits.

APPLICABILITY

The OPDMS-monitored power distribution parameter limits must be maintained in MODE 1 above 50% RTP~~RTD~~ to preclude core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES, and ~~MODE 1~~ below 50% RTP, is not required because there is either insufficient stored energy in the fuel or insufficient energy transferred to the reactor coolant to require a limit on the distribution of core power. The OPDMS monitoring of SDM is applicable~~must be OPERABLE~~ in MODES 1 and 2 with $K_{eff} \geq 1.0$.

Specifically for $F^N_{\Delta H}$, the design bases accidents (DBAs) that are sensitive to $F^N_{\Delta H}$ in other MODES (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

In addition to the alarms discussed in the LCO section above (alarms on OPDMS-monitored power distribution parameters), there is an alarm indicating the potential inoperability of the OPDMS itself.

Should the OPDMS be determined to be inoperable for other than reasons of alarms inoperable, this LCO is no longer applicable and LCOs 3.2.1 through 3.2.4 become applicable.

BASES

ACTIONS (continued)

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event that air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is reasonable based on engineering judgement and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified to be locked closed by administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

BASES

ACTIONS

A.1, A.2, and A.3

Condition A represents one division with one or two battery chargers inoperable (e.g., the voltage limit of SR 3.8.1.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 6 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within 24 hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability.

Because of the passive system design and the use of fail-safe components, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate most DBAs following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one DC subsystem against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 24 hours, avoiding a premature shutdown with its own attendant risk.

If established battery terminal float voltage cannot be restored to greater than or equal to the minimum established float voltage within 6 hours, and the charger is not operating in the current-limiting mode, a faulty charger is indicated. A faulty charger that is incapable of maintaining established battery terminal float voltage does not provide assurance that it can revert to and operate properly in the current limit mode that is necessary during the recovery period following a battery discharge event that the DC system is designed for.

BASES

ACTIONS (continued)

Because of the passive system design and the use of fail-safe components, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate most DBAs following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one DC subsystem against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

The installed spare battery bank and charger may be used to restore an inoperable Class 1E DC electrical power subsystem; however, all applicable Surveillances must be met by the spare equipment used, prior to declaring the subsystem OPERABLE.

D.1

Condition D represents two divisions with one or more batteries inoperable. With one or more batteries inoperable, the DC bus is being supplied by the OPERABLE battery charger. Any event that results in a loss of the AC bus supporting the battery charger will also result in loss of DC to that train. The 2 hour limit allows sufficient time to effect restoration of an inoperable battery given that the majority of the conditions that lead to battery inoperability (e.g., loss of battery charger, battery cell voltage less than 2.07 V, etc.) are identified in Specifications 3.8.1, 3.8.2, and 3.8.7 together with additional specific completion times.

The installed spare battery bank and charger may be used to restore an inoperable Class 1E DC electrical power subsystem; however, all applicable Surveillances must be met by the spare equipment used, prior to declaring the subsystem OPERABLE.

E.1

If one of the Class 1E DC electrical power subsystems is inoperable, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate all design basis accidents, based on conservative analysis.

Because of the passive system design and the use of fail-safe components, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate most DBAs

BASES

ACTIONS (continued)

following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one DC subsystem against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

The 6 hour Completion Time is also consistent with the time specified for restoration of one (of four) Protection and Safety Monitoring System (PMS) (LCO 3.3.2, ESFAS Instrumentation). Depending on the nature of the DC electrical power subsystem inoperability, one supported division of instrumentation could be considered inoperable. Inoperability of a PMS Division is similar to loss of one DC electrical power subsystem. In both cases, actuation of the safety functions associated with one of the four subsystems/divisions may no longer be available.

F.1

Condition F represents two subsystems with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected distribution subsystems. The 2 hour limit is consistent with the allowed time for two inoperable DC distribution subsystems.

If two of the required DC electrical power subsystems are inoperable (e.g., inoperable battery, inoperable battery charger(s), or inoperable battery charger and associated inoperable battery), the two remaining DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate all but the very worst case events. Since a subsequent worst case single failure would, however, result in the loss of the third subsystem, leaving only one subsystem with limited capacity to mitigate events, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 11) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

BASES

APPLICABILITY

The Class 1E AC and DC electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The Class 1E AC and DC electrical power distribution subsystem requirements for MODES 5 and 6 are covered in the Bases for Specification 3.8.6, "Class 1E Distribution Systems – Shutdown."

ACTIONS

A.1

With one division of the Class 1E AC instrument and control bus inoperable the remaining Class 1E AC instrument and control buses have the capacity to support a safe shutdown and to mitigate all DBAs, based on conservative analysis.

Because of the passive system design and the use of fail-safe components, the remaining Class 1E AC instrument and control buses have the capacity to support a safe shutdown and to mitigate most design basis accidents following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one AC instrument and control bus against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

This 6 hour limit is shorter than Completion Times allowed for most supported systems which would be without power. Taking exception to LCO 3.0.2 for components without adequate DC Power, which would have Required Action Completion Times shorter than 6 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

ACTIONS (continued)

B.1

With one Division of the Class 1E DC electrical power distribution subsystem inoperable, the remaining Divisions have the capacity to support a safe shutdown and to mitigate all DBAs, based on conservative analysis.

Because of the passive system design and the use of fail-safe components, the remaining Divisions have the capacity to support a safe shutdown and to mitigate most design basis accidents following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one Division against the risks of a forced shutdown. Additionally, the completion time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

The 6 hour Completion Time is also consistent with the time specified for restoration of one (of four) Protection and Safety Monitoring System division (LCO 3.3.2, ESFAS Instrumentation). Depending on the nature of the DC electrical power distribution subsystem inoperability, one supported division of instrumentation could be considered inoperable. Inoperability of a PMS division is similar to loss of one DC electrical power distribution subsystem. In both cases, actuation of the safety functions associated with one of the four subsystems/divisions may no longer be available.

This 6 hour limit is shorter than Completion Times allowed for most supported systems which would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 6 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions to restore power to the affected division; and

BASES

SURVEILLANCE REQUIREMENTS (continued)

walkdown. The 31 day Frequency is based on engineering judgement and is considered reasonable in view of other administrative controls that will verify that the valve opening is an unlikely possibility.

REFERENCES

1. Chapter 15, "Accident Analyses."

2. NUREG-0800, Standard Review Plan, Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the RCS."

BASES

ACTIONS

LCO 3.0.8 is applicable while in MODE 5 or 6. Since irradiated fuel assembly movement can occur in MODE 5 or 6, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, the fuel movement is independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement to safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4 1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgement and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.183, "Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

2. Section 15.7.4, "Fuel Handling Accident."

~~DCD~~ Table 17.4-1 (Sheet 1 of 8)

RISK-SIGNIFICANT SSCs ~~WITHIN THE~~ SCOPE OF D-RAP

System, Structure, or Component (SSC)(1)	Rationale(2)	Insights and Assumptions
System: Component Cooling Water (CCS)		
Component Cooling Water Pumps (CCS-MP-01A/B)	EP	These pumps provide cooling of the normal residual heat removal system (RNS) and the spent fuel pool heat exchanger. Cooling the RNS heat exchanger is important to investment protection during shutdown reduced-inventory conditions. CCS valve realignment is not required for reduced-inventory conditions.
System: Containment System (CNS)		
Containment Vessel (CNS-MV-01)	EP, L2	The containment vessel provides a barrier to steam and radioactivity released to the atmosphere following accidents.
Hydrogen Igniters (VLS-EH-1 through -64)	RAW/CCF~~EP~~, L2, Regulations	The hydrogen igniters provide a means to control H_2 concentration in the containment atmosphere, consistent with the hydrogen control requirements of 10 CFR 50.34f.
System: Chemical and Volume Control System (CVS)		
Makeup Pumps (CVS-MP-01A/B)	EP	These pumps provide makeup to the RCS to accommodate leaks and to provide negative reactivity for shutdowns, steam line breaks, and ATWS.
Makeup Pump Suction and Discharge Check Valves (CVS-PL-V113, -V160A/B)	EP	These CVS check valves are normally closed and have to open to allow makeup pump operation.
Letdown Isolation Valves (CVS-PL-V045, -V047)	RAW	The CVS letdown isolation valves automatically close to prevent excessive reactor coolant letdown and provide containment isolation. These containment isolation valves are important in limiting offsite releases following core melt accidents.
System: Diverse Actuation System (DAS)		
DAS Processor Cabinets and Control Panel (used to provide automatic and manual actuation) (DAS-JD-001, -002, -003, -004, OCS-JC-020)	RAW	The DAS is diverse from the PMS and provides automatic and manual actuation of selected plant features including control rod insertion, turbine trip, passive residual heat removal (PRHR) heat exchanger actuation, core makeup tank actuation, isolation of critical containment lines, and passive containment cooling system (PCS) actuation.
Annex Building UPS Distribution Panels (EDS1-EA-1, EDS1-EA-14, EDS2-EA-1, EDS2-EA-14)	RAW	These panels distribute power to the DAS equipment.

43. American National Standards Institute, 1988, "ANSI/HF 100-1988, American National Standard for Human Factors Engineering of Visual Display Terminal Workstations," (Santa Monica, CA, Human Factors Society, American National Standards Institute).

44. WCAP-14694, "Designer's Input to Determination of the AP600 Main Control Room Staffing Level," Revision 0, July 1996.

45. APP-GW-GL-022, "AP1000 Probabilistic~~y~~ Risk Assessment," Revision 8.

[*46. WCAP-14396, "Man-in-the-Loop Test Plan Description," Revision 3, November 2002.*]*

47. APP-GW-GLR-082, "Execution and Documentation of the Human System Interface Design Implementation Plan," Westinghouse Electric Company LLC.

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

19.1.6 Plant Definition

19.1.6.1 General Description

See Chapter 1.

19.1.6.2 AP1000 Design Improvement as a Result of Probabilistic Risk Assessment Studies

Design improvements were incorporated in the AP600 design based on the results of the AP600 PRA and other design analyses and are discussed in Reference 19.1-3. These improvements have been retained in the AP1000 design. Additional design changes have been incorporated in the AP1000 as a result of the AP1000 PRA. The most significant design changes prompted by the AP1000 PRA are:

- Two recirculation lines, each containing a motor-operated valve and a squib valve or a check valve and a squib valve in series, are used to provide recirculation flow from containment sump to the core through direct vessel injection line. Diversity is provided in the actuation by using diverse squib valves. The motor-operated valve is designed so that it remains open in case of failure.
- Three parallel supply lines allow water flow from PCCWST to the containment shell. Diversity is provided in the actuation by using motor-operated valves for one path.

19.1.7 References

19.1-1 Advanced Light Water Reactor Requirements Document, Volume III, Appendix A to Chapter 1, "PRA Key Assumptions and Groundrules," Revisions 5 and 6, December 1993.

19.1-2 EPRI MAAP 4.0 Users Manual.

19.1-3 GW-GL-022, "AP600 Probabilistic Risk Assessment," August 1998~~AP600 PRA~~.

important during drained conditions. This indicates that maintaining the reliability of the recirculation line squib valves is important.

- Human errors are not overly important to shutdown core damage frequency. There is no particular dominant contributor. Sensitivity results show that the shutdown core damage frequency would remain very low even with little credit for operator actions.

  One action, operator failure to recognize the need for reactor coolant system depressurization during safe/cold shutdown conditions, is identified as having a significant risk increase value. This indicates it is important that the procedures include this action and the operators understand and are appropriately trained for it.

- Individual component failures are not significant contributors to shutdown core damage frequency, and there is no particular dominant contributor. This confirms the at-power conclusion that single independent component failures do not have a large impact on core damage frequency for AP1000 and reflects the redundancy and diversity of protection at shutdown as well.

- The in-containment refueling water storage tank provides a significant benefit during shutdown because it serves as a passive backup to the normal residual heat removal system.

19.59.5.2 Large Release Frequency for Shutdown and Low-Power Events

The baseline PRA shutdown large release frequency for AP600 was calculated to be 1.5E-08 per reactor-year, associated with a shutdown CDF of 9.0E-08 per year. The AP1000 LRF is estimated to be 2.05E-08 per year, with the same risk profile as that of AP600 (see Table 19.59-15). This LRF compares well with the at-power LRF of 1.95E-08 per year.

19.59.5.3 Shutdown Results Summary

The results of the low-power and shutdown assessment show that the AP1000 design includes redundancy and diversity at shutdown not found in current plants. In particular, the in-containment refueling water storage tank provides a unique safety backup to the normal residual heat removal system. Maintenance at shutdown has less impact on the defense-in-depth features for AP1000 than for current plants. In accordance with plant technical specifications, safety-related system planned maintenance is performed only during those shutdown modes when the protection provided by the safety-related system is not required. Further, maintenance of nonsafety systems, such as the normal residual heat removal system, component cooling water system, and service water system, is performed at power to avoid adversely affecting shutdown risk. These contribute to the extremely low shutdown core damage and the small release frequency.

19.59.6 Results from Internal Flooding, Internal Fire, and Seismic Margin Analyses

19.59.6.1 Results of Internal Flooding Assessment

A scoping internal flooding analysis was performed based on AP1000 design information, with conservative assumptions or engineering judgment~~judgement~~ used for simplifying the analysis.

Because the approach taken in performing the internal fire analysis makes various conservative assumptions and is bounding, the results of uncertainty, sensitivity, or importance analyses would be biased. Therefore, these analyses were not performed based on the judgment~~judgement~~ that they would be of little value in providing additional insights to determine whether fire vulnerabilities exist for beyond-design-basis fires.

The major reasons for the AP1000's relatively low overall fire-induced core damage frequency, even on a bounding basis, include the following:

- The fire protection design provides, to the extent possible, separation of the alternate safety-related shutdown components and cabling using 3-hour-rated fire barriers. For example, areas containing safety-related cabling or components are physically separated from one another and from the areas that do not contain any safety-related equipment by 3-hour-rated fire barriers. This defense-in-depth feature diminishes the probability of a fire to impact more than one safety-related shutdown system.

- Since the passive safety-related systems do not require cooling water or ac power, they are less susceptible to being unavailable due to a fire than currently operating plants' active safe shutdown equipment. As a result, the impact of fires on the shutdown capability is significantly reduced compared to current plants.

The results of this analysis show that the AP1000 design is sufficiently robust that internal fires during either power operation or shutdown do not represent a significant contribution to core damage frequency.

19.59.6.3 Results of Seismic Margin Analysis

The seismic margin analysis (SMA) shows the systems, structures, and components required for safe shutdown. The high confidence, low probability of failure (HCLPF) values are greater than or equal to 0.50g. This HCLPF is determined by the seismically induced failure of the fuel in the reactor vessel, core assembly failures, IRWST failure, or containment interior failures. The SMA result assumes no credit for operator actions at the 0.50g review level earthquake, and assumes a loss of offsite power for all sequences.

The seismic margin analysis shows the plant to be robust against seismic event sequences that contain station blackout coupled with other seismic or random failures. The analysis also shows the plant's capability to respond to seismic events without benefit of the operators' actions.

19.59.7 Plant Dose Risk From Release of Fission-Products

The design certification of the AP1000 included consideration by the NRC of the topic referred to in this section.

19.59.8 Overall Plant Risk Results

The total plant risk expressed in terms of plant core damage frequency and severe release frequency for all events studied in this PRA are summarized in Table 19.59-17.

Change Number 11

The reactor vessel cavity and the two steam generator compartments are interconnected by a large vertical access tunnel. These four compartments are treated, in this discussion, as one large floodable volume and they are referred to as the reactor coolant system compartment. Flooding of this compartment above elevation 107′-2″ also includes the maintenance floor outside the curbs around the other three compartments.

The PXS-A compartment (Room 11206), PXS-B compartment (Room 11207), and the chemical and volume control system compartment (Room 11209~~8~~) are physically separated and isolated from each other by structural walls and curbs such that flooding in any one of these compartments or in the reactor coolant system compartment cannot cause flooding in any of the other compartments. The access hatch to the PXS-B compartment is located near the containment wall and is normally closed to address severe accident considerations. The access hatch to the PXS-B compartment is accessible from Room 11300 on elevation 107′-2″.

The fire protection system and the demineralized water transfer and storage system are open-cycle systems that enter the containment. During plant operation, the containment piping for these systems is isolated by containment isolation valves and is not a potential flooding source. These systems are not open systems as defined in Bulletin 80-24 (one that has an essentially unlimited source).

Reactor Coolant System Compartment

The reactor coolant system compartment, represented by the reactor vessel cavity, the two steam generator compartments, and the large vertical access tunnel, is the largest of the separate floodable compartments. With the exception of the pressurizer which is at a higher elevation, the principal components of the reactor coolant system are contained in this compartment.

The reactor vessel cavity and the adjoining equipment room are at the lowest level in the containment. The floor level of these rooms is at elevation 71′-6″ The floor level of the two steam generator compartments is at elevation 83′-0″. A portion of each compartment has low point areas at elevation 80′-0″.

The containment sump pumps are located in the equipment room at elevation 71′-6″. The arrangement for the floor drains from the two passive core cooling system compartments and the chemical and volume control system compartment provide a drain path for each compartment to the lowest level of containment (elevation 71′-6″) where the containment sump is located. Therefore, the source of the flooding in the reactor coolant system compartment is not limited to the components or systems contained within this compartment.

Any leakage that occurs within the containment drains by gravity to the elevation 71′-6″ equipment room. Reverse flow into the two passive core cooling system compartments and the chemical and volume control system compartment is prevented by redundant backflow preventers in each of the three compartment drain lines.

Flooding in any compartment of the containment is detected by the containment sump level monitoring system and the containment flood-up level instrumentation.

Change Number 12

Lifting Devices Not Specially Designed

Slings or other lifting devices not specially designed are selected in accordance with ANSI B30.9 (Reference 15), except that the load rating is based on the combined maximum static and dynamic loads that could be imparted to the sling.

For the handling of critical loads, dual or redundant slings are used, or a sling having a load rating twice that required for a non-critical load is used and should~~shall~~ be constructed of metallic material (chain or wire rope) per NRC Regulatory Issue Summary 2005-25, Supplement 1 (Reference 23).

Load Lift Points

The design stress safety factors for heavy load lift points, such as lifting lugs or cask trunnions, are consistent with the safety factors used for special lifting devices. The design of lift points for critical loads is in accordance with NUREG-0612, Paragraph 5.1.6.(3).

9.1.5.2.1.3 Instrumentation Applications

Limit switches are used to initiate protective responses to:

- Hoist overtravel
- Hoist overspeed
- Hoist overload or unbalanced load
- Improper winding of hoist rope on the drum
- Bridge or trolley overtravel

Redundant limit switches are used with the main hoist and the auxiliary hoists to limit the extent of travel in both the hoisting and lowering directions. The primary protection for each hoist in each direction is a limit switch which interrupts power to the hoist motor via the control circuitry. Interruption of power to the hoist motor causes the hoist brakes to set. The hoist may be operated in the safe direction to back out of the overtravel condition.

The secondary protection for each hoist in the raising direction is a block-actuated limit switch which directly interrupts power to the hoist motor and causes the brake(s) to set. The secondary protection for each hoist in the lowering direction is a limit switch which is mechanically and electrically independent of the primary switch but also interrupts power to the hoist motor via the control circuitry. Actuation of the secondary limit switches prevents further hoisting or lowering until specific corrective action is taken.

A centrifugal-type limit switch, located on the drum shaft, provides overspeed protection for each hoist. Hoist speeds in excess of 115 percent of the rated lowering speed for a critical load cause the hoist motor to stop and the holding brakes to set.

A load-sensing system is used to detect overloading of the hoists. Hoisting motion is stopped when the overload setpoint is exceeded. Similarly, an unbalanced load is detected by a system that stops the hoist motion when there is excessive movement of the equalizer mechanism.

9.1.5.2.2.2 Component Descriptions

The cask handling crane is designed according to NUREG-0554 supplemented by ASME NOG-1. Table 9.1.5-1 lists the design characteristics of this crane. This subsection describes how the code requirements are implemented in the design of key safety-related components. Associated lifting devices and load lift points are also described.

Hoist System

The hoisting rope is wound around the drum in a single layer. If the rope becomes dislodged from its proper groove, the crane drives are automatically shut down and the brakes are set. Features are also provided to contain the drum and prevent disengagement of the gearing in the event of drum shaft or bearing failure. A control brake and two redundant holding brakes are provided.

Two separate, redundant reeving systems are used, so that a single rope failure will not result in the dropping of the load. Two wire ropes are reeved side-by-side through the sheave. Each cable passes through an equalizer that adjusts for unequal cable length. The equalizer is also a load transfer safety system, eliminating sudden load displacement and shock to the crane in the unlikely event of a cable break. Overtravel protection is provided (see subsection 9.1.5.2.2.3); however, even in the event of hook overtravel in the raising direction to the point the load block contacts the crane structure, the ropes cannot be cut or crushed.

The load block provides two separate load attachment points; the main hook is a two-pronged sister hook with safety latches.

Auxiliary Hoist System

The auxiliary hoist system is similar to that of the main hoist.

Special Lifting Devices

Special lifting devices for critical and non-critical loads are designed to meet the applicable requirements of ANSI N14.6 (Reference 14). The stress design safety factors are based on the combined maximum static and dynamic loads that could be imparted to the handling device, based on the characteristics of the crane. Special lifting devices used for the handling of critical loads are listed in Table 9.1.5-2.

Lifting Devices Not Specially Designed

Slings or other lifting devices not specially designed are selected in accordance with ANSI B30.9 (Reference 15), except that the load rating is based on the combined maximum static and dynamic loads that could be imparted to the sling.

For the handling of critical loads, dual or redundant slings are used, or a sling having a load rating twice that required for a non-critical load is used and should~~shall~~ be constructed of metallic material (chain or wire rope) per NRC Regulatory Issue Summary 2005-25, Supplement 1 (Reference 23).

Change Number 13

Table 8.3.1-2 (Sheet 4 of 4)

ONSITE STANDBY DIESEL GENERATOR ZOS MG 02B NOMINAL LOADS

Manual Loads (Note 2)

Item No.	Time Seq. (sec)	Event or Load Description	Rating (hp/kW)	Operating Load (kW)
53.	--	Class 1E Div. B Battery Charger 1	78 kVA	26
54.	--	Class 1E Div. B Battery Charger 2	78 kVA	15
55.	--	Class 1E Div. D Battery Charger 1	78 kVA	26
56.	--	Supplemental Air Filtration System Fan B	15 hp	15
57.	--	Supplemental Air Filtration System Electric Heater B	20 kW	20
58.	--	Backup Group 4B Pressurizer Heaters	246 kW	246
59.	--	CRDM Fan 01C	75 hp	62
60.	--	CRDM Fan 01D	75 hp	62
61.	--	Spent Fuel Cooling Pump B	150~~250~~ hp	125~~200~~
62.	--	Make-Up Pump B	600 hp	498
63.	--	Non-1E Regulating XFMR EDS2-DT-1	75 kVA	25
64.	--	Annex Bldg Equipment Room Return/Exhaust Fan B	20 hp	17
65.	--	Annex Bldg Equipment Room AHU MS02B Fan	50 hp	42
66.	--	Annex Bldg Swgr Rm AHU MS 05B Fan	50 hp	42
67.	--	Annex Bldg Swgr Rm Ret/Exhaust Fan 06B	25 hp	21
		Total Manually Sequenced Loads (kW)		1242~~1317~~

Change Number 15

Table 3.3.1-1 (page 4 of 5)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
20. ADS Stages 1, 2, and 3 Actuation input from engineered safety feature actuation system						
a. Manual	1,2	2 switch sets	B	SR 3.3.1.11	NA	NA
	$3^{(j)},4^{(j)},5^{(j)}$	2 switch sets	B	SR 3.3.1.11	NA	NA
b. Automatic	1,2	4	L	SR 3.3.1.7	NA	NA
	$3^{(j)},4^{(j)},5^{(j)}$	4	P	SR 3.3.1.7	NA	NA
21. Core Makeup Tank Actuation input from engineered safety feature actuation system						
a. Manual	1,2	2 switch sets	B	SR 3.3.1.11	NA	NA
	$3^{(j)},4^{(j)},5^{(j)}$	2 switch sets	B	SR 3.3.1.11	NA	NA
b. Automatic	1,2	4	L	SR 3.3.1.7	NA	NA
	$3^{(j)},4^{(j)},5^{(j)}$	4	P	SR 3.3.1.7	NA	NA

(j) With Reactor Trip Breakers closed and Plant Control System capable of rod withdrawal.

Table 3.3.2-1 (page 9 of 13)
Engineered Safeguards Actuation System Instrumentation

Spaces Added

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
15.	Boron Dilution Block						
	a. Source Range Neutron Flux Doubling	$2^{(f)},3^{(f)},4^{(m)}$	4	B,T	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ ~~Source Range Flux X 2.201 in 50 minutes~~	Source Range Flux X ~~2.2 in 50 minutes~~
		$5^{(m)}$	4	B,P	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ ~~Source Range Flux X 2.201 in 50 minutes~~	Source Range Flux X ~~2.2 in 50 minutes~~
	b. Reactor Trip	Refer to Function 18.b~~a~~ (ESFAS Interlocks, Reactor Trip, P-4) for all requirements.					
16.	Chemical Volume and Control System Makeup Isolation						
	a. SG Narrow Range Water Level – High 2	$1,2,3^{(m)},4^{(j,m)}$	4 per SG	B,R	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ ~~82.05% span~~	~~82% span~~
	b. Pressurizer Water Level – High 1	$1,2,3^{(m)}$	4	B,Q	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ ~~23.05% span~~	23% span
	Coincident with Safeguards Actuation	$1,2,3^{(m)}$	Refer to Function 1 (Safeguards Actuation) for initiating functions and requirements.				
	c. Pressurizer Water Level – High 2	$1,2,3,4^{(j,m,p)}$	4	B,T	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ ~~59.05% span~~	59% span
	d. Containment Radioactivity – High 2	$1,2,3^{(m)}$	4	B,Q	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ ~~150 R/hr~~	100 R/hr
	e. Manual Initiation	$1,2,3^{(m)},4^{(j,m)}$	2 switches	E,R	SR 3.3.2.3	NA	NA
	f. Source Range Neutron Flux Doubling	Refer to Function 15.a (Boron Dilution Block, Source Range Neutron Flux Doubling) for all requirements.					
	g. SG Narrow Range Water Level High	$1,2,3^{(m)},4^{(j,m)}$	4 per SG	B,R	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 72.85% span	72.8% span
	Coincident with Reactor Trip (P-4)	Refer to Function 18.a (ESFAS Interlocks, Reactor Trip, P-4) for all requirements.					

(f) Not applicable when critical or during intentional approach to criticality.

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(m) Not applicable for valve isolation Functions whose associated flow path is isolated.

(p) Above the P-19 (RCS Pressure) interlock.

5.5 Programs and Manuals

5.5.6 Technical Specifications (TS) Bases Control Program (continued)

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of (b) above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.7 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate action taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the supported system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirement of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support systems inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

BASES

APPLICABILITY (continued)

In MODES 1 and 2, the detection and mitigation of a boron dilution event does not assume the detection of the event by the source range instrumentation. In these MODES, the event would be signal~~l~~ed by an intermediate range trip, a trip on the Power Range Neutron Flux - High (low setpoint nominally at 25% RTP), or Overtemperature delta T. The two demineralized water isolation valves close automatically upon reactor trip.

In MODE 6, a dilution event is precluded by the requirement in LCO 3.9.2 to close, lock and secure at least one valve in each unborated water source flow path.

ACTIONS

A.1

If only one of the demineralized water isolation valve and/or the makeup line isolation valve is/are OPERABLE, the redundant valve must be restored to OPERABLE status in 72 hours. The allowed Completion Time assures expeditious action will be taken, and is acceptable because the safety function of automatically isolating the clean water source can be accomplished by the redundant isolation valve(s).

B.1

If the Required Actions and associated Completion Time of Condition A are not met, or if both CVS demineralized water isolation valves or both makeup line isolation valves are not OPERABLE (i.e., not able to be closed automatically), then the demineralized water supply flow path to the RCS must be isolated. Isolation can be accomplished by manually isolating the CVS demineralized water isolation valve(s) or by positioning the 3-way blend valve to only take suction from the boric acid tank. Alternatively, the dilution path may be isolated by closing appropriate isolation valve(s) in the flow path(s) from the demineralized water storage tank to the reactor coolant system.

The Action is modified by a Note allowing the flow path to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the main control room. In this way, the flow path can be rapidly isolated when a need for isolation is indicated.

BASES

ACTIONS (continued)

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency of pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed differential pressure flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

either to ≥ 100°F, or decreases to ≤ 50°F. Since the maximum tank temperature variation during the normal surveillance Frequency of 7 days is only about 1°F, the tank temperature cannot exceed its limits before the increased surveillance Frequency takes effect.

SR 3.6.6.2

Verification that the cooling water volume is above the required minimum ensures that a sufficient supply is available for containment cooling. Since the cooling water volume is normally stable and low level is indicated by a main control room alarm, a 7 day Frequency is appropriate and has been shown to be acceptable in similar applications.

SR 3.6.6.3

Verifying the correct alignment of power operated, and automatic valves, excluding check valves, in the Passive Containment Cooling System provides assurance that the proper flow paths exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct positions prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through control room instrumentation or a system walkdown, that valves capable of potentially being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single flow path. This Frequency has been shown to be acceptable through operating experience.

SR 3.6.6.4

This SR requires verification that each automatic isolation valve actuates to its correct position upon receipt of an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this~~ese~~ Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirmed by operating experience) of the equipment. Operating experience has shown that these components usually pass the Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Containment Penetrations

BASES

BACKGROUND

Containment closure capability is required during shutdown operations when there is fuel inside containment. Containment closure is required to maintain within containment the cooling water inventory. Due to the large volume of the IRWST and the reduced sensible heat during shutdown, the loss of some of the water inventory can be accepted. Further, accident analyses have shown that containment closure capability is not required to meet offsite dose requirements. Therefore, containment does not need to be leak tight as required for MODES 1 through 4.

In MODES 5 and 6, the LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no requirement for containment leak tightness, compliance with the Appendix J leakage criteria and tests are not required.

In MODES 5 and 6, there is no potential for steam release into the containment immediately following an accident. Pressurization of the containment could only occur after heatup of the IRWST due to PRHR HX operation (MODE 5 with RCS intact) or after heatup of the RCS with direct venting to the containment (MODE 5 with reduced RCS inventory or MODE 6 with the refueling cavity not fully flooded) or after heatup of the RCS and refueling cavity (MODE 6 with refueling cavity fully flooded). The time from loss of normal cooling until steam release to the containment for four representative sets of plant conditions is shown in Figure B 3.6.8-1 as a function of time after shutdown. Because local manual action may be required to achieve containment closure it is assumed that the containment hatches, air locks and penetrations must be closed prior to steaming into containment.

Figure B 3.6.8-1 provides allowable closure times for four representative sets of plant conditions. The time to steaming is dependent on various plant parameters (RCS temperature, IRWST temperature, etc.) and plant configuration (RCS Pressure Boundary Intact, RCS Open, etc.). Therefore, the actual representation of the time to steaming may be different than that provided in Figure B 3.6.8-1. In determining the minimum time to steaming, conservative assumptions regarding core decay heat, RCS configuration, and initial RCS inventory are used to minimize the calculated time to steaming. The curves are based on the core decay heat prior to refueling so that closure times are longer following the core reload.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 pH Adjustment

BASES

BACKGROUND

The Passive Core Cooling System (PXS) includes four~~two~~ pH adjustment baskets which provide adjustment of the pH of the water in the containment following an accident where the containment floods.

Following an accident with a large release of radioactivity, the containment pH is automatically adjusted to greater than or equal to 7.0, to enhance iodine retention in the containment water. Chemical addition is necessary to counter the affects of the boric acid contained in the safety injection supplies and acids produced in the post-LOCA environment (nitric acid from the irradiation of water and air and hydrochloric acid from irradiation and pyrolysis of electric cable insulation). The desired pH values significantly reduce formation of elemental iodine in the containment water, which reduces the production of organic iodine and the total airborne iodine in the containment. This pH adjustment is also provided to prevent stress corrosion cracking of safety related containment components during long-term cooling.

Dodecahydrate trisodium phosphate (TSP) contained in baskets provides a passive means of pH control for such accidents. The baskets are made of stainless steel with a mesh front that readily permits contact with water. These baskets are located inside containment at an elevation that is below the minimum floodup level. The baskets are placed at least a foot above the floor to reduce the chance that water spills will dissolve the TSP. Natural recirculation of water inside the containment, following a LOCA, is driven by the core decay heat and provides mixing to achieve a uniform pH. The dodecahydrate form of TSP ($Na_3PO_4 \cdot 12H_2O$) is initially loaded into the baskets because it is hydrated and will undergo less physical and chemical change than would anhydrous TSP as a result of the humidity inside containment. (Refs. 1 and 2)

APPLICABLE SAFETY ANALYSES

In the event of a Design Basis Accident (DBA), iodine may be released from the fuel to containment. To limit this iodine release from containment, the pH of the water in the containment sump is adjusted by the addition of TSP. Adjusting the sump water to neutral or alkaline pH ($pH \geq 7.0$) will augment the retention of the iodine, and thus reduce the iodine available to leak to the environment.

pH adjustment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

ACTIONS (continued)

action places the flow path in a condition which assures the safety function is performed. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the SG isolation valves. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk in Reference 3~~6~~.

D.1 and D.2

With two valves in one or more blowdown flow paths inoperable, action must be taken to isolate the flow path with a closed valve. This action places the flow path in a condition which assures the safety function is performed. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the SG isolation valves. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk in Reference 3.

Since the blowdown isolation valve is not deactivated, periodic verification is required to assure that the flow path remains isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of status indications available in the control room, and other administrative controls, to ensure that the valve remains in the closed position.

E.1 and E.2

If the SG isolation valves cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 with the RCS cooling provided by the RNS within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions in an orderly manner and without challenging unit systems.

BASES

ACTIONS (continued)

D.1

With two divisions of DC electrical power distribution subsystems inoperable, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the required DC buses must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or charger.

Condition D represents two subsystems without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining divisions and restoring power to the affected divisions.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected divisions; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 4~~3~~).

BASES

SURVEILLANCE REQUIREMENTS (continued)

readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the Class 1E AC and DC electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. Section 8.3.2, "DC Power Systems."
2. Chapter 6, "Engineering Safety Features."
3. Chapter 15, "Accident Analyses."
4. Regulatory Guide 1.93, "Availability of Electric Power Sources," U.S. Nuclear Regulatory Commission, December 1974.

Table B 3.8.5-1 (page 1 of 1)
Class 1E AC and DC Electrical Power Distribution System

TYPE	VOLTAGE	DIVISION A*	DIVISION B*	DIVISION C*	DIVISION D*
DC Buses	250 Vdc	IDSA-DS-1	IDSB-DS-1 IDSB-DS-2	IDSC-DS-1 IDSC-DS-2	IDSD-DS-1
DC Distribution Panels	250 Vdc	IDSA-DD-1 IDSA-DK-1	IDSB-DD-1 IDSB-DK-1	IDSC-DD-1 IDSC-DK-1	IDSD-DD-1 IDSD-DK-1
AC Instrumentation and Control Buses	120 Vac	IDSA-EA-1	IDSB-EA-1 IDSB-EA-3	IDSC-EA-1 IDSC-EA-3	IDSD-EA-1

* Each Division of the AC and DC electrical power distribution systems is a subsystem.

Change Number 16

TIER 2 REVISION 17 CHANGE ROADMAP (Cont.)

Section	Page No.	Type of Change	10 CFR 52.63 Justification Keyword (note)	Reference NRC Letter
14.2	14.2-77	Technical	(vi) Improve:	
14.2	14.2-78	Technical	(v) Correct:	
14.2	14.2-82	Technical	(iv) Explain:	DCP/NRC 2178
14.3	14.3-19	Technical	(vi) Improve:	DCP/NRC 2111
14.3	14.3-25	Technical	(v) Correct:	DCP/NRC 2020
14.3	14.3-27	Technical	(v) Correct:	DCP/NRC 2020
14.3	14.3-34	Editorial	(v) Correct:	DCP/NRC 1869
14.3	14.3-45	Editorial	(v) Correct:	DCP/NRC 1869
14.3	14.3-49	Editorial	(v) Correct:	DCP/NRC 1952
14.3	14.3-16	Technical	(vii) Standardize:	
14.3	14.3-25	Editorial	(v) Correct:	
14.3	14.3-28	Technical	(vi) Improve:	
14.3	14.3-33	Technical	(v) Correct:	
14.3	14.3-33	Technical	(vi) Improve:	
14.3	14.3-34	Technical	(iv) Explain:	DCP/NRC 2245
14.3	14.3-45	Technical	(iv) Explain:	DCP/NRC 2245
14.3	14.3-46	Technical	(vi) Improve:	
14.3	14.3-49	Technical	(ii) Protect:	
14.4	14.4-1	Editorial	(v) Correct:	DCP/NRC 1895
14.4	14.4-2	Technical	(vi) Improve:	DCP/NRC 1823
14.4	14.4-1	Technical	(vii) Standardize:	DCP/NRC 2161, DCP/NRC 1922
14.4	14.4-1	Editorial	(v) Correct:	
VOLUME 10				
15, T of C	iii thru v	Editorial	(v) Correct:	
15, T of C	viii thru x	Editorial	(v) Correct:	
15, T of C	xv and xvi	Editorial	(v) Correct:	
15, T of C	xviii and xxiii	Editorial	(v) Correct:	
15.0	15.0-23	Editorial	(v) Correct:	DCP/NRC 2020
15.0	15.0-29	Editorial	(v) Correct:	DCP/NRC 2020
15.0	15.0-3 and 15.0-4	Technical	(ii) Protect:	
15.0	15.0-5	Technical	(iv) Explain:	DCP/NRC 2321
15.0	15.0-6	Technical	(v) Correct:	
15.0	15.0-13	Technical	(v) Correct:	
15.0	15.0-15	Technical	(v) Correct:	

TIER 2 REVISION 17 CHANGE ROADMAP (Cont.)

Section	Page No.	Type of Change	10 CFR 52.63 Justification Keyword (note)	Reference NRC Letter
15.0	15.0-21	Technical	(ii) Protect:	DCP/NRC 2182
15.0	15.0-21	Technical	(iv) Explain:	DCP/NRC 2321
15.0	15.0-22 thru 15.0-25	Technical	(v) Correct:	
15.0	15.0-24	Technical	(ii) Protect:	DCP/NRC 2182
15.0	15.0-24	Technical	(ii) Protect:	DCP/NRC 2216
15.0	15.0-24 and 15.0-24	Technical	(vi) Improve:	
15.0	15.0-26	Technical	(iv) Explain:	DCP/NRC 2321
15.0	15.0-27	Technical	(v) Correct:	
15.0	15.0-27 thru 15.0-31	Editorial	(v) Correct:	
15.0	15.0-30	Technical	(v) Correct:	
15.0	15.0-33	Technical	(ii) Protect:	DCP/NRC 2182
15.0	15.0-33	Technical	(v) Correct:	
15.0	15.0-34	Technical	(vi) Improve:	
15.1	15.1-2	Technical	(v) Correct:	
15.1	15.1-15	Editorial	(v) Correct:	
15.1	15.1-24 thru 15.1-27	Technical	(v) Correct:	
15.2	15.2-7 and 15.2-8	Technical	(v) Correct:	
15.2	15.2-10	Technical	(v) Correct:	
15.2	15.2-11	Technical	(iv) Explain:	DCP/NRC 2321
15.2	15.2-12	Technical	(v) Correct:	
15.2	15.2-14	Technical	(iv) Explain:	DCP/NRC 2321
15.2	15.2-14	Technical	(v) Correct:	
15.2	15.2-17 and 15.2-18	Technical	(v) Correct:	
15.2	15.2-18	Technical	(iv) Explain:	DCP/NRC 2321
15.2	15.2-22	Technical	(v) Correct:	
15.2	15.2-26 and 15.2-27	Technical	(v) Correct:	
15.3	15.3-10	Technical	(v) Correct:	
15.3	15.3-12	Technical	(v) Correct:	
15.4	15.4-37	Technical	(v) Correct:	DCP/NRC 1952
15.4	15.4-4	Editorial	(v) Correct:	
15.4	15.4-10 and 15.4-11	Technical	(v) Correct:	
15.4	15.4-10	Editorial	(v) Correct:	
15.4	15.4-19	Technical	(v) Correct:	
15.4	15.4-22	Technical	(v) Correct:	
15.4	15.4-24	Technical	(v) Correct:	
15.4	15.4-28	Technical	(vi) Improve:	DCP/NRC 2190

TIER 2 REVISION 17 CHANGE ROADMAP (Cont.)

Section	Page No.	Type of Change	10 CFR 52.63 Justification Keyword (note)	Reference NRC Letter
15.4	15.4-28	Technical	(vii) Standardize:	
15.4	15.4-42	Technical	(v) Correct:	
15.4	15.4-43	Technical	(iv) Explain:	DCP/NRC 2321
15.4	15.4-45	Technical	(v) Correct:	
15.5	15.5-3	Technical	(iv) Explain:	DCP/NRC 2321
15.5	15.5-4	Technical	(v) Correct:	
15.5	15.5-7	Technical	(iv) Explain:	DCP/NRC 2321
15.5	15.5-8	Technical	(v) Correct:	
15.6	15.6-22	Technical	(vi) Improve:	DCP/NRC 1952, DCP/NRC 2102
15.6	15.6-63	Technical	(vi) Improve:	DCP/NRC 1952, DCP/NRC 2102
15.6	15.6-65	Technical	(vi) Improve:	DCP/NRC 1952, DCP/NRC 2102
15.6	15.6-2	Technical	(vi) Improve:	
15.6	15.6-2	Technical	(v) Correct:	
15.6	15.6-7	Technical	(v) Correct:	
15.6	15.6-12 and 15.6-13	Technical	(v) Correct:	
15.6	15.6-15	Technical	(v) Correct:	
15.6	15.6-17	Technical	(ii) Protect:	DCP/NRC 2182
15.6	15.6-20	Technical	(iv) Explain:	DCP/NRC 2321
15.6	15.6-21 thru 15.6-23	Technical	(ii) Protect:	
15.6	15.6-23	Technical	(v) Correct:	
15.6	15.6-25	Technical	(v) Correct:	
15.6	15.6-24 thru 16.6-30	Technical	(ii) Protect:	DCP/NRC 2182
15.6	15.6-33	Technical	(iv) Explain:	DCP/NRC 2321
15.6	15.6-33	Technical	(v) Correct:	
15.6	15.6-35	Technical	(v) Correct:	
15.6	15.6-38	Technical	(iv) Explain:	DCP/NRC 2321
15.6	15.6-43	Technical	(v) Correct:	
15.6	15.6-44	Technical	(iv) Explain:	
15.6	15.6-46	Technical	(iv) Explain:	
15.6	15.6-52	Technical	(v) Correct:	
15.6	15.6-53	Technical	(ii) Protect:	DCP/NRC 2182
15.6	15.6-53	Technical	(vii) Standardize:	
15.6	15.6-55	Technical	(ii) Protect:	DCP/NRC 2182
15.6	15.6-55	Technical	(vii) Standardize:	
15.6	15.6-52	Technical	(ii) Protect:	

TIER 2 REVISION 17 CHANGE ROADMAP (Cont.)

Section	Page No.	Type of Change	10 CFR 52.63 Justification Keyword (note)	Reference NRC Letter
15.6	15.6-56	Technical	(v) Correct:	
15.6	15.6-58 thru 15.6-60	Technical	(v) Correct:	
15.6	15.6-55	Technical	(ii) Protect:	
15.6	15.6-62 thru 15.6-64	Technical	(ii) Protect:	
15.6	15.6-62 and 15.6-63	Technical	(v) Correct:	
15.6	15.6-65	Technical	(v) Correct:	
15.6	15.6-66	Technical	(ii) Protect:	DCP/NRC 2182
15.6	15.6-68 thru 15.6-70	Technical	(ii) Protect:	DCP/NRC 2182
15.6	15.6-72	Technical	(vi) Improve:	
15.6	15.6-72	Technical	(v) Correct:	
15.6	15.6-78	Technical	(iv) Explain:	
15.6	15.6-94 and 15.6-95	Technical	(v) Correct:	
15.6	15.6-97	Technical	(v) Correct:	
15.6	15.6-99 thru 15.6-111	Technical	(ii) Protect:	DCP/NRC 2182
15.6	15.6-210 thru 15.6-221	Technical	(iv) Explain:	
15.7	15.7-6	Technical	(iv) Explain:	DCP/NRC 2321
15A	15A-18	Technical	(iv) Explain:	DCP/NRC 2099
15A	15A-5	Editorial	(v) Correct:	DCP/NRC 1952
15A	15A-15 and 15A-16	Technical	(iv) Explain:	DCP/NRC 2099
15A	15A-17	Technical	(iv) Explain:	DCP/NRC 2099
15A	15A-4	Technical	(iv) Explain:	DCP/NRC 2321
15A	15A-8 and 15A-9	Technical	(iv) Explain:	DCP/NRC 2321
15A	15A-14 thru 15A-16	Technical	(ii) Protect:	
15A	15A-17	Technical	(iv) Explain:	DCP/NRC 2204
VOLUME 11				
16, T of C	i thru viii	Editorial	(v) Correct:	
16.1	16.1-2	Technical	(iv) Explain:	DCP/NRC 2098
16.1	16.1-2	Technical	(i) Regulate:	
16.1	1.1-5	Editorial	(v) Correct:	
16.1	1.1-6	Technical	(vi) Improve:	
16.1	3.1.4-1	Technical	(vii) Standardize:	
16.1	3.1.4-4	Technical	(iv) Explain:	DCP/NRC 2098
16.1	3.1.8-1	Technical	(v) Correct:	
16.1	3.3.1-2 thru 3.3.1-6	Editorial	(v) Correct:	DCP/NRC 2183
16.1	3.3.1-2 thru 3.3.1-6	Technical	(iv) Explain:	DCP/NRC 2098

Change Number 17

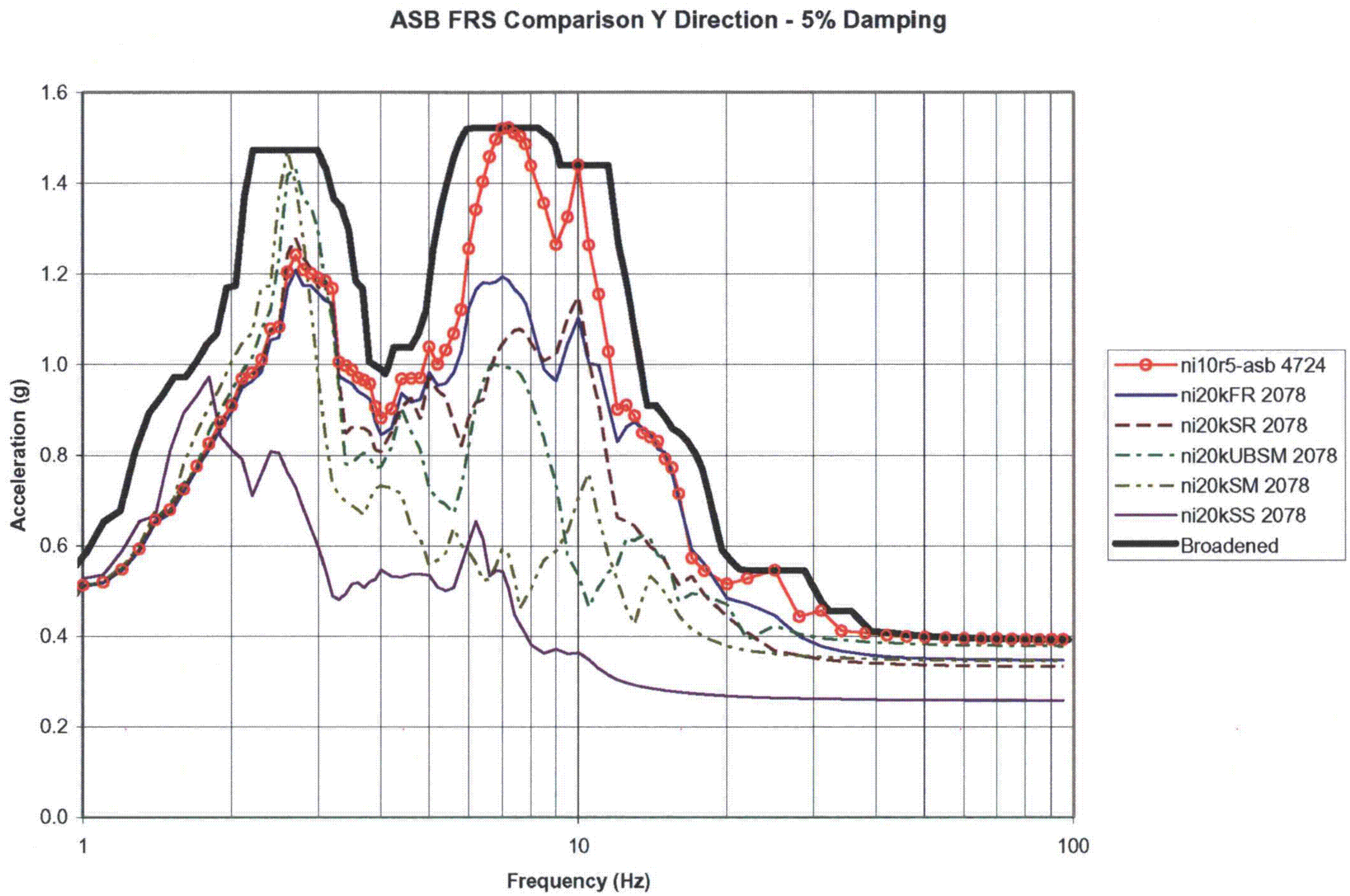


Figure 3G.4-7Y

Y Direction FRS for Node 4724 (NI10) or 2078 (NI20)
ASB Control Room Side Elevation 116.50′

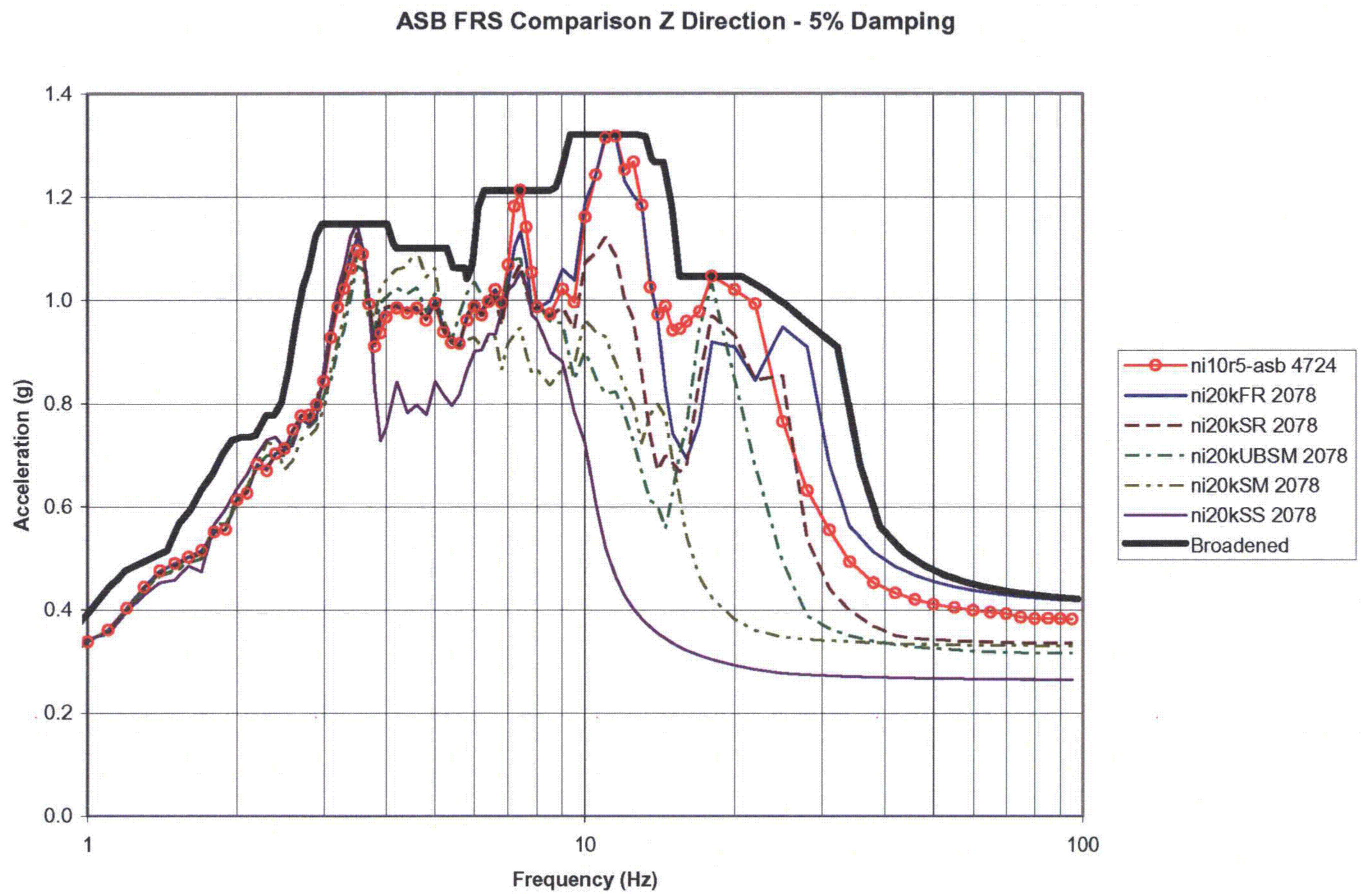


Figure 3G.4-7Z

Z Direction FRS for Node 4724 (NI10) or 2078 (NI20)
ASB Control Room Side Elevation 116.50′

Change Number 18

Table 9.1-3 (Sheet 1 of 2)

COMPONENT DATA –
SPENT FUEL POOL COOLING AND PURIFICATION SYSTEM

Spent Fuel Pool Pump		
Number		2
Design pressure (psig)		150
Design temperature (°F)		250
Nominal flow (gallons/minute)		1200
Minimum flow to support normal cooling (gpm)		900
Material		Stainless Steel
Spent Fuel Pool Heat Exchangers		
Number		2
Type		Plate
Design heat transfer (Btu/hour)		14.75×10^6
Design capacity (Btu/hour-°F)		24.0×10^5
Minimum capacity to support normal cooling (Btu/hour-°F)		22.0×10^5
	Side 1	**Side 2**
Design pressure (psig)	200~~150~~	150
Design temperature (°F)	250	250
Nominal flow (pounds/hour)	6.23×10^5	5.94×10^5
Inlet temperature (°F), typ.	89.5	120
Outlet temperature (°F), typ.	113.3	95.1
Fluid circulated	Component cooling water	Spent fuel pool water
Material	Stainless steel	Stainless steel
Spent Fuel Pool Demineralizers		
Number		2
Design pressure (psig)		150
Design temperature (°F)		200
Nominal flow (gallons/minute)		250
Nominal resin volume (cubic feet)		75
Material		Stainless steel

Change Number 19

Table 5.2-3

ASME CODE CASES

Code Case Number	Title
N-4-11	Special Type 403 Modified Forgings or Bars, Section III, Division 1, Class 1 and Class CS
N-20-4	SB-163 Nickel-Chromium-Iron Tubing (Alloys 600 and 690) and Nickel-Iron-Chromium Alloy 800 at a Specified Minimum Yield Strength of 40.0 ksi and Cold Worked Alloy 800 at Yield Strength of 47.0 ksi, Section III, Division 1, Class 1
N-60-5	Material for Core Support Structures, Section III, Division 1[a]
N-71-18	Additional Material for Subsection NF, Class 1, 2, 3 and MC Component Supports Fabricated by Welding, Section III Division 1
[N-122-2	*Stress Indices for integral Structural Attachments Section III, Division 1, Class 1*]*
N-249-14	Additional Materials for Subsection NF, Class 1, 2, 3, and MC Supports Fabricated Without Welding, Section III, Division 1[b]
[N-284-1	*Metal Containment Shell Buckling Design Methods, Section III, Division 1 Class MC*]*
[N-318-5	*Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping Section III, Division*]*
[N-319-3	*Alternate Procedure for Evaluation of Stresses in Butt Welding Elbows in Class 1 Piping Section III, Division 1*]*
[N-391-2	*Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping Section III, Division 1*]*
[N-392-3	*Procedure for Valuation of the Design of Hollow Circular Cross Section Welded Attachments on Class 2 and 3 Piping Section III, Division 1[c]*]*
N-474-2	Design Stress Intensities and Yield Strength Values for UNS06690 With a Minimum Yield Strength of 35 ksi, Class 1 Components, Section III, Division 1
~~2142-1~~	~~F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Metal, Section IX~~
2142-2	F-Number Grouping for Ni-Cr-Fe Filler Metals Section IX (Applicable to all Sections, including Section III, Division 1, and Section XI)
2143-1	F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode, Section IX
N-655-1	Use of SA-738, Grade B, for Metal Containment Vessels, Class MC, Section III~~II~~, Division 1
N-757-1	Alternative Rules for Acceptability for Class 2 and 3 Valves, NPS 1 (DN25) and Smaller with Welded and Nonwelded End Connections other than Flanges, Section III, Division 1[d]
N-759-2~~1~~	Alternative Rules for Determining Allowable External Pressure and Compressive Stresses for Cylinders, Cones, Spheres, and Formed Heads, Class 1, 2, and 3, Section III, Division 1
N-782	Use of Code Editions, Addenda, and Cases Section III, Division 1

Notes:

(a) Use of this code case will meet the conditions for Code Case N-60-4 in Reg. Guide 1.85 Revision 30.
(b) Use of this code case will meet the conditions for Code Case N-249-10 in Reg. Guide 1.85 Revision 30.
(c) Use of this code case will meet the conditions for Code Case N-392-1 in Reg. Guide 1.84 Revision 30.

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

Change Number 20

For enclosed areas of the containment at least two igniters are installed. The separation between igniter locations is selected to prevent the velocity of a flame front initiated by one igniter from becoming significant before being extinguished by a similar flame front propagating from another igniter. The number of hydrogen igniters and their locations are selected considering the behavior of hydrogen in the containment during severe accidents. The likely hydrogen transport paths in the containment and hydrogen burn physics are the two important aspects influencing the choice of igniter location.

The primary objective of installing an igniter system is to promote hydrogen burning at a low concentration and, to the extent possible, to burn hydrogen more or less continuously so that the hydrogen concentration does not build up in the containment. To achieve this goal, igniters are placed in the major regions of the containment where hydrogen may be released, through which it may flow, or where it may accumulate. The criteria utilized in the evaluation and the application of the criteria to specific compartments is provided in Table 6.2.4-6. The location of igniters throughout containment is provided in Figures 6.2.4-5 through 6.2.4-13. The location of igniters is also summarized in Table 6.2.4-7 identifying subcompartment/regions and which igniters by power group provide protection. The locations identified are considered approximations (± 2.5 feet) with the final locations governed by the installation details.

The igniter assembly is designed to maintain the surface temperature at~~within a range of 1600° to~~ 1700°F in the anticipated containment environment following a loss of coolant accident. A spray shield is provided to protect the igniter from falling water drops (resulting from condensation of steam on the containment shell and on nearby equipment and structures). Design parameters for the igniters are provided in Table 6.2.4-3.

6.2.4.2.4 Containment Purge

Containment purge is not part of the containment hydrogen control system. The purge capability of the containment air filtration system (see subsection 9.4.7) can be used to provide containment venting prior to post-loss of coolant accident cleanup operations.

6.2.4.3 Design Evaluation (Design Basis Accident)

A design basis accident evaluation is not required.

6.2.4.4 Design Evaluation (Severe Accident)

Although a severe accident involving major core degradation or core melt is not a design basis accident, the containment hydrogen control system contains design features to address this potential occurrence. The hydrogen monitoring subsystem has sufficient range to monitor concentrations up to 20 percent hydrogen. The hydrogen ignition subsystem is provided so that hydrogen is burned off in a controlled manner, preventing the possibility of deflagration with supersonic flame front propagation which could result in large pressure spikes in the containment.

It is assumed that 100 percent of the active fuel cladding zirconium reacts with steam. This reaction may take several hours to complete. The igniters initiate hydrogen burns at concentrations less than 10 percent by volume and prevent the containment hydrogen concentration from

Table 6.2.4-3 **COMPONENT DATA - HYDROGEN IGNITER (NOMINAL)**	
Number	64
Minimum Surface Temperature (°F)	~~1600 to~~ 1700

Change Number 21

do not terminate outside the positive pressure boundary without a closed valve, plugged drain, or water seal to maintain the integrity of the positive pressure boundary.

Chemical Waste Drains

The radioactive waste drain system collects chemical wastes from the auxiliary building chemical laboratory and decontamination solution drains from the annex building and directs these wastes to the chemical waste tank of the liquid radwaste system.

Detergent Waste Drains

The laundry and respirator cleaning functions that generate detergent wastes are performed offsite. Detergent wastes from hot sinks and showers are routed to the chemical waste tank.

Oily Waste Drains

The waste water system collects nonradioactive, oily, liquid waste in drain tanks and sumps. Drain tank and sump liquid wastes are pumped through an oil separator prior to further processing. The oil is collected in a tank for disposal.

Sampling for oil in the waste holdup tank of the liquid radwaste system is provided to detect oil contamination before the ion exchanger resins are damaged. Oily water is pumped from the tank through an oil adsorbing bag filter before further processing. The spent bag filters are transferred to drums and stored in the radwaste building as described in Section 11.4.

9.3.5.2.2 Component Description

General description and summaries of the design requirements for these components are provided below. Key equipment parameters are contained in Tables 9.3.5-1 and 11.2-2. Principal construction codes and standards and the classification applicable to the floor and equipment drainage systems are listed in Section 3.2.

Sumps and Drain Tanks

In general, the inlet drain lines to the sump or drain tank are kept submerged a minimum of 6 inches below pump shutoff level to prevent backgassing. The containment sump inlet is submerged.

Sumps are covered to keep out debris. Covers are removable, or manholes are provided for access. The total capacity of each sump includes a 10 percent freeboard allowance to permit operation of high-high level alarms and associated controls before the overflow point is reached.

Each sump is fitted with a vent connection to exhaust potential sump gases ~~into the VAS exhaust system~~. Nonradioactive drain tanks are vented to the atmosphere. The reactor coolant drain tank is vented to the gaseous radwaste system (Section 11.3). Where necessary for the control of airborne radioactivity, the sump vents are routed to the ventilation system exhaust duct for the room.

Change Number 22

Table 5.2-1 (Sheet 4 of 6~~5~~)

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS SPECIFICATIONS

Component	Material	Class, Grade, or Type
RCP piping other than loop and surge line	SA-312[1] and SA-376	TP304, TP304L, TP304LN, TP316, TP316L, TP316LN TP304, TP304L, TP304LN, TP316, TP316L, TP316LN
CRDM		
Latch housing	SA-182~~336~~	F304, F304L, F304LN, F316, F316L, F316LN
Rod travel housing	SA-182~~336~~	F304, F304L, F304LN, F316, F316L, F316LN
Valves		
Bodies	SA-182 or SA-351	F304, F304L, F304LN, F316, F316L, F316LN or CF3A, CF3M, CF8
Bonnets	SA-182 SA-240 or SA-351	F304, F304L, F304LN, F316, F316L, F316LN, 304, 304L, 304LN, 316, 316L, 316LN or CF3A, CF3M, CF8
Discs	SA-182 SA-564 or SA-351	F304, F304L, F304LN, F316, F316L, F316LN Type 630 (H1100 or H1150), or CF3A, CF3M, CF8
Stems	SA-479 SA-564 or SB-637	316, 316LN or XM-19 Type 630 (H1100 or H1150) Alloy N07718
Pressure retaining bolting	SA-453 SA-564 SA-193	GR 660 Type 630 (H1100) GR B8

Change Number 24

9A.3.4.14 Fire Area 4034 AF 01

This fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 4034 AF 40311	40311	Corridor
• 4034 AF 40311	40312	Corridor
• 4034 AF 40311	40373	Corridor
• 4034 AF 40313	40313	Office
• 4034 AF 40313	40314	Office
• ~~4034 AF 40313~~	~~40315~~	~~Office~~
• 4034 AF 40316	40316	Computer Room L1/2
• 4034 AF 40317	40317	Computer Room L3
• 4034 AF 40318	40318	ALARA briefing room and HP monitoring room
• 4034 AF 40320	40320	Women's change room
• 4034 AF 40322	40321	Janitor closet
• 4034 AF 40322	40322	Men's change room
• 4034 AF 40322	40323	Water heater room
• 4034 AF 40322	40324	Drying area
• 4034 AF 40322	40325	Shower room
• 4034 AF 40370	40370	Rest room
• 4034 AF 40370	40371	Conference room
• 4034 AF 40370	40372	Conference room
• 4034 AF 40370	40374	Corridor
• 4034 AF 40370	40375	Conference room
• 4034 AF 40370	40376	Kitchen
• 4034 AF 40370	40377	Rest room
• 4034 AF 40378	40378	Office area west
• 4034 AF 40379	40379	Office area east
• ~~4034 AF 40380~~	~~40380~~	~~Communications room~~

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The general area HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply duct. Fire dampers close automatically on high temperature to isolate portions of this fire area. These actions control the spread of fire and smoke. Other VXS subsystems continue to provide ventilation to the

Table 9A-3 (Sheet 19 of 24)

FIRE PROTECTION SUMMARY

Fire Area/ Zone(1)	Safety Area?(2)	Floor Area Sq Ft	Combust. Material(3)	Fire Sev. Cat.	Amount	Heat Value (Btu)	Comb. Load, Btu/ Sq Ft	Equiv. Dur. (Min)	Boundary Fire Res.(4) (Hours)	Detect. Cap.	Fixed Suppression Capability(5)
4033 AF 01	NO								2/0	SMOKE	HOSE STATION
HOT MACHINE SHOP				ACETYLENE		E	150	3.2E+06			
			LUBE OIL	E	20	3.0E+06					
			WOOD	C	1500	1.3E+07					
			CABLE INS	C	3610	3.7E+07					
			LUBRICANT	E	30	5.9E+05					
			PLASTIC	D	150	2.0E+06					
			RUBBER	D	300	3.7E+06					
			CLOTH	B	100	8.0E+05					
			PAPER	C	50	3.9E+05					
			TRASH	B	50	3.9E+05					
			VOLATILES	E	20	2.7E+06					
FIRE AREA TOTAL:		2290	NET CAT.	E	TOTAL:	6.6E+07	29000	22			
4034 AF 01	NO								2/0	SMOKE	NONE
4034 AF 40311 CORRIDORS 40311, 40312 AND 40373			CABLE INS	C	4000	4.0E+07					
			CLOTH	B	1000	8.0E+06					
			PAPER	C	2000	1.5E+07					
			PLASTIC	D	1000	1.3E+07					
			WOOD	C	1000	8.4E+06					
		3160	NET CAT.	C	TOTAL:	8.5E+07	26700	21			
4034 AF 40313 OFFICES			CABLE INS	C	100	1.0E+06					
			CLOTH	B	100	8.0E+05					
			PAPER	C	3500	2.7E+07					
			PLASTIC	D	125	1.6E+06~~1.7E+06~~					
			WOOD	C	250	2.1E+06					
		456	NET CAT.	C	TOTAL:	3.3E+07	72300~~72400~~	55~~77~~			
4034 AF 40316 COMPUTER ROOM L1/2			CABLE INS	C	1000	1.0E+07					
			PLASTIC	D	650	8.6E+06					
			WOOD	C	125	1.0E+06					
			CLOTH	B	50	4.0E+05					
		554	NET CAT.	C	TOTAL:	2.0E+07	36000~~36500~~	26~~39~~			
4034 AF 40317 COMPUTER ROOM L3			CABLE INS	C	1000	1.0E+07					
			PLASTIC	D	100	1.3E+06					
			WOOD	C	250	2.1E+06					
			CLOTH	B	50	4.0E+05					
		571	NET CAT.	C	TOTAL:	1.4E+07	24500	19~~26~~			
4034 AF 40318 ALARA BRIEFING AND HP MONITORING ROOM			CABLE INS	C	500	5.0E+06~~5.1E+06~~					
			CLOTH	B	250	2.0E+06~~2.6E+06~~					
			PAPER	C	2000	1.5E+07					
			PLASTIC	D	125	1.6E+06~~1.7E+06~~					
			WOOD	C	250	2.1E+06					
		799	NET CAT.	C	TOTAL:	2.7E+07~~2.6E+07~~		34000~~32500~~	25~~35~~		

Table 9A-3 (Sheet 20 of 24)

FIRE PROTECTION SUMMARY

Fire Area/ Zone(1)	Safety Area?(2)	Floor Area Sq Ft	Combust. Material(3)	Fire Sev. Cat.	Amount	Heat Value (Btu)	Comb. Load, Btu/ Sq Ft	Equiv. Dur. (Min)	Boundary Fire Res.(4) (Hours)	Detect. Cap.	Fixed Suppression Capability(5)
4034 AF 40320			CABLE INS	C	1000	1.0E+07					
WOMEN'S CHANGE			CLOTH	B	760	6.1E+06					
ROOM			PAPER	C	560	4.3E+06					
			PLASTIC	D	280	3.7E+06					
			TRASH	B	50	3.9E+05					
			WOOD	C	2300	1.9E+07					
		1230	NET CAT.	C	TOTAL:	4.4E+07	36000	26			
4034 AF 40322			CABLE INS	C	1000	1.0E+07					
MEN'S CHANGE			CLOTH	B	1740	1.4E+07					
ROOM			PAPER	C	1440	1.1E+07					
			PLASTIC	D	720	9.5E+06					
			TRASH	B	100	7.7E+05					
			WOOD	C	5500	4.6E+07					
		2860	NET CAT.	C	TOTAL:	9.2E+07	32000	24			
4034 AF 40370			CABLE INS	C	1000	1.0E+07					
KITCHEN, CONF. ROOMS,			CLOTH	B	500	4.0E+06					
REST ROOMS, CORRIDOR			PAPER	C	500	3.9E+06					
			PLASTIC	D	500	6.6E+06					
			TRASH	B	150	1.2E+06					
			WOOD	C	500	4.2E+06					
		1500	NET CAT.	C	TOTAL:	3.0E+07	20000	15			
4034 AF 40378			CABLE INS	C	575	5.7E+06					
OFFICE AREA WEST			CLOTH	B	575	4.6E+06					
			PAPER	C	20200	1.5E+08					
			PLASTIC	D	720	9.0E+06					
			WOOD	C	1440	1.2E+07					
		3165~~2895~~	NET CAT.	C	TOTAL:	1.8E+08	56875~~62625~~	46~~54~~			
4034 AF 40379			CABLE INS	C	1320	1.3E+07					
OFFICE AREA EAST			CLOTH	B	1320	1.1E+07					
			PAPER	C	46200	3.6E+08					
			PLASTIC	D	1650	2.2E+07					
			WOOD	C	3300	2.8E+07					
		6700	NET CAT.	C	TOTAL:	4.3E+08	64800	55			
~~4034 AF 40380~~			~~CABLE INS~~	~~C~~	~~1000~~	~~1.0E+07~~					
~~COMMUNICATIONS ROOM~~			~~PLASTIC~~	~~D~~	~~650~~	~~8.6E+06~~					
		~~270~~	~~NET CAT.~~	~~C~~	~~TOTAL:~~	~~1.9E+07~~	~~68900~~	~~51~~			
FIRE AREA TOTAL:		20995	NET CAT.	C	TOTAL:	9.6E+08~~9.8E+08~~	45725~~47540~~	34~~35~~			
4035 AF 01	NO		FUEL OIL	E	650	9.4E07			3/0	NONE	DRY PIPE
ANCILLARY DIESEL			LUBE OIL	E	16	2.4E06					SPRINKLERS
GENERATOR			CABLE INS	C	20	2.0E05					HOSE STATION
ROOM			PLASTIC	D	20	2.6E05					
			VOLATILES	E	10	1.4E06					
			CLOTH	B	10	8.0E04					
FIRE AREA TOTAL		230	NET CAT.	E	TOTAL	9.8E07	426,000	320			

Change Number 25

5.3 Reactor Vessel

5.3.1 Reactor Vessel Design

5.3.1.1 Safety Design Bases

The reactor vessel, as an integral part of the reactor coolant pressure boundary will be designed, fabricated, erected and tested to quality standards commensurate with the requirements set forth in 10 CFR 50, 50.55a and General Design Criterion 1. Design and fabrication of the reactor vessel is carried out in accordance with ASME Code, Section III, Class 1 requirements. Subsections 5.2.3 and 5.3.2 provide further details.

The performance and safety design bases of the reactor vessel follow:

- The reactor vessel provides a high integrity pressure boundary to contain the reactor coolant, heat generating reactor core, and fuel fission products. The reactor vessel is the primary pressure boundary for the reactor coolant and the secondary barrier against the release of radioactive fission products.
- The reactor vessel provides support for the reactor internals, flow skirt, and core to ensure that the core remains in a coolable configuration.
- The reactor vessel directs main coolant flow through the core by close interface with the reactor internals and flow skirt.
- The reactor vessel provides for core internals location and alignment.
- The reactor vessel provides support and alignment for the control rod drive mechanisms and in-core instrumentation assemblies.
- The reactor vessel provides support and alignment for the integrated head assembly.
- The reactor vessel provides an effective seal between the refueling cavity and sump during refueling operations.
- The reactor vessel supports and locates the main coolant loop piping.
- The reactor vessel provides support for safety injection flow paths.
- The reactor vessel serves as a heat exchanger during core meltdown scenario with water on the outside surface of the vessel.

5.3.1.2 Safety Description

The reactor vessel consists of a cylindrical section with a transition ring, hemispherical bottom head, and a removable flanged hemispherical upper head (Figure 5.3-1). Key dimensions are shown in Figures 5.3-5 and 5.3-6. The cylindrical section consists of two shells, the upper shell and the lower shell. The upper and lower shells and the lower hemispherical head are fabricated smoke into the stairwell. For fires not affecting the ventilation system, the system continues to provide ventilation to the stairwell, minimizing the potential for migration of smoke.

9A.3.4.3B Fire Area 4002 AF 03

This fire area is comprised of the elevator shaft and~~,~~ elevator ~~machine room and security room~~.

This elevator is enclosed by fire barrier walls having a minimum rating of 2 hours. ~~The west wall of elevator shaft, elevator machine room and security room are exposed to the annex building interior, and this wall is constructed with a concrete/steel composite material having a minimum fire rating of 2 hours. The walls of the enclosures that face the yard area would not be exposed to the annex building interior. Therefore, these outside walls are constructed with an exterior siding common to the overall siding used for the annex building. The wall between the elevator and the stairwell also has a fire barrier wall with a minimum rating of 2 hours. The floor of the security room has a minimum rating of 2 hours.~~

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

A wall exhaust fan and air intake louvers provide normal ventilation for the elevator shaft ~~and elevator machine room~~. After the fire, smoke is removed from the fire area by using the wall exhaust fan or portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and the security central alarm station. The fire is extinguished manually using hose streams or portable extinguishers.

Combustible materials in this fire area are listed in Table 9A-3, and they consist primarily of cable insulation and lubricant associated with the elevator hoisting machinery~~, paper and cable insulation in the security room~~. This is a light hazard fire area, and the rate of fire growth is expected to be slow. Two-hour fire barriers provide adequate separation from adjacent fire areas since the fire will be contained within this fire area.

9A.3.4.4 Fire Area 4002 AF 02

This fire area is comprised of the following room(s):

Room No.	
S04	Stairwell

Table 9A-3 (Sheet 5 of 24)

FIRE PROTECTION SUMMARY

Fire Area/ Zone(1)	Safety Area?(2)	Floor Area Sq Ft	Combust. Material(3)	Fire Sev. Cat.	Amount	Heat Value (Btu)	Comb. Load, Btu/ Sq Ft	Equiv. Dur. (Min)	Boundary Fire Res.(4) (Hours)	Detect. Cap.	Fixed Suppression Capability(5)
1250 AF 12561			CABLE INS	C	1000	1.0E+07					HOSE STATION
CCS VALVE ROOM AND			VOLATILES	E	5	6.8E+05					
ACCESS CORRIDOR		630	NET CAT.	D	TOTAL:	1.1E+07	17000	13			
1254 AF 12553			CABLE INS	C	5000	5.1E+07					
ELEVATION 135'-3"			PAPER	C	1000	7.7E+06					
PERSONNEL ACCESS			TRASH	B	500	3.9E+06					
AREA			VOLATILES	E	10	1.4E+06					
		1350	NET CAT.	D	TOTAL:	6.4E+07	47000	36			
1244 AF 12451			CABLE INS	C	400	4.1E+06					
SECURITY ROOM~~S~~			PAPER	C	500	3.8E+06					
		308	NET CAT.	C	TOTAL:	7.9E+06	25900~~25700~~	22~~27~~			
1254 AF 12554			CABLE INS	C	400	4.1E+06					
SECURITY ROOM			PAPER	C	1000	7.7E+06					
		268	NET CAT.	C	TOTAL:	1.2E+07	62000	51			
1264 AF 12651			CABLE INS	C	5000	5.1E+07					
VAS EQUIPMENT			LUBE OIL	E	10	1.5E+06					
ROOM			VOLATILES	E	10	1.4E+06					
		1480	NET CAT.	D	TOTAL:	5.4E+07	36000	27			
FIRE AREA TOTAL:		22501	NET CAT.	D	TOTAL:	6.9E+08	30700	24~~29~~			HOSE STATION
1200 AF 02	YES								3/0	SMOKE	SEE ZONE
1200 AF 12562			CABLE INS	C	10000	1.0E+08					HOSE STATION
FUEL HANDLING AREA			PAPER	C	1500	1.2E+07					
			WOOD	C	1000	8.4E+06					
			TRASH	B	1000	7.7E+06					
			CLOTH	B	500	4.0E+06					
			PLASTIC	D	500	6.6E+06					
			LUBE OIL	E	15	2.3E+06					
			VOLATILES	E	5	6.8E+05					
		4725	NET CAT.	D	TOTAL:	1.4E+08	30000	23			
1230 AF 12371			CABLE INS	C	3000	3.1E+07					WET PIPE(6)
RAIL CAR BAY/			PAPER	C	1000	7.7E+06					SPRINKLER
FILTER STORAGE AREA			WOOD	C	1000	8.4E+06					HOSE STATION
			TRASH	B	1000	7.7E+06					
			LUBE OIL	E	10	1.5E+06					
			FUEL OIL	E	100	1.4E+07					
		1460	NET CAT.	D	TOTAL:	7.0E+07	48000	37			
1236 AF 12372			CABLE INS	C	200	2.0E+06					HOSE STATION
RESIN TRANSFER PUMP/			LUBE OIL	E	5	7.6E+05					
VALVE ROOM		80	NET CAT.	E	TOTAL:	2.8E+06	35000	26			
1236 AF 12373			CABLE INS	C	200	2.0E+06					HOSE STATION
SPENT RESIN TANK ROOM		70	NET CAT.	C	TOTAL:	2.0E+06	29000	22			
1246 AF 12471			CABLE INS	C	200	2.0E+06					HOSE STATION
WSS VALVE/PIPING AREA		90	NET CAT.	C	TOTAL:	2.0E+06	23000	17			
FIRE AREA TOTAL:		6425	NET CAT.	D	TOTAL:	2.2E+08	34000	26			HOSE STATION

level are required through the second time frame. The only long term application is the containment pressure transmitter which may eventually be impacted by the severe accident radiation dose.

19D.8.2.2 Thermocouples

The function~~s~~ defined for severe accident management that utilizes thermocouples is ~~are~~ core-exit temperature ~~and containment water level~~. The core-exit temperature is only required during Time Frame 1 ~~and the containment water level is required through Time Frame 2~~. The temperatures to which the thermocouples are exposed during the defined time frames do not exceed the thermocouple design.

19D.8.2.3 Resistance Temperature Detectors (RTDs)

Both hot and cold leg temperatures are defined as parameters for severe accident management in Time Frame 1. RTDs are utilized for these measurements and will perform until their temperature range is exceeded. The hot leg RTDs could fail as the temperature increases well above the design conditions of the RTDs but the cold leg RTDs should perform throughout Time Frame 1. RTDs are also utilized through Time Frame 3 for the containment temperature measurement and are exposed to temperature transients that exceed design basis qualification conditions. EPRI NP-4354 documents RTD performance during several temperature transients with acceptable results.

19D.8.2.4 Hydrogen Monitors

Containment hydrogen is defined as a parameter to be monitored throughout the severe accident scenarios. Note that the design of the hydrogen monitors has not been finalized and both in-containment and outside containment monitors are being considered. Early in the accident, the hydrogen may be monitored by a device that operates on the basis of catalytic oxidation of hydrogen on a heated element. The hydrogen monitors are located in the main containment area. The design limits of this device may be exceeded after the first few hours of some of the postulated accidents and performance may be uncertain. If the device fails, hydrogen concentration may be determined through online containment hydrogen monitors located in the auxiliary building that take a sample from the containment atmosphere. Also, post-accident sampling of containment atmosphere using analysis of grab samples may be used to determine containment hydrogen concentrations.

19D.8.2.5 Radiation Monitors

Containment radiation is defined as a parameter to be monitored throughout the severe accident scenarios. The containment radiation monitors are located in the main containment area. Early in the accident, the design basis event qualified containment radiation monitor provides the necessary information until the environment exceeds the design limits of the monitor. If the device fails, containment radiation is determined through the containment atmosphere sampling function or by portable monitors located against the outside of the containment shell.

19D.8.2.13 Float Level Sensors

The function defined for severe accident management that uses float level sensors is containment water level. Containment water level is required through Time Frame 2. The temperature to which the float level sensors are exposed during the time frame does not exceed the float level sensor design.

19D.8.2.1~~3~~4 Assessment of Equipment for Sustained Burning

The equipment necessary for equipment survivability in sustained burning environments is defined in Tables 19D-3 through 19D-5. The equipment in Table 19D-3 includes equipment and instrumentation operation during Time Frame 1 - core uncovery and heatup, and is prior to the release of significant quantities of hydrogen. Therefore, it does not have to be qualified for sustained hydrogen burning. Table 19D-7 specifies the equipment and instrumentation used in Time Frames 2 and 3 to provide reasonable assurance of achieving a controlled stable state.

19D.8.3 Equipment Located Outside Containment

Other functions defined for severe accident management are performed outside containment and the equipment is not subjected to the harsh environment of the event. This equipment includes, but is not limited to:

- Steam line radiation monitor,
- Transmitters for monitoring steam line pressure,
- Passive containment cooling system flow and tank level,
- Containment atmosphere sampling function,
- Makeup pumps and flow measurement,
- RNS pumps and flow measurement,
- SFS pumps and flow measurement,
- RNS MOVs
- MFW pumps and valves,
- SFW pumps and valves,
- Steam generator PORVs and main steam bypass valves for depressurization,
- Recirculation pumps, PCS valves and fire water pumps and valves for containment heat removal,
- Containment isolation valves (outside containment),
- Auxiliary building radiation monitor,
- MOV and manual valve from RNS hot leg suction lines to the spent fuel pool and
- Fire water, fire pumps, valves and flow measurement used to provide containment spray and backup containment cooling.

19D.9 Conclusions of Equipment Survivability Assessment

The equipment defined for severe accident management was reviewed for performance during the environments postulated for these events. Survivability of the equipment was evaluated based on design basis event qualification testing, severe accident testing, and the survival time required following the initiation of the severe accident. The equipment that is qualified for design basis

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Level instrumentation on the surge tank provides both high- and low-level alarms. At tank low-level, makeup is provided from the demineralized transfer and storage system. At a low-low-level point in the tank, a signal is sent to stop the hot water heating system pumps.

9.2.11 Combined License Information

9.2.11.1 Potable Water

The Combined License applicant will address the components of the potable water system outside of the power block, including supply source required to meet design pressure and capacity requirements, specific chemical selected for use as a biocide, and any storage requirements deemed necessary. A biocide such as sodium hypochlorite is recommended. Toxic gases such as chlorine are not recommended. The impact of toxic gases on the main control room habitability~~compatibility~~ is addressed in Section 6.4.

9.2.11.2 Waste Water Retention Basins

The Combined License applicant will address the final design and configuration of the plant waste water retention basins and associated discharge piping, including piping design pressure, basin transfer pump size, basin size, and location of the retention basins.

9.2.12 References

1. ASME Code, Section IV, Pt. HWL, 1998.
2. Uniform Plumbing Code, Section 318, 2000.

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Table 10.1-1

SIGNIFICANT DESIGN FEATURES AND PERFORMANCE CHARACTERISTICS FOR MAJOR STEAM AND POWER CONVERSION SYSTEM COMPONENTS

Nuclear Steam Supply System, Full Power Operation	
Rated NSSS power (MWt)	3415
Steam generator outlet pressure (psig)	821~~823~~
Steam generator inlet feedwater temperature (°F)	440
Maximum steam generator separator outlet steam moisture (%)	0.25
Steam generator outlet steam temperature (°F)	523
Quantity of steam generators	2
Flow rate per steam generator (lb/hr)	7.49×10^6
Turbine	
Nominal output (kW)	1,199,500
Turbine type	Tandem-compound 6-flow, 52 in. last-stage blade
Turbine elements	1 high pressure 3 low pressure
Operating speed (rpm)	1800

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- Limits pressure differentials on internal steam generator components, particularly the tube support plates

The restrictor is configured to minimize the unrecovered pressure loss across the restrictor during normal operation.

5.4.4.2 Design Description

The flow restrictor consists of seven nickel-chromium-iron Alloy 690 (ASME SB-564) venturi inserts which are installed in holes in an integral steam outlet nozzle forging. The inserts are arranged with one venturi at the centerline of the outlet nozzle, and the other six are equally spaced around it. After insertion into the nozzle forging holes, the venturi inserts are welded to the nickel-chromium-iron alloy buttering~~cladding~~ on the inner surface of the forging.

5.4.4.3 Design Evaluation

The flow restrictor design has been analyzed to determine its structural adequacy. The equivalent throat area of the steam generator outlet is 1.4 square feet. The resultant pressure drop through the restrictor at 100 percent steam flow is approximately 15 psig. This is based on a design flow rate of 7.49 x 10^6 pounds per hour. Materials of construction of the flow restrictor are in accordance with Code Class 1 Section III of the ASME Code. The material of the inserts is not an ASME Code pressure boundary, nor is it welded to an ASME Code pressure boundary. The method for seismic analysis is dynamic.

5.4.4.4 Inspections

Since the restrictor is not part of the steam system pressure boundary, inservice inspections are not required.

5.4.5 Pressurizer

The pressurizer provides a point in the reactor coolant system where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control of the reactor coolant system during steady-state operations and transients. The pressurizer provides a controlled volume from which level can be measured.

The pressurizer contains the water inventory used to maintain reactor coolant system volume in the event of a minor system leak for a reasonable period without replenishment. The pressurizer surge line connects the pressurizer to one reactor coolant hot leg. This allows continuous coolant volume and pressure adjustments between the reactor coolant system and the pressurizer.

5.4.5.1 Design Bases

The pressurizer is sized to meet following requirements:

- The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.

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Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
C.1(b)		Conforms	The welding procedures are qualified within the preheat temperature ranges required by ASME Code, Section IX. Experience has shown excellent quality of welds using the ASME qualification procedures.
C.2		Exception	The AP1000 position is that the guidance specified in this regulatory guide is both unnecessary and impractical. Code acceptable low-alloy steel welds have been and are being made under present procedures. It is not necessary to maintain the preheat temperature until a post-weld heat treatment has been performed in accordance with the guidance provided by this regulatory guide, in the case of large components. In some cases of reactor vessel main structural welds, the practice of maintaining preheat until the intermediate or final post-weld heat treatment has been followed. In other cases, an extended preheat practice has been utilized in accordance with the reactor vessel design specification. In this practice, the weld temperature is maintained at 400°F to 750°F for 4 hours after welding. The weld temperature may then be lowered to ambient without performing an intermediate or final post-weld heat treatment ~~pressurized water heat transfer~~ at 1100°F. The welds have shown high integrity. Westinghouse practices are documented in WCAP-8577 (Reference 9) which has been accepted by the Nuclear Regulatory Commission.

Reg. Guide 1.51 – Withdrawn

Reg. Guide 1.52, Rev. 3 6/01~~2, 3/78~~ – Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants

General		Conforms~~N/A~~	The AP1000 main control room emergency habitability system (VES) includes a passive filtration system that is contained entirely within the main control room envelope. The passive filtration portion of the AP1000 VES contains no active equipment. This regulatory guide is followed where appropriate for the AP1000 MCR passive filtration system.~~There are no ESF atmosphere cleanup systems for the AP1000. The AP1000 does not require engineered safety feature atmosphere cleanup systems to meet limits on doses offsite or onsite.~~

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Table 3.2-3 (Sheet 10 of ~~65~~69)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Containment Cooling System (Continued)					
PCS-PL-V013B	Flow Transmitter FT004 Root Valve	C	I	ASME III-3	
PCS-PL-V015	Water Bucket Makeup Line Drain Valve	C	I	ASME III	
PCS-PL-V016	PCCWST Drain Isolation Valve	C	I	ASME III-3	
PCS-PL-V017	Chemical Addition Tank Vent Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V018	Recirculation Pump Throttle Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V019	Chemical Addition Tank Fill Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V020	Water Bucket Makeup Line Isolation Valve	C	I	ASME III-3	
PCS-PL-V021	PCCWST TO Recirculation Pump Suction Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V022	Chemical Addition Tank Drain Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V023	PCS Recirculation Return Isolation	C	I	ASME III-3	
PCS-PL-V025	Pressure Transmitter PT 031 Root Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V026	Makeup to Distribution Bucket Isolation Valve	C	I	ASME III-3	
PCS-PL-V029	PCCWST Isolation Valve Leakage Detection Drain	C	I	ASME III-3	

Table 3.11-1 (Sheet 33 of 5~~0~~1)

ENVIRONMENTALLY QUALIFIED ELECTRICAL AND MECHANICAL EQUIPMENT

Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Makeup to Distribution Bucket Isolation Valve	PCS-PL-V026	10	PB	1 yr	M
~~PCCWST Isolation Valve Leakage~~ Detection Drain	~~PCS-PL-V029~~	~~9~~	~~PB~~	~~1 yr~~	~~M~~
PCCWST Isolation Valve Leakage Detection Crossconn	PCS-PL-V030	9	PB	1 yr	M
PCCWST Level Instrument Root Valve	PCS-PL-V031A	9	PB	1 yr	M
PCCWST Level Instrument Root Valve	PCS-PL-V031B	9	PB	1 yr	M
Recirculation Pump Suction from Long Term Makeup Isolation Valve	PCS-PL-V033	10	ESF	2 wks	M *
Spent Fuel Pool Emergency Makeup Isolation	PCS-PL-V052	7	PB	1 yr	M
Hot Leg 1 Sample Isolation Limit Switch	PSS-PL-V001A PSS-PL-V001A-L	1 1	PB PAMS	1 yr 1 yr	M * E *
Hot Leg 2 Sample Isolation Limit Switch	PSS-PL-V001B PSS-PL-V001B-L	1 1	PB PAMS	1 yr 1 yr	M * E *
Pressurizer Sample Isolation	PSS-PL-V003	1	PB	1 yr	M *
PXS Accumulator Sample Isolation	PSS-PL-V004A	1	PB	1 yr	M *
PXS Accumulator Sample Isolation	PSS-PL-V004B	1	PB	1 yr	M *
PXS CMT A Sample Isolation	PSS-PL-V005A	1	PB	1 yr	M *
PXS CMT B Sample Isolation	PSS-PL-V005B	1	PB	1 yr	M *
PXS CMT A Sample Isolation	PSS-PL-V005C	1	PB	1 yr	M *
PXS CMT B Sample Isolation	PSS-PL-V005D	1	PB	1 yr	M *
Liquid Sample Check Valve	PSS-PL-V012A	1	PB	1 yr	M *
Liquid Sample Check Valve	PSS-PL-V012B	1	PB	1 yr	M *
Containment Testing Boundary Isolation Valve	PSS-PL-V076A	1	PB	1 yr	M *
Containment Testing Boundary Isolation Valve	PSS-PL-V076B	1	PB	1 yr	M *
Containment Isolation Test Connection Isolation Valve	PSS-PL-V082	1	PB	1 yr	M *
Containment Isolation Test Connection Isolation Valve	PSS-PL-V083	1	PB	1 yr	M *
~~MCR Potable Water Inlet Check Valve~~	~~PWS-PL-V418~~	~~3~~	~~PB~~	~~1 y~~	~~M~~

Table 3I.6-3 (Sheet 13 of 32)

LIST OF AP1000 SAFETY-RELATED ELECTRICAL AND MECHANICAL EQUIPMENT NOT HIGH FREQUENCY SENSITIVE

Description	AP1000 Tag Number	Comment
Flow Transmitter FT003 Root Valve	PCS-PL-V012B	2
Flow Transmitter FT004 Root Valve	PCS-PL-V013A	2
Flow Transmitter FT004 Root Valve	PCS-PL-V013B	2
PCCWST Drain Isolation Valve	PCS-PL-V016	2
Makeup to Distribution Bucket Isolation Valve	PCS-PL-V026	2
PCCWST Isolation Valve Leakage Detection Drain	PCS-PL-V029	2
PCCWST Isolation Valve Leakage Detection Crossconn	PCS-PL-V030	2
PCCWST Level Instrument Root Valve	PCS-PL-V031A	2
PCCWST Level Instrument Root Valve	PCS-PL-V031B	2
Recirculation Pump Suction from Long Term Makeup Isolation Valve	PCS-PL-V033	2
Spent Fuel Pool Emergency Makeup Isolation	PLS-PL-V052	2
Hot Leg 1 Sample Isolation	PSS-PL-V001A	2
Hot Leg 2 Sample Isolation	PSS-PL-V001B	2
Pressurizer Sample Isolation	PSS-PL-V003	2
PXS Accumulator Sample Isolation	PSS-PL-V004A	2
PXS Accumulator Sample Isolation	PSS-PL-V004B	2
PXS CMT A Sample Isolation	PSS-PL-V005A	2
PXS CMT B Sample Isolation	PSS-PL-V005B	2
PXS CMT A Sample Isolation	PSS-PL-V005C	2
PXS CMT B Sample Isolation	PSS-PL-V005D	2
Liquid Sample Check Valve	PSS-PL-V012A	2
Liquid Sample Check Valve	PSS-PL-V012B	2
Containment Testing Boundary Isolation Valve	PSS-PL-V076A	2
Containment Testing Boundary Isolation Valve	PSS-PL-V076B	2
Containment Isolation Test Connection Isolation Valve	PSS-PL-V082	2

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Table 1.8-1 (Sheet 1 of 6~~7~~)

SUMMARY OF AP1000 PLANT INTERFACES WITH REMAINDER OF PLANT

Item No.	Interface	Interface Type	Matching Interface Item	Section or Sub-section
~~1.1~~	~~Post accident Radio Iodine sampling capability per NUREG 0737~~	~~Requirement of AP1000~~	~~Combined License applicant program~~	~~1.9.3~~
2.1	Envelope of AP1000 plant site related parameters	Site Interface	Site specific parameters	2.0
2.2	External missiles from man-made hazards and accidents	Site Interface	Site specific parameters	2.2
2.3	Maximum loads from man-made hazards and accidents	Site Interface	Site specific parameters	2.2
2.4	Limiting meteorological parameters (χ/Q) for design basis accidents and for routine releases and other extreme meteorological conditions for the design of systems and components exposed to the environment.	Site Interface	Site specific parameters	2.3
2.5	Tornado and operating basis wind loadings	Site Interface	Site specific parameters	2.3
2.6	External missiles generated by natural phenomena	Site Interface	Site specific parameters	2.3
2.7	Snow, ice and rain loads	Site Interface	Site specific parameters	2.3
2.8	Ambient air temperatures	Site Interface	Site specific parameters	2.3
2.9	Onsite meteorological measurement program	Requirement of AP1000	Combined License applicant program	2.3.3
2.10	Flood and ground water elevations	Site Interface	Site specific parameters	2.4
2.11	Hydrostatic loads on systems, components and structures	Site Interface	Site specific parameters	2.4
2.12	Seismic parameters peak ground acceleration response spectra shear wave velocity	Site Interface	Site specific parameters	 2.5 2.5 2.5
2.13	Required bearing capacity of foundation	Site Interface	Site specific	2.5

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Table 2.6.9-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The external walls, doors, ceiling, and floors in the main control room, the central alarm station, and the secondary alarm station ~~last access control function for access to the protected area~~ are bullet-resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.	See Tier 1 Material, Table 3.3-6, item 14.	See Tier 1 Material, Table 3.3-6, item 14.
2. Not used.~~Physical barriers for the protected area perimeter are not part of the vital area barriers.~~	~~An inspection of the protected area perimeter barrier will be performed to verify that physical barriers at the perimeter of the protected area are separated from any other barrier designated as a vital area barrier.~~	~~Physical barriers at the perimeter of the protected area are separated from any other barrier designated as a vital area barrier.~~
3. Secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within the vital area.	See Tier 1 Material, Table 3.3-6, item 16.	See Tier 1 Material, Table 3.3-6, item 16.
4. Vital areas are locked and alarmed with active intrusion detection systems that annunciate in the central and secondary alarm stations upon intrusion into a vital area.	See Tier 1 Material, Table 3.3-6, item 17.	See Tier 1 Material, Table 3.3-6, item 17.
5.a) Security alarm annunciation and video assessment information is displayed concurrently~~occurs~~ in the central alarm station and the secondary alarm station, and the video image recording with real time playback capability can provide assessment of activities before and after each alarm annunciation within the perimeter area barrier~~in at least one other continuously manned station not necessarily onsite~~.	Test, inspection, or a combination of test and inspections of the installed systems will be performed~~to ensure that security alarms annunciate in the central alarm station and in at least one other continuously manned station~~.	Security alarms annunciation and video assessment information is displayed concurrently~~annunciate~~ in the ~~continuously manned~~ central alarm station and the secondary alarm station, and the video image recording with real time playback capability provides assessment of activities before and after alarm annunciation within the perimeter barrier~~located within the protected area and in at least one other continuously manned station~~.

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Table 9A-3 (Sheet 12 of 24)

FIRE PROTECTION SUMMARY

Fire Area/ Zone[1]	Safety Area?[2]	Floor Area Sq Ft	Combust. Material[3]	Fire Sev. Cat.	Amount	Heat Value (Btu)	Comb. Load, Btu/ Sq Ft	Equiv. Dur. (Min)	Boundary Fire Res.[4] (Hours)	Detect. Cap.	Fixed Suppression Capability[5]
2000 AF 01	NO										
				0	SEE ZONE	SEE ZONE					
2030 AF 20300			CABLE INS	C	87,800	9.0E+08				HEAT	WET PIPE
ELEVATION 100'-0"			LUBE OIL	E	4050	6.1E+08					SPRINKLERS[6]
(BASE SLAB)			PLASTIC	D	12,500	1.6E+08~~5.1E+08~~					HOSE STATION
GENERAL FLOOR AREA			VOLATILES	E	375	5.1E+07~~3.4E+07~~					
			FUEL OIL	E	125	1.8E+07					
			TRASH	B	1000	7.7E+06					
		38062	NET CAT.	E	TOTAL:	1.8E+09	47,300	35			
2038 AF 20300			CABLE INS	C	3000	3.1E+07				HEAT	PREACTION
MAIN FEEDWATER			LUBE OIL	E	2250	3.4E+08					SPRINKLERS
PUMP AREA			PLASTIC	D	150	2.0E+06					
			VOLATILES	E	55	7.5E+06					
			TRASH	B	200	1.5E+06					
		4542	NET CAT.	E	TOTAL:	3.8E+08	83670	63			
2039 AF 20301			CABLE INS	C	300	3.1E+06				HEAT	WATER SPRAY
CHEMICAL			LUBE OIL	E	250	3.8E+07					HOSE STATION
STORAGE AREA			PLASTIC	D	125	1.7E+06					
			VOLATILES	E	600	8.2E+07					
			TRASH	B	250	1.9E+06					
		1684	NET CAT.	E	TOTAL:	1.3E+08	75000	56			
2040 AF 20400			CABLE INS	C	87,800	9.0E+08				HEAT	WET PIPE
ELEVATION 117'-6"			LUBE OIL	E	1450	2.2E+08					SPRINKLERS
GENERAL FLOOR AREA			PLASTIC	D	6500	8.6E+07					HOSE STATION
			VOLATILES	E	180	2.4E+07					
			TRASH	B	1500	1.2E+07					
		42,606	NET CAT.	E	TOTAL:	1.2E+09	28,170	21			
2050 AF 20500			CABLE INS	C	87000	8.9E+08				HEAT	WET PIPE
ELEVATION 135'-3"			LUBE OIL	E	5400	8.2E+08					SPRINKLERS
GENERAL FLOOR AREA			PLASTIC	D	6000	7.9E+07					HOSE STATION
			VOLATILES	E	100	1.4E+07					
			HYDROGEN	E	50	7.6E+06					
			TRASH	B	50	3.9E+06					
		378900	NET CAT.	E	TOTAL:	1.8E+09	47510	36			
2050 AF 20502			CABLE INS	C	600	6.1E+06				SMOKE	WATER SPRAY
DEH SKID			LUBE OIL	E	250	3.8E+07					HOSE STATION
			PLASTIC	D	150	2.0E+06					
			TRASH	B	100	7.7E+05					
		149	NET CAT.	E	TOTAL:	4.7E+07	313000	235			
2052 AF 20504			CABLE INS	C	150	3.1E+06				SMOKE	HOSE STATION
HVAC EQUIPMENT			PAPER	C	250	3.9E+06					
AREA			PLASTIC	D	125	3.3E+06					
			RUBBER	D	13	3.1E+05					
			TRASH	B	13	2.0E+05					
		1231	NET CAT.	D	TOTAL:	1.1E+07	8700	7			

Change Number 34

Table 9.3.3-2 (Sheet 3 of 4)

LOCAL SAMPLE POINT NOT IN THE PRIMARY SAMPLING SYSTEM (NORMAL PLANT OPERATIONS)

Sample Point Name	Available Number of Points	Type of Sample[(a)]	Process Measurement
21. WLS chemical waste tank	1	Grab	Tritium, gross radioactivity and identification and concentration of principal radionuclide and alpha emitters
22. WSS spent resin tank (liquid)	1	Grab	Tritium, gross radioactivity and identification and concentration of principal radionuclide and alpha emitters
23. SWS blowdown (service water)	1 1	Continuous Grab	Radiation monitor (See Section 11.5, Table 11.5-1) Tritium, gross radioactivity and identification and concentration of principal radionuclide and alpha emitters
24. WWS turbine building sump	2	Grab	Tritium, gross radioactivity and identification and concentration of principal radionuclide and alpha emitters
25. CPS (secondary coolant) spent resin sluice line (liquid)	1	Grab	Tritium, gross radioactivity and identification and concentration of principal radionuclide and alpha emitters
Gaseous Sample			
26. VES MCR emergency air supply headers	2	Grab	Air quality, oxygen, carbon monoxide, carbon dioxide, contaminants
27. WGS effluent discharge to environment	1	Continuous	Radiation monitor (See Section 11.5, Table 11.5-1)
28. WGS inlet	1	Continuous	Oxygen, hydrogen~~, moisture~~
29. WGS carbon bed vault	1	Continuous	Hydrogen
30. WGS delay bed outlets MV02A, B (waste gas holdup)	2	Grab	Moisture, noble gases, iodine, particulates, tritium
31. Condenser air removal system[(b)] (including hogging)	1	Grab	Iodine, noble gases, tritium

Change Number 35

Table 9.3.4-1 (Sheet 1 of 2)

SECONDARY SAMPLING SYSTEM (CONTINUOUS MEASUREMENTS)

Continuous Sample Points	Process Measurements
Hotwell (Tube Bundle Condenser Shell A)	Specific Conductivity Cation Conductivity Sodium
Hotwell (Tube Bundle Condenser Shell B)	Specific Conductivity Cation Conductivity Sodium
Hotwell (Tube Bundle Condenser Shell C)	Specific Conductivity Cation Conductivity Sodium
Condensate Pump Discharge	Specific Conductivity Cation Conductivity Sodium pH Dissolved Oxygen
Deaerator Inlet (Condensate)	Specific Conductivity Cation Conductivity Sodium pH Oxygen Scavenger Residual Dissolved Oxygen
Feedwater	Specific Conductivity Cation Conductivity Sodium Dissolved Oxygen pH Oxygen Scavenger Residual
Steam Generator Blowdown (SG 1)	Specific Conductivity Cation Conductivity Sodium pH Sulfate Dissolved Oxygen

Table 9.3.4-1 (Sheet 2 of 2) **SECONDARY SAMPLING SYSTEM (CONTINUOUS MEASUREMENTS)**	
Steam Generator Blowdown (SG 2)	Specific Conductivity Cation Conductivity Sodium pH Sulfate Dissolved Oxygen
Main Steam System (SG 1)	~~Cation-~~Specific Conductivity Cation Conductivity Sodium pH Dissolved Oxygen
Main Steam System (SG 2)	~~Cation-~~Specific Conductivity Cation Conductivity Sodium pH Dissolved Oxygen

Change Number 36

1.8 Interfaces for Standard Design

This section identifies the AP1000 standard plant scope, interfaces related to design certification between the AP1000 plant design and the Combined License applicant, and the site-specific items to be included in an application for a Combined License. It is submitted to satisfy the requirements of 10 CFR 52.47(a)(1)(vii).

The AP1000 is a plant incorporating six buildings, the equipment in them, and the associated yard structures and tankage. This includes the nuclear island (consisting of the containment/shield building and the auxiliary building), the annex building and associated equipment, the diesel/generator building and associated equipment, the turbine generator building, the turbine/generator equipment, and the radwaste facilities. The physical boundary of the portion of the AP1000 included in this application is shown on the site plan, Figure 1.2-2. It includes arrangement and placement of structures within the indicated boundary. Additionally, the red zone delay barrier necessary for security is included, but the boundary fence and vehicle barrier are not included since they are site-specific. As a result, no interfaces need to be identified between or among the portions of the plant within the boundary. They are addressed in their appropriate section of this DCD. There are no safety-related interfaces to site-specific elements of the plant outside the scope of this certification application. The following site-specific elements are outside the scope of the AP1000 standard plant:

(1) The portions of the circulating water system and its heat sink outside the AP1000 buildings, as well as the specific design details of the main condenser. A conceptual design is presented, delineated by Double Brackets ([[ ]]), in subsection 10.4.5, based upon a cooling tower approach.

(2) The offsite power transmission system outside the low voltage terminals of the main and reserve transformers. Location and design of the main switchyard area and the equipment located therein, as well as design details such as voltage level for the main step-up transformers. A conceptual design of this system is included, delineated by Double Brackets ([[ ]]), in Section 8.2 for reference.

(3) Raw water source and treatment outside the turbine building. An interface specification of amount and water chemistry limits is provided.

(4) Sanitary and other drain systems outside the buildings identified above. This DCD is based upon the COL applicant providing adequate overall site drain collection and processing systems

(5) Communications systems and equipment outside the buildings identified above. This DCD is based upon the COL applicant providing adequate external communications.

(6) Location and design of administrative and training structures.

(7) Landscaping features.

(8) Size and location of the waste water retention basins and the associated plant outfall piping.

Table 1.8-1 (Sheet 5 of 6~~7~~)

SUMMARY OF AP1000 PLANT INTERFACES WITH REMAINDER OF PLANT

Item No.	Interface	Interface Type	Matching Interface Item	Section or Sub-section
9.5	Requirements for location and arrangement of raw and sanitary water systems	NNS	Site implementation	9
9.6	Ventilation requirements for diesel-generator room	NNS and Not an Interface	N/A	9
9.7	Requirements to satisfy fire protection program	AP1000 Interface	Combined License applicant program	9.5.1
9.8	Requirements for location and size of waste water retention basins and associated plant outfall	NNS	Site Implementation	9
11.1	Expected release rates of radioactive material from the Liquid Waste System including: Location of release points Effluent temperature Effluent flow rate Size and shape of flow orifices	Site Interface	Site specific parameters	11.2
11.2	Expected release rates of radioactive materials from the Gaseous Waste System including: Location of release points Height above grade Height relative to adjacent buildings Effluent temperature Effluent flow rate Effluent velocity Size and shape of flow orifices	Site Interface	Site specific parameters	11.3
11.3	Expected release rates of radioactive material from the Solid Waste System including: Location of release points Material types Material qualities Size and shape of material containers	Site Interface	Site specific parameters	11.4

- Water heaters provide a storage capacity equal to the probable hourly demand for potable hot water usage and provide hot water for the main lavatory, shower areas, and other locations where needed.

- A minimum pressure of 20 psig is maintained at the furthermost point in the distribution system.

- No interconnections exist between the potable water system and any potentially radioactive system or any system using water for purposes other than domestic water service.

9.2.5.2 System Description

9.2.5.2.1 General Description

Classification of components and equipment for the potable water system is given in Section 3.2.

The source of water for the potable water system is a site-specific water system. The potable water system consists of a distribution header around the power block, hot water storage heaters, and necessary interconnecting piping and valves. All other components of the potable water system outside the power block are site-specific and will be addressed in accordance with subsection 9.2.11.

9.2.5.2.2 Component Description

Hot Water Heaters

Electric immersion heating elements located inside the potable water hot water tank are used to produce hot water. This hot water is routed to the shower and toilet areas and to other plumbing fixtures and equipment requiring domestic hot water service. Point of use, inline electric water heating elements are used to generate hot water for the main control room and the turbine building secondary sampling laboratory.

9.2.5.3 System Operation

Filtered water is supplied from a site-specific water source for the potable water distribution system.

The onsite water supply system will maintain an appropriate pressure throughout the distribution system.

Potable water is supplied to areas that have the potential to be contaminated radioactively. Where this potential for contamination exists, the potable water system is protected by a reduced pressure zone type backflow prevention device.

No interconnections exist between the potable water system and any system using water for purposes other than domestic water service including any potentially radioactive system. ~~The common supply from the onsite raw water system is designed to use an air gap to prevent contamination of the potable water system from other systems supplied by the raw water system.~~

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Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.131, Rev. 0, 8/77 – Qualification Tests of Electric Cables, Field Splices and Connections for Light-Water-Cooled Nuclear Power Plants			
General	IEEE Std. 383-1974	Conforms	The insulating and jacketing material for electrical cables are selected to meet the fire and flame test requirements of IEEE Standard 1202 or IEEE Standard 383 excluding the option to use the alternate flame source, oil or burlap.
C.1-14		Conforms	
Reg. Guide 1.132, Rev. 1, 3/79 – Site Investigations for Foundations of Nuclear Power Plants			
General		N/A	Not applicable to AP1000 design certification. Section 2.5 defines the responsibility for site investigations and the site specific information related to basic geological, seismological, and geotechnical engineering of the site.
Reg. Guide 1.133, Rev. 1, 5/81 – Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors			
General		Conforms	A digital metal impact monitoring system (DMIMS) monitors the reactor coolant system for the presence of loose metallic parts. The system actuates audible and visual alarms if a signal exceeds the preset alarm level. The digital metal impact monitoring system is not a Class 1E system. It serves as a diagnostic aid to detect loose parts in the reactor coolant system before damage occurs. Database calibration is made prior to plant startup and the capability for periodic online channel checks and channel functional tests are incorporated in the digital metal impact monitoring system design.
C.1.a-i		Conforms	
C.2		Conforms	
C.3.a		N/A	Not applicable to AP1000 design certification. Section 13.5 defines the responsibility for development of procedures.
C.3.b		Conforms	
C.4~~-5~~		Conforms	

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
C.5		N/A	Not applicable to AP1000 design certification. Loose part detection system excluded from Technical Specification requirements with WCAP-11618 (page 79, Loose Parts Monitoring discussion) per NRC-established screening selection criteria, as coded in 10 CFR 50.36(c)(2)(ii).
C.6		N/A	Not applicable to AP1000 design certification. ~~Reporting Requirements associated with the technical specification are identified in Tech. Spec. Section 5.6. DCD subsection 1.1.1 defines the responsibility to finalize the technical specification.~~ Section 13.5 defines the responsibility for plant procedures.

Reg. Guide 1.134, Rev. 3, 3/98 – Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses

General		N/A	Not applicable to AP1000 plant design certification. DCD Section 13.5 defines the responsibility for administrative procedures.

Reg. Guide 1.135, Rev. 0, 9/77 – Normal Water Level and Discharge at Nuclear Power Plants

General		Conforms	The normal ground and surface water levels and surface water discharges for the AP1000 are determined using the postulated site parameters. Chapter 2 provides additional information.

Reg. Guide 1.136, Rev. 2, 6/81 – Materials, Construction, and Testing of Concrete Containments

General		N/A	The AP1000 does not have a concrete containment. Therefore, this guideline is not applicable to the AP1000.

Reg. Guide 1.137, Rev. 1, 10/79 – Fuel-Oil Systems for Standby Diesel Generators

General		N/A	The AP1000 diesel-generators and the associated fuel-oil systems are non-safety-related. Therefore, this guideline is not applicable to the AP1000.

Reg. Guide 1.138, Rev. 0, 4/78 – Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants

General		N/A	Not applicable to AP1000 design certification. Subsection 2.5.4.6.2 defines the responsibility to establish the properties of the foundation soils including laboratory investigations of underlying materials.

Change Number 39

- Maintenance plan – This plan describes the nature and frequency of maintenance activities to be performed on plant equipment. The plan includes the selected SSCs identified in the D-RAP.

17.5.8 The Combined License applicant is responsible for integrating the objectives of the OPRAAs into the Quality Assurance Program developed to implement 10 CFR 50, Appendix B. This program will address failures of non-safety-related, risk-significant SSCs that result from design and operational errors in accordance with SECY-95-132, Item E.

17.6 References

1. "Energy Systems Business Unit – Quality Management System," Revision 2.

2. WCAP-8370, Revision 12a, "Energy Systems Business Unit - Power Generation Business Unit Quality Assurance Plan."

3. WCAP-8370/7800, Revision 11A/7A, "Energy Systems Business Unit - Nuclear Fuel Business Unit Quality Assurance Plan."

4. WCAP-12600, Revision 4, "AP600 Advanced Light Water Reactor Design Quality Assurance Program Plan," January 1998.

5. APP-GW-GL-022, Revision 8~~0~~, AP1000 Probabilistic Risk Assessment.

6. Not used.

7. NRC/DCP0669, "Criteria for Establishing Risk Significant Structures, Systems, and Components (SSCs) to be Considered for the AP600 Reliability Assurance Program," January 16, 1997.

8. Lofgren, E. V., Cooper, et al., "A Process for Risk-Focused Maintenance," NUREG/CR-5695, March 1991.

9. Westinghouse Electric Company Quality Management System (QMS), Revision 5, dated October 1, 2002.

10. NEI 07-02, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52."

11. APP-GW-GLR-117, "Incorporation of the Maintenance Rule," Westinghouse Electric Company LLC.

12. SECY 95-132, "Policy and Technical Issue With the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY 94-084)."

Change Number 40

Table 3.3.3-1 (page 1 of 1)
Post-Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS/ DIVISIONS	CONDITION REFERENCED FROM REQUIRED ACTION D.1
1.	Neutron Flux (Intermediate Range)	2	E
2.	Reactor Coolant System (RCS) Hot Leg Temperature (Wide Range)	2	E
3.	RCS Cold Leg Temperature (Wide Range)	2	E
4.	RCS Pressure (Wide Range)	2	E
5.	Pressurizer Pressure and RCS Subcooling Monitor[a]	2	E
6.	Containment Water Level	2	E
7.	Containment Pressure	2	E
8.	Containment Pressure (Extended Range)	2	E
9.	Containment Area Radiation (High Range)	2	E
10.	Pressurizer Level and Associated Reference Leg Temperature	2	E
11.	IRWST Water Level	2	E
12.	PRHR Flow and PRHR Outlet Temperature	2 flow & 1 temperature	E
13.	Core Exit Temperature -- Quadrant 1	2[b]	E
14.	Core Exit Temperature -- Quadrant 2	2[b]	E
15.	Core Exit Temperature -- Quadrant 3	2[b]	E
16.	Core Exit Temperature -- Quadrant 4	2[b]	E
17.	PCS Storage Tank Level and PCS Flow	2 level & 1 flow	E
18.	Remotely Operated Containment Isolation Valve Position	1/valve[c]	E
19.	IRWST to RNS Suction Valve Status	1/valve[c] ~~2~~	E

(a) RCS Subcooling calculated from pressurizer pressure and RCS hot leg temperature.

(b) A channel consists of two thermocouples within a single division. Each quadrant contains two divisions. The minimum requirement is two OPERABLE thermocouples in each of the two divisions.

(c) Not required for isolation valves whose associated penetration or flow path is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

BASES

LCO (continued)

An evaluation was made of the minimum number of valid core exit thermocouples necessary for In-Core Cooling (ICC) detection. The evaluation determined the reduced complement of core exit thermocouples necessary to detect initial core recovery and trend the ensuing core heatup. The evaluations account for core nonuniformities including incore effects of the radial decay power distribution and excore effects of condensate runback in the hot legs and nonuniform inlet temperatures. Based on these evaluations, adequate ICC detection is assured with two valid core exit thermocouples per quadrant. Core Exit Temperature is also used for plant stabilization and cooldown monitoring.

Two OPERABLE channels of Core Exit Temperature are required in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Two thermocouples in each of the two divisions ensure a single failure will not disable the ability to determine the temperature at two locations within a quadrant.

17. Passive Containment Cooling System (PCS) Storage Tank Level and PCS Flow

The PCS must be capable of removing the heat from the containment following a postulated LOCA or steam line break (SLB). The tank level instruments provide indication that sufficient water is available to meet this requirement. The PCS flow instrument provides a diverse indication of the PCS heat removal capability.

18. Remotely Operated Containment Isolation Valve Position

The Remotely Operated Containment Isolation Valve Position is provided for verification of containment OPERABILITY.

19. IRWST to RNS Suction Valve Status

The position of the two motor-operated valves (RNS-V022 and RNS-V023) in the line from the IRWST to the RNS pump suction header is monitored to verify that the flow path is isolated following postulated events. The flow path must be isolated to prevent loss of IRWST inventory into the RNS.

Change Number 41

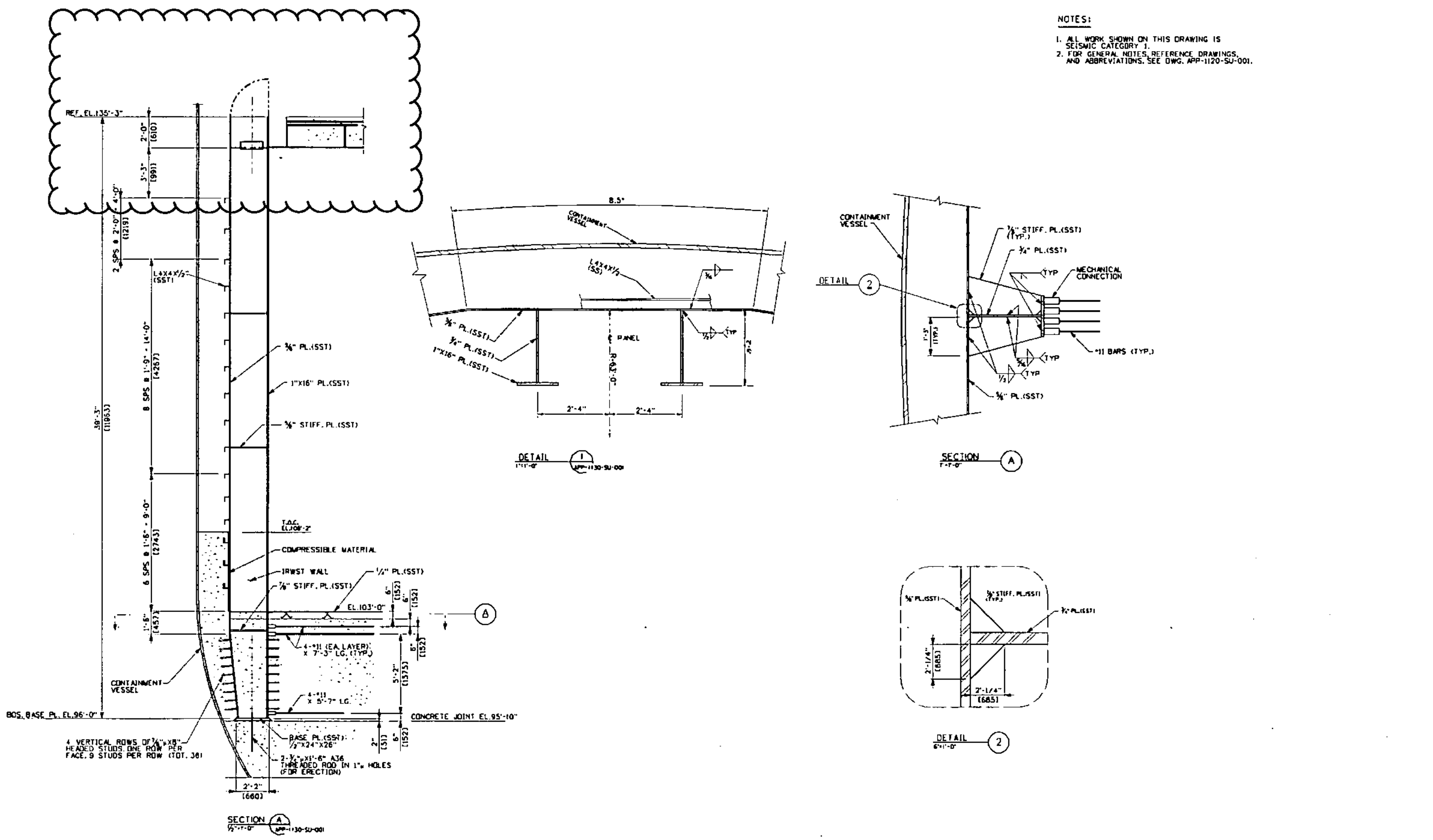

Figure 3.8.3-8 (Sheet 3 of 3)

*[Structural Modules – Typical Design Details]**

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

Change Number 42

removal system design pressure is 900 psig. Therefore the ultimate rupture strength of the system prevents it from failing when exposed to the normal reactor coolant system operating pressure (2250 psia~~g~~). See the position on intersystem LOCA for additional information on the normal residual heat removal system design against overpressurization.

Steam generator tube rupture core melt frequency is reduced by incorporating multiple levels of defense that are both redundant and diverse. The first level of defense relies on the use of nonsafety-related active systems and operator action. The second level of defense uses safety-related passive systems and equipment, such as the core makeup tanks and passive residual heat removal heat exchangers, without the safety-related automatic depressurization of the reactor coolant system. The third level of defense uses the redundant and diverse safety-related automatic depressurization system valves to depressurize the reactor coolant system and initiate low-pressure passive injection. Any of these levels of defense can prevent core damage during a steam generator tube rupture event.

Finally, containment isolation capabilities are substantially improved by reducing the number of penetrations and the number of open paths. Most of the open containment penetration lines use fail-closed valves for automatic isolation.

1.9.5.1.12 ABWR Containment Vent Design

This issue is specific to BWRs and PWRs with ice condenser containments. Therefore this issue does not apply to the AP1000 design.

1.9.5.1.13 Equipment Survivability

NRC Position:

Safety-related equipment used to mitigate design basis events is subject to a comprehensive set of criteria such as redundancy, diversity, environmental qualification, and quality assurance to provide reasonable assurance that they perform their intended functions, if needed. However, equipment used to mitigate the effects of severe accidents should not be treated in the same manner because of large differences in the likelihood of occurrence. There should be reasonable assurance that the equipment will operate in the severe accident environment for which they are intended and over the time span for which they are needed. However, equipment provided only for severe accident protection need not be subject to the 10 CFR 50.49, environmental qualification requirements, 10 CFR 50, Appendix B quality assurance requirements, and 10 CFR 50 Appendix A, redundancy and diversity requirements.

AP1000 Response:

The equipment used to mitigate severe accidents is identified in the AP1000 PRA evaluation report. Because of the nature of the passive safety features of the AP1000, there is very little equipment in this category. Equipment used to mitigate severe accidents is designed to survive the environmental conditions identified in the AP1000 PRA evaluation.

Change Number 43

1.7 Drawings and Other Detailed Information

The figures referenced in subsections 1.7.1 and 1.7.2 may represent a functional diagram, general structural representation, or another general illustration. For instrumentation and control (I&C) systems, figures may also represent aspects of the relevant logic of the system or part of the system. Unless specified explicitly, the figures are not indicative of the scale, location, dimensions, shape, or spatial relationships of as-built structures, systems, and components. In particular, the as-built attributes of structures, systems, and components may vary from the attributes depicted on the figures, provided that those safety functions discussed in the design description pertaining to the figure are not adversely affected.

1.7.1 Electrical and Instrumentation and Control Drawings

Instrument and control functional diagrams, electrical one-line diagrams, and onsite standby diesel generator loading sequence and initiating circuit logic diagrams are listed in Table 1.7-1.

The legend for electrical power, control, lighting, and communication drawings is provided in Figure 1.7-1, sheets 1, 2, and 3. The index, notes, and symbols for instrument and control functional diagrams are provided in Figure 7.2-1.

1.7.2 Piping and Instrumentation Diagrams

Table 1.7-2 contains a list of piping and instrumentation diagrams (P&IDs) and the corresponding DCD figure numbers. The three letter system names are provided in Table 1.7-2. Figures appear at the end of the respective text section. The P&ID legend, Figure 1.7-2, sheets 1, 2, and 3, provides an explanation of AP1000 symbols and characters used in these DCD figures.

1.7.3 Combined License Information

This section has no requirement for additional information to be provided in support of the combined license application.

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Table 5.2-1 (Sheet 2 of 6~~5~~)

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS SPECIFICATIONS

Component	Material	Class, Grade, or Type
Vent pipe	SB-166 SB-167 or SA-312[1] SA-376	N06690 N06690 TP304, TP304L, TP304LN, TP316, TP316L, TP316LN TP304, TP304LN, TP316, TP316LN
Steam Generator Components		
Pressure plates	SA-533	Type B, CL 1 or CL 2
Pressure forgings~~-~~(including primary side nozzles and tube sheet)	SA-508	~~CL 1A or~~ GR 3, CL 2
Nozzle safe ends	SA-182 SA-336 or SB-564	F316, F316L, F316LN F316LN N06690
Channel heads	SA-508	GR 3, CL 2
Tubes	SB-163	N06690
Manway studs/ Nuts	SA-193 SA-194	GR B7 GR 7
Pressurizer Components		
Pressure plates	SA-533	Type B, CL 1
Pressure forgings	SA-508	GR 3, CL 2
Nozzle safe ends	SA-182 SA-336~~338~~ or SB-564~~163~~	F316, F316L, F316LN F316, F316L, F316LN N06690
Manway studs/ Nuts	~~SA-193~~ SA-194	GR B7 GR 7

Change Number 46

The springs in the control rod drive mechanism are made from nickel-chromium-iron alloy (Alloy 750), ordered to Aerospace Material Specification (AMS) 5698 or AMS 5699 with additional restrictions on prohibited materials. Operating experience has shown that springs made of this material are not subject to stress-corrosion cracking in pressurized water reactor primary water environments. Alloy 750 is not used for bolting applications in the control rod drive mechanisms.

4.5.1.4 Contamination Protection and Cleaning of Austenitic Stainless Steel

The control rod drive mechanisms are cleaned prior to delivery in accordance with the guidance provided in NQA-1 (see Chapter 17). Process specifications in packaging and shipment are discussed in subsection 5.2.3. Westinghouse personnel conduct surveillance of these operations to verify that manufacturers and installers adhere to appropriate requirements as described in subsection 5.2.3.

Tools used in abrasive work operations on austenitic stainless steel, such as grinding or wire brushing, do not contain and are not contaminated with ferritic carbon steel or other materials that could contribute to intergranular cracking or stress-corrosion cracking.

4.5.2 Reactor Internal and Core Support Materials

4.5.2.1 Materials Specifications

The major core support material for the reactor internals is SA-182, SA-336, SA-376, SA-479, or SA-240 Types 304, 304L, 304LN, or 304H stainless steels. Fabricators performing welding of any of these materials are required to qualify the welding procedures for maximum carbon content and heat input for each welding process in accordance with Regulatory Guide 1.44. For threaded structural fasteners, the material used is strain hardened Type 316 stainless steel, and for the clevis insert-to-vessel bolts, either UNS N07718 or N07750. Remaining ~~internals~~ parts not fabricated from Types 304, 304L, 304LN, or 304H stainless steels typically include wear surfaces such as hardfacing on the radial keys, clevis inserts, and alignment ~~pins~~ plates (Stellite™ 6 or 156 or low cobalt hardfaces); dowel pins (Type 316); ~~hold down spring (Type 403 stainless steel (modified));~~ and clevis inserts (UNS N06690)~~; and irradiation specimen springs (UNS N07750)~~.

Additional product forms that may be used by the reactor internals structures are SA-193, SA-213, SA-312, SA-358, SB-166, SB-168, and SB-637. Reactor internals structures~~Instrument guide assembly~~ materials that are not Types 304, 304L, 304LN, or 304H stainless steel are the guide bushings, locating and support pins, instrumentation adapter, instrument tube tip, instrument tube sleeve top,~~and~~ guide stud ~~tip~~ (all UNS S21800), Swagelok fitting (Type 316), hold down spring (Type 403 stainless steel [modified]), irradiation specimen access plug springs and instrument stalk springs (UNS N07750), ~~and the~~ instrument ~~guide~~ tube sleeve spring (UNS N07718), and flow skirt (UNS N06690). Additionally, reactor internals structures will use either threaded structural fasteners of strain hardened Type 316 stainless steel, UNS N07718 or N07750.

Core support structure, internals structures, and threaded structural fastener materials are specified in the ASME Code, Section II~~I~~, Part D, Subpart 1, Tables 2A, 2B, and 4~~Appendix I~~ as supplemented by Code Cases N-60 and N-4. The qualification of cobalt free wear resistant alloys for use in reactor coolant is addressed in subsection 4.5.1.3.

Change Number 47

performing periodic operability testing of the passive residual heat removal heat exchanger discharge valves when the reactor is operating. Once the manual operation is complete, the control circuit returns to automatic operation.

- The control circuit has a valve maintain closed actuation function to provide an administratively controlled manual block of the automatic opening of the valve. This function allows the valve to be maintained closed if needed for leakage isolation. The maximum permissible time that a passive residual heat removal heat exchanger inlet isolation valve can be closed is specified in technical specifications. An alarm is actuated when the maintain closed function is instated.

The valve is interlocked so that:

- If the maintain closed actuation has not been manually initiated, it opens automatically on receipt of a confirmatory open signal with the control circuit in automatic control or during the manual valve close function.

- It cannot be manually closed when a confirmatory open signal is present.

During plant operation and shutdown, the passive residual heat removal heat exchanger inlet isolation valve is open. ~~To prevent an inadvertent closure of the valve, redundant output cards are used in the protection and safety monitoring system cabinet.~~ Power to this valve is normally locked out at power to prevent a fire-induced spurious closing.

Figure 7.2-1, sheet 17 illustrates the interlock logic which applies to the passive residual heat removal heat exchanger inlet isolation valve.

This normally open motor-operated valve has alarms, indicating valve mispositioning (with regard to their passive core cooling function). The alarm actuates in the main control room and the remote shutdown workstation.

An alarm actuates for the passive residual heat removal heat exchanger inlet isolation valve under the following conditions when the passive residual heat removal heat exchanger is required:

- Sensors on the motor operator for the valve indicate when the valve is not fully open.

- Redundant sensors on the valve stem indicate when the valve is not fully open.

7.6.2.2 Core Makeup Tank Cold Leg Balance Line Isolation Valves

Each core makeup tank has a cold leg balance line which is provided with a normally open, motor-operated, isolation valve. The balance line isolation valves, for each core makeup tank, may be manually controlled from either the main control room or the remote shutdown workstation. The generation of the confirmatory open signal to these valves is described in subsection 7.3.1.2.3.

A confirmatory open signal to these valves automatically overrides any bypass features that are provided to allow the balance line isolation valve to be closed for short periods of time. As a result

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Table 3.2-3 (Sheet 21 of ~~65~~69)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V117B	Containment Recirculation B Isolation	C	I	ASME III-3	
PXS-PL-V118A	Containment Recirculation A Isolation	C	I	ASME III-3	
PXS-PL-V118B	Containment Recirculation B Isolation	C	I	ASME III-3	
PXS-PL-V119A	Containment Recirculation A Check	C	I	ASME III-3	
PXS-PL-V119B	Containment Recirculation B Check	C	I	ASME III-3	
PXS-PL-V120A	Containment Recirculation A Isolation	C	I	ASME III-3	
PXS-PL-V120B	Containment Recirculation B Isolation	C	I	ASME III-3	
PXS-PL-V121A	IRWST Line A Isolation	C	I	ASME III-3	
PXS-PL-V121B	IRWST Line B Isolation	C	I	ASME III-3	
PXS-PL-V122A	IRWST Injection A Check	A	I	ASME III-1	
PXS-PL-V122B	IRWST Injection B Check	A	I	ASME III-1	
PXS-PL-V123A	IRWST Injection A Isolation	A	I	ASME III-1	
PXS-PL-V123B	IRWST Injection B Isolation	A	I	ASME III-1	
PXS-PL-V124A	IRWST Injection A Check	A	I	ASME III-1	
PXS-PL-V124B	IRWST Injection B Check	A	I	ASME III-1	
PXS-PL-V125A	IRWST Injection A Isolation	A	I	ASME III-1	
PXS-PL-V125B	IRWST Injection B Isolation	A	I	ASME III-1	
PXS-PL-V126A	IRWST Injection Check Test	C	I	ASME III-3	
PXS-PL-V126B	IRWST Injection Check Test	C	I	ASME III-3	
PXS[rmk38]-PL-V127A	IRWST Injection Line A Drain	C	I	ASME III-3	
PXS-PL-V128A	IRWST Injection Check Test	~~B~~ A	I	ASME III-~~2~~1	
PXS-PL-V128B	IRWST Injection Check Test	~~B~~ A	I	ASME III-~~2~~1	
PXS-PL-V129A	IRWST Injection Check Test	~~B~~ A	I	ASME III-~~2~~1	
PXS-PL-V129B	IRWST Injection Check Test	~~B~~ A	I	ASME III-~~2~~1	
PXS-PL-V130A	IRWST Gutter Bypass A Isolation	C	I	ASME III-3	

Change Number 49

17.4.7.2 Not Used

17.4.7.2.1 Not Used

17.4.7.3 Not Used

17.4.7.4 D-RAP Implementation

The following is an example of a system that was reviewed and modified under the D-RAP. The design and analytical results presented here are intended as an example.

The automatic depressurization system, which is part of the reactor coolant system, acts in conjunction with the passive core cooling system to mitigate design basis accidents. The automatic depressurization system valves are discussed in subsection 5.4.6 of the DCD.

An earlier AP600 automatic depressurization system design contained four depressurization stages, with motor-operated valves in all stages. Preliminary PRA analysis established that fourth stage failure, in certain combination with failures of other stages, was a major contributor to core damage frequency. Thus, it was concluded that the fourth stage valves should be diverse in design from the valves in other stages to reduce common cause failure.

As a result of joint meetings among the AP600 PRA, Design, and staff organizations to discuss core melt frequency improvements, the fourth stage automatic depressurization system was changed from a motor-operated valve to a squib (explosively actuated) valve. The new configuration of the system is shown in the reactor coolant system P&ID (Figure 5.1-5 of the DCD). An example of the analytical results that reflect this change is provided in Table 17.4-2. This design feature is included in the AP1000 design to maintain the core melt frequency improvements included in the AP600 design.

As part of the evaluation of the squib valves, a failure modes and effects analysis (FMEA) was prepared to identify subcomponent failures and critical items that could lead to hazardous or abnormal conditions of the automatic depressurization system and the plant. The identification of failure modes facilitated the development of recommended maintenance and in-service testing activities to maximize valve reliability.

The squib valve is a completely static electromechanical assembly. Prior to activation, there are no moving parts. No powered components are needed to hold a stem seat or globe in place by torque, solenoid coils, or friction. The explosive actuator is a simple, passive device that is triggered by an applied voltage.

Because the automatic depressurization system fourth stage valves perform safety-related functions, they will be subject to in-service testing to verify that they are ready to function in an accident. Subsection 3.9.6 of the DCD includes in-service testing requirements for these valves.

Example FMEA results for the fourth stage squib valves and the second and third stage motor-operated valves are included in DCD Table 6.3-3. Table 3.9-16~~DCD subsection 3.9.6.3.1~~ provides testing recommendations for the second and third stage valves.

Change Number 50

Table 3.9-16 (Sheet 10 of 23)

VALVE INSERVICE TEST REQUIREMENTS

Valve Tag Number	Description[1]	Valve/Actuator Type	Safety-Related Missions	Safety Functions[2]	ASME Class/ IST Category	Inservice Testing Type and Frequency	IST Notes
PXS-PL-V043	Nitrogen Supply Containment Isolation IRC	Check	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	Class 2 Category AC	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test Check Exercise/Quarterly	27
PXS-PL-V101	PRHR HX Inlet Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years	
PXS-PL-V108A	PRHR HX Control	Remote AO GLOBE	Maintain Open Transfer Open	Active-to-Failed Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test	31
PXS-PL-V108B	PRHR HX Control	Remote AO GLOBE	Maintain Open Transfer Open	Active-to-Failed Remote Position	Class 1 Category B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test	31
PXS-PL-V117A	Containment Recirculation A Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 3 Category B	Remote Position Indication, Exercise/2 Years	31
PXS-PL-V117B	Containment Recirculation B Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 3 Category B	Remote Position Indication, Exercise/2 Years	
PXS-PL-V118A	Containment Recirculation A Isolation	Squib	Maintain Open Maintain Close Transfer Open	Active Remote Position	Class 3 Category D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5
PXS-PL-V118B	Containment Recirculation B Isolation	Squib	Maintain Open Maintain Close Transfer Open	Active Remote Position	Class 3 Category D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5
PXS-PL-V119A	Containment Recirculation A Check	Check	Maintain Open Maintain Close Transfer Open Transfer Close	Active Remote Position	Class 3 Category BC	Remote Position Indication, Exercise/2 Years Check-Initial Open Differential Pressure/Refueling~~2 Years~~ Check Exercise/Refueling Shutdown	11
PXS-PL-V119B	Containment Recirculation B Check	Check	Maintain Open Maintain Close Transfer Open Transfer Close	Active Remote Position	Class 3 Category BC	Remote Position Indication, Exercise/2 Years Check-Initial Open Differential Pressure/Refueling~~2 Years~~ Check Exercise/Refueling Shutdown	11
PXS-PL-V120A	Containment Recirculation A Isolation	Squib	Maintain Open Maintain Close Transfer Open	Active Remote Position	Class 3 Category D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5
PXS-PL-V120B	Containment Recirculation B Isolation	Squib	Maintain Open Maintain Close Transfer Open	Active Remote Position	Class 3 Category D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5

Table 3.9-16 (Sheet 11 of 23)

VALVE INSERVICE TEST REQUIREMENTS

Valve Tag Number	Description[1]	Valve/Actuator Type	Safety-Related Missions	Safety Functions[2]	ASME Class/ IST Category	Inservice Testing Type and Frequency	IST Notes
PXS-PL-V121A	IRWST Line A Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 3 Category B	Remote Position Indication, Exercise/2 Years	
PXS-PL-V121B	IRWST Line B Isolation	Remote MO GATE	Maintain Open	Remote Position	Class 3 Category B	Remote Position Indication, Exercise/2 Years	
PXS-PL-V122A	IRWST Injection A Check	Check	Maintain Open Maintain Close Transfer Open Transfer Close	Active RCS Pressure Boundary Remote Position	Class 1 Category BC	Remote Position Indication, Exercise/2 Years Check-Initial Open Differential Pressure/Refueling~~2 Years~~ Check Exercise/Refueling Shutdown	12
PXS-PL-V122B	IRWST Injection B Check	Check	Maintain Open Maintain Close Transfer Open Transfer Close	Active RCS Pressure Boundary Remote Position	Class 1 Category BC	Remote Position Indication, Exercise/2 Years Check-Initial Open Differential Pressure/Refueling~~2 Years~~ Check Exercise/Refueling Shutdown	12
PXS-PL-V123A	IRWST Injection A Isolation	Squib	Maintain Open Maintain Close Transfer Open	Active RCS Pressure Boundary Remote Position	Class 1 Category D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5
PXS-PL-V123B	IRWST Injection B Isolation	Squib	Maintain Open Maintain Close Transfer Open	Active RCS Pressure Boundary Remote Position	Class 1 Category D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5
PXS-PL-V124A	IRWST Injection A Check	Check	Maintain Open Maintain Close Transfer Open Transfer Close	Active RCS Pressure Boundary Remote Position	Class 1 Category BC	Remote Position Indication, Exercise/2 Years Check-Initial Open Differential Pressure/Refueling~~2 Years~~ Check Exercise/Refueling Shutdown	12
PXS-PL-V124B	IRWST Injection B Check	Check	Maintain Open Maintain Close Transfer Open Transfer Close	Active RCS Pressure Boundary Remote Position	Class 1 Category BC	Remote Position Indication, Exercise/2 Years Check-Initial Open Differential Pressure/Refueling~~2 Years~~ Check Exercise/Refueling Shutdown	12
PXS-PL-V125A	IRWST Injection A Isolation	Squib	Maintain Open Maintain Close Transfer Open	Active RCS Pressure Boundary Remote Position	Class 1 Category D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5
PXS-PL-V125B	IRWST Injection B Isolation	Squib	Maintain Open Maintain Close Transfer Open	Active RCS Pressure Boundary Remote Position	Class 1 Category D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5
PXS-PL-V130A	IRWST Gutter Isolation	Remote AO GLOBE	Maintain Close Transfer Close	Active-to-Failed Remote Position	Class 3 Category B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test	31

Change Number 51

- Limits pressure differentials on internal steam generator components, particularly the tube support plates

The restrictor is configured to minimize the unrecovered pressure loss across the restrictor during normal operation.

5.4.4.2 Design Description

The flow restrictor consists of seven nickel-chromium-iron Alloy 690 (ASME SB-564) venturi inserts which are installed in holes in an integral steam outlet nozzle forging. The inserts are arranged with one venturi at the centerline of the outlet nozzle, and the other six are equally spaced around it. After insertion into the nozzle forging holes, the venturi inserts are welded to the nickel-chromium-iron alloy buttering ~~cladding~~ on the inner surface of the forging.

5.4.4.3 Design Evaluation

The flow restrictor design has been analyzed to determine its structural adequacy. The equivalent throat area of the steam generator outlet is 1.4 square feet. The resultant pressure drop through the restrictor at 100 percent steam flow is approximately 15 psi. This is based on a design flow rate of 7.49×10^6 pounds per hour. Materials of construction of the flow restrictor are in accordance with Code Class 1 Section III of the ASME Code. The material of the inserts is not an ASME Code pressure boundary, nor is it welded to an ASME Code pressure boundary. The method for seismic analysis is dynamic.

5.4.4.4 Inspections

Since the restrictor is not part of the steam system pressure boundary, inservice inspections are not required.

5.4.5 Pressurizer

The pressurizer provides a point in the reactor coolant system where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control of the reactor coolant system during steady-state operations and transients. The pressurizer provides a controlled volume from which level can be measured.

The pressurizer contains the water inventory used to maintain reactor coolant system volume in the event of a minor system leak for a reasonable period without replenishment. The pressurizer surge line connects the pressurizer to one reactor coolant hot leg. This allows continuous coolant volume and pressure adjustments between the reactor coolant system and the pressurizer.

5.4.5.1 Design Bases

The pressurizer is sized to meet following requirements:

- The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.

Change Number 52

B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 pH Adjustment

BASES

BACKGROUND

The Passive Core Cooling System (PXS) includes four~~two~~ pH adjustment baskets which provide adjustment of the pH of the water in the containment following an accident where the containment floods.

Following an accident with a large release of radioactivity, the containment pH is automatically adjusted to greater than or equal to 7.0, to enhance iodine retention in the containment water. Chemical addition is necessary to counter the affects of the boric acid contained in the safety injection supplies and acids produced in the post-LOCA environment (nitric acid from the irradiation of water and air and hydrochloric acid from irradiation and pyrolysis of electric cable insulation). The desired pH values significantly reduce formation of elemental iodine in the containment water, which reduces the production of organic iodine and the total airborne iodine in the containment. This pH adjustment is also provided to prevent stress corrosion cracking of safety related containment components during long-term cooling.

Dodecahydrate trisodium phosphate (TSP) contained in baskets provides a passive means of pH control for such accidents. The baskets are made of stainless steel with a mesh front that readily permits contact with water. These baskets are located inside containment at an elevation that is below the minimum floodup level. The baskets are placed at least a foot above the floor to reduce the chance that water spills will dissolve the TSP. Natural recirculation of water inside the containment, following a LOCA, is driven by the core decay heat and provides mixing to achieve a uniform pH. The dodecahydrate form of TSP ($Na_3PO_4 \cdot 12H_2O$) is initially loaded into the baskets because it is hydrated and will undergo less physical and chemical change than would anhydrous TSP as a result of the humidity inside containment. (Refs. 1 and 2)

APPLICABLE SAFETY ANALYSES

In the event of a Design Basis Accident (DBA), iodine may be released from the fuel to containment. To limit this iodine release from containment, the pH of the water in the containment sump is adjusted by the addition of TSP. Adjusting the sump water to neutral or alkaline pH (pH ≥ 7.0) will augment the retention of the iodine, and thus reduce the iodine available to leak to the environment.

pH adjustment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

BACKGROUND (continued)

The accumulator size, water volume, and nitrogen cover pressure are selected so that both of the accumulators are sufficient to recover the core cooling before significant clad melting or zirconium water reaction can occur following a large break LOCA. One accumulator is adequate during a small break LOCA where the entire contents of one accumulator can possibly be lost via the pipe break. This accumulator performance is based on design basis accident (DBA) assumptions and models (Ref. 3). The Probabilistic Risk Assessment (PRA) (Ref. 4) shows that one of the two accumulators is sufficient for a large break LOCA caused by spurious ADS actuation and that none of the accumulators are required for small break LOCAs, assuming that at least one core makeup tank (CMT) is available. In addition, both accumulators are required for a large break LOCA caused by the break of a cold leg pipe; the probability of this break has been significantly reduced by incorporation of leak-before-break.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed to be OPERABLE in both the large and small break LOCA analyses at full power (Ref. 3) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

For a small break LOCA, a large range of break sizes and locations were analyzed to verify the adequacy of the design. The cases analyzed include the rupture of one 8 inch direct vessel injection line and several smaller break sizes. Acceptable PXS performance was demonstrated.

For a larger LOCA, including a double ended RCS piping rupture, the PXS can provide a sufficiently large flow rate, assuming both accumulators are OPERABLE, to quickly fill the reactor vessel lower plenum and downcomer. Both accumulators, in conjunction with the CMTs, ensure rapid reflooding of the core. For a large LOCA, both lines are available since an 8 inch line break would be a small LOCA.

Following a non-LOCA event such as a steam line break, the RCS experiences a decrease in temperature and pressure due to an increase in energy removal by the secondary system. The cooldown results in a reduction of the core SHUTDOWN MARGIN with a potential for return to power. During such an event the accumulators provide injection of borated water to assist the CMT's boration to mitigate the reactivity transient and ensure the core remains shut down.

The accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES

The CMTs are assumed to be OPERABLE to provide emergency boration and core makeup when the Chemical and Volume Control System (CVS) is inoperable, and to mitigate the consequences of any DBA which requires the safety injection of borated water (Ref. 2).

Following a non-LOCA event such as a steam line break, the RCS experiences a decrease in temperature and pressure due to an increase in energy removal by the secondary system. The cooldown results in a reduction of the core SHUTDOWN MARGIN due to the negative moderator temperature coefficient, with a potential for return to power. The actuation of the CMTs following this event provides injection of borated water to mitigate the reactivity transient and ensure the core remains shut down.

In the case of a steam generator tube rupture (SGTR), CMT injection provides borated water to compensate for RCS LEAKAGE.

In the case of an RCS leak of 10 gallons per minute, the CMTs can delay depressurization for at least 10 hours, providing makeup to the RCS and remain able to provide the borated water to compensate for RCS shrinkage and to assure the RCS boration for a safe shutdown.

In the case of a LOCA, the CMTs provide a relatively large makeup flow rate for approximately 20 minutes, in conjunction with the accumulators to provide the initial core cooling.

CMTs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO establishes the minimum conditions necessary to ensure that sufficient CMT flow will be available to meet the initial conditions assumed in the safety analyses. The volume of each CMT represents 100% of the total injected flow assumed in LOCA analysis. If the injection line from a single CMT to the vessel breaks, no single active failure on the other CMT will prevent the injection of borated water into the vessel. Thus the assumptions of the LOCA analysis will be satisfied.

For non-LOCA analysis, two CMTs are assumed. Note that for non-LOCA analysis, the accident cannot disable a CMT.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND	Each main steam line has one safety related MSIV to isolate steam flow from the secondary side of the steam generators following a high energy line break. MSIV closure terminates flow from the unaffected (intact) steam generator. One MSIV is located in each main steam line outside containment. The MSIVs are downstream from the main steam safety valves (MSSVs). Downstream from the MSIVs, main steam enters the high pressure turbine through four stop valves and four governing control valves. Closing the MSIVs isolates each steam generator from the other and isolates the turbine bypass system, and other steam supplies from the steam generator. The MSIVs, turbine stop and control valves, turbine bypass valves, and moisture separator reheater 2nd stage steam isolation valves close on a main steam isolation signal generated by either low steam line pressure, high containment pressure, Low T_{cold}, or high negative steam pressure rate. The MSIVs fail closed on loss of control air or actuation signal from either of two 1E power divisions. Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually. A description of the MSIVs is found in the Section 10.3 (Ref. 1). Descriptions for the turbine bypass valves, and moisture separator reheater 2nd stage steam isolation valves are found in the Section 10.4 (Ref. 6).
APPLICABLE SAFETY ANALYSES	The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the Section 6.2 (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the Section 15.1 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand). Design basis events of concern for containment analysis are SLB inside containment with the failure of the associated MSIV to close, or a main feedline break with the associated failure of a feedline isolation or control

BASES

APPLICABLE SAFETY ANALYSES (continued)

MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIV in the unaffected loop. Closure of the MSIV isolates the break from the unaffected steam generator.

b. A break outside of containment, and upstream or downstream from the MSIVs, is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs or alternate downstream valves isolates the break, and limits the blowdown to a single steam generator.

c. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator to minimize radiological releases.

d. The MSIVs are also utilized during other events such as a feedwater line break; however, these events are less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs and the alternate downstream valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Following an SLB and main steam isolation signal, the analyses assume continued steam loss through the steam line condensate drain lines, turbine gland seal system, and the main steam to auxiliary steam header which supplies the auxiliary steam line to the deaerator. Since these valves are not assumed for steam isolation, they do not satisfy the 10 CFR 50.36(c)(2)(ii) criteria.

LCO

This LCO requires that one MSIV in each of the two steam lines be OPERABLE. The MSIVs are considered OPERABLE when their isolation times are within limits, and they close on an isolation actuation signal.

This LCO requires that four turbine stop valves or their associated turbine control valve, six turbine bypass valves, and two moisture separator reheater 2nd stage steam isolation valves be OPERABLE. A valve is considered OPERABLE when its isolation time is within the safety analysis isolation time limit of 5 seconds and it closes on an MSIV

Change Number 56 A

TIER 1 LIST OF FIGURES

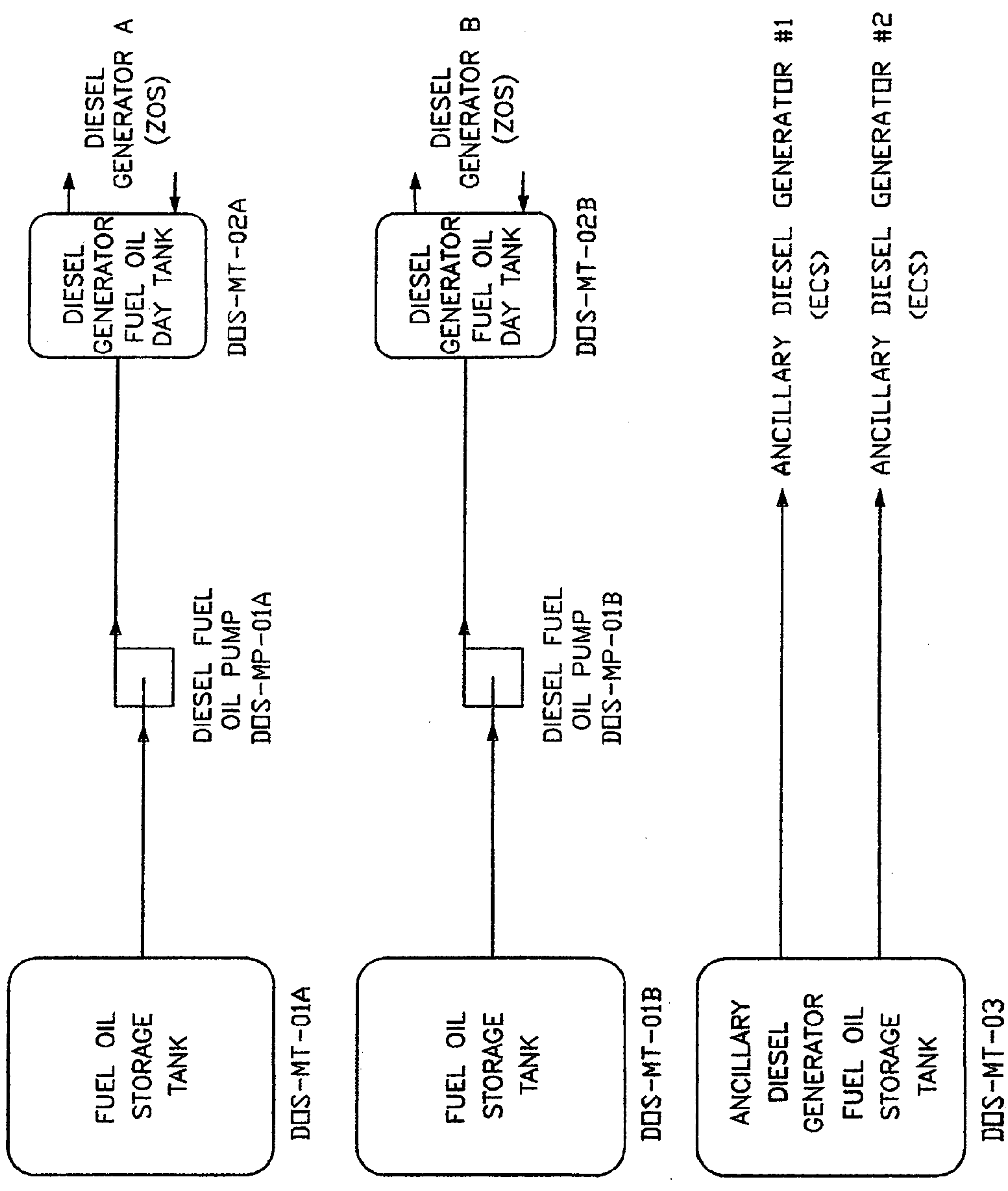

Figure 2.3.3-1
Standby Diesel ~~and Auxiliary Boiler~~ Fuel Oil System

Change Number 56 B

Table 14.3-1 (Sheet 1 of 4)

ITAAC SCREENING SUMMARY

Structure/ System Acronym	Structure/ System Description	Selected for ITAAC
ADS	Automatic Depressurization System	X
ASS	Auxiliary Steam Supply System	X
BDS	Steam Generator Blowdown System	X
CAS	Compressed Air System	X
CCS	Component Cooling Water System	X
CDS	Condensate System	X
CES	Condenser Tube Cleaning System	X
CFS	Turbine Island Chemical Feed System	X
CMS	Condenser Air Removal System	X
CNS	Containment System	X
CPS	Condensate Polishing System	X
CVS	Chemical and Volume Control System	X
CWS	Circulating Water System	X
DAS	Diverse Actuation System	X
DDS	Data Display Processing System	X
DOS	Standby Diesel ~~and Auxiliary Boiler~~ Fuel Oil System	X
DRS	Storm Drain System	
DTS	Demineralized Water Treatment System	X
DWS	Demineralized Water Transfer and Storage System	X
ECS	Main AC Power System	X
EDS	Non Class 1E DC and UPS System	X
EFS	Communication System	X
EGS	Grounding and Lightning Protection System	X

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May 21, 2010

ENCLOSURE 7

DCD Pages for Change Numbers 53, 54, and 57

(Public Redacted Version)

Change Number 53 A

Security-Related Information, Withhold Under 10 CFR 2.390d

SRI

Figure 3.3-11
Annex Building Plan View at Elevation 100'-0"

Security-Related Information, Withhold Under 10 CFR 2.390d

SRI

Figure 3.3-12
Annex Building Plan View at Elevation 117'-6"

Security-Related Information, Withhold Under 10 CFR 2.390d

SRI

Figure 3.3-13
Annex Building Plan View at Elevation 135′-3″

Change Number 53 B

SRI

Security-Related Information, Withhold Under 10 CFR 2.390d

Figure 3.7.2-19 (Sheet 1 of 10)

**Annex Building Key Structural Dimensions
Plan at Elevation 100′-0″**

SRI

Security-Related Information, Withhold Under 10 CFR 2.390d

Figure 3.7.2-19 (Sheet 2 of 10)

Annex Building Key Structural Dimensions
Plan at Elevation 107′-2″ and 117′-6″

SRI

Security-Related Information, Withhold Under 10 CFR 2.390d

Figure 3.7.2-19 (Sheet 3 of 10)

Annex Building Key Structural Dimensions Plan at Elevation 135′-3″

SRI

Security-Related Information, Withhold Under 10 CFR 2.390d

Figure 3.7.2-19 (Sheet 5 of 10)

Annex Building Key Structural Dimensions
Roof Plan at Elevation 154′-0″ and 181′-11 3/4″

Security-Related Information, Withhold Under 10 CFR 2.390d

SRI

Figure 3.7.2-19 (Sheet 6 of 10)

Annex Building Key Structural Dimensions
Section A - A

Security-Related Information, Withhold Under 10 CFR 2.390d

SRI

Figure 3.7.2-19 (Sheet 7 of 10)

Annex Building Key Structural Dimensions
Section B - B

Security-Related Information, Withhold Under 10 CFR 2.390d

SRI

Figure 3.7.2-19 (Sheet 8 of 10)

Annex Building Key Structural Dimensions
Section C - C

Change Number 54

SRI

Security-Related Information, Withhold Under 10 CFR 2.390d

Figure 1.2-8

Nuclear Island General Arrangement Plan at Elevation 117'-6" & 130'-0"

Security-Related Information, Withhold Under 10 CFR 2.390d

SRI

Figure 1.2-9

Nuclear Island General Arrangement
Plan at Elevation 117'-6" with Equipment

Security-Related Information, Withhold Under 10 CFR 2.390d

SRI

Figure 6.4-1

Main Control Room Envelope

Security-Related Information, Withhold Under 10 CFR

SRI

Figure 9A-1 (Sheet 6 of 16)

[Nuclear Island Fire Area Plan at Elevation 117'-6"]*

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

Security-Related Information, Withhold Under 10 CFR

SRI

Figure 12.3-1 (Sheet 7 of 16)

Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 117′-6″

Security-Related Information, Withhold Under 10 CFR

SRI

Figure 12.3-2 (Sheet 7 of 15)

**Radiation Zones, Post-Accident
Nuclear Island, Elevation 117′-6″**

Security-Related Information, Withhold Under 10 CFR

SRI

Figure 12.3-3 (Sheet 7 of 16)

Radiological Access Controls, Normal Operations/Shutdown Nuclear Island, Elevation 117′-6″

Change Number 57

Security-Related Information, Withhold Under 10 CFR 2.390d

SRI

Figure 1.2-9

Nuclear Island General Arrangement Plan at Elevation 117′-6″ with Equipment

ENCLOSURE 8

Responses to Action Items from March 17, 2010 Meeting, Non-Proprietary

1. CN21: Sumps and Drain Tanks
Question: Where do radioactive and non-radioactive drains go?
Response: Inside containment, all drains will potentially be radioactive. They are routed to the Liquid Radwaste System (WLS) Containment Sump and then get pumped to the WLS Waste Holdup Tanks.

   In the Radiation Control Areas of the Auxiliary, Annex, and Radwaste buildings, the drains go to the Radioactive Waste Drain System (WRS) Auxiliary Building Sump then get pumped to the WLS Waste Holdup Tanks.

   The non-radioactive drains are handled separately by systems such as Waste Water System and Sanitary Drainage System.

2. CN19: Primary Pressure Boundary Materials
Question: Is a new material being introduced in this change?
Response: A new material is not being introduced in the design. An existing material is just being grouped for convenience. This code case was approved by ASME Section IX but is applicable to Section III.

3. CN22: CRDM Material Consistency
Comment: Westinghouse is to clearly identify that the change to Table 5.2-1 is being made to identify the correct material.
Response: The correct material for the CRDM latch housing and rod travel housing is SA-182 not SA-336. This is consistent with the material shown for the reactor vessel components shown on sheet 1 of this same table.

4. CN40: MOV Post – Accident Monitoring Requirements
Comment: Provide a better explanation of the reason for the change. Does this change result in a decrease in safety?
Response: The change is for position indication only and does not impact the safety function of the valve.
Description of DCD Change:
DCD Chapter 16 Technical Specification, Table 3.3.3-1, Function 19, IRWST to RNS Suction Valve Status, is revised to specify that one position indication channel per valve (RNS-V022 and RNS-V023) must be OPERABLE. Table Note C has been revised to add "or flow path," since "penetration" applies to the containment isolation function.

   The Technical Specification Bases B 3.3.3 for Function 19 has been revised to include monitoring of the position of the two motor operated valves (RNS-V022 and RNS-V023) to verify that the flow path is isolated following postulated events.

   Reason for Design Change:

DCD Chapter 16 Technical Specification, Table 3.3.3-1 currently requires the In - Containment Refueling Water Storage Tank (IRWST) to Normal Residual Heat Removal System (RNS) Suction Valve Status (RNS-V023) to have two Required Channels/Divisions. Function 19, IRWST to RNS Suction Valve Status, which is equivalent to Function 18, Remotely Operated Containment Isolation Valve Position, is being revised since redundant indication of flow path isolation is provided by one 1E position indicator on each of the two valves (RNS-V022 and RNS-V023).

The Technical Specification Bases B 3.3.3 for Function 19 has been revised to include monitoring of the position of the two motor operated valves (RNS-V022 and RNS-V023) to verify that the flow path is isolated following postulated events to prevent loss of IRWST inventory into the RNS.

5. <u>CN47: PRHR Inlet Isolation Valve</u>
Comment: Westinghouse shall provide a better explanation of this change in order to clarify that this is an errata change and not something that reduces safety system functionality.
Response:
<u>Description of DCD Change</u>:
Remove the following sentence from DCD 7.6.2.1 describing the Passive Residual Heat Removal Heat Exchanger Inlet Isolation Valve: to prevent an inadvertent closure of the valve, redundant output cards are used in the protection and safety monitoring system cabinet.

<u>Reason for Design Change</u>:
DCD 6.3.2.2.8.4, Motor-Operated Valve Controls includes the following: Spurious movement of a motor-operated valve due to an electrical fault in the motor actuation circuitry, coincident with loss of coolant accident, has been analyzed and found to be an acceptably low probability event. In addition, power lockout in accordance with Branch Technical Position ICSB-18 is provided for those valves whose spurious movement could result in degraded passive core cooling system performance.

Table 6.3-1 provides a list of the remotely operated isolation valves in the passive core cooling system. These valves have various interlocks, automatic features, and position indication. Some valves have their control power locked out during normal plant operation. Periodic visual inspection and operability testing of the motor-operated valves in the passive core cooling system confirm valve operability.

Redundant controllers are provided for valves that are normally in the correct post-accident position that do not have power removed and locked out. This valve is normally in the correct post-accident position with power removed and locked out; therefore, redundant controllers are not required.

This design is consistent with the DCD 6.3.2.5.1 and DCD Table 6.3-1 valve requirements.

6. CN38: Loose Parts Detection
Comment: Clarify the reason for this change.
Response: The reason that NA is not applicable for C.6 has been revised to be similar to the reason given for C.3.a. Reporting requirements will be addressed by the COL applicants in accordance with 10CFR50.72 and 50.73 utilizing the guidance of NUREG-1022.

7. CN42: Operating Pressure
Question: The NRC asked that Westinghouse be clearer with respect to consistency issues and state what is right and what we are being consistent with.
Response: The correct units for operating pressure in this case are psia.

8. CN56: Deletion of "and Auxiliary Boiler" from Figure Titles
Comment: The NRC requested better justification. They asked Westinghouse to indicate where in the DCD the correct information is found.
Response: The diesel fired boiler was changed to an electric boiler since utilities reported problems with fuel fouling in stand-by diesel boilers with significant periods of downtime. This design change was incorporated into DCD-17 and necessitated a change to the title of the DOS system from Standby Diesel and Auxiliary Boiler Fuel Oil System to Standby Diesel Fuel Oil System everywhere it appears in Tier 1 and 2 of the DCD. These places included: Tier 1 – Master Table of Contents page i; List of Figures, page iv, Sections 1.4 and 2.3.3; Tier 2 – Master Table of Contents page xvi; page 3.2-15, Table 3.2-4, Chapter 9 TOC, pages vi and x; Chapter 9 list of figures, page xi; Section 9.5 (several places); Title of Fig 9.5-4 sheets 1, 2, and 3. The list of figures and title of Fig 2.3.3-1 in Tier 1 and title of Table 14.3-1 were missed.

9. CN3: Change PRHR Valves from Globe Valve to Ball Valve
Comment: The NRC asked Westinghouse to confirm that there is no degradation in safety due to changing PRHR valves from globe to ball type.
Response: No new failure modes were introduced by changing from a globe valve to a ball valve. Both valves are designed to fail open. Changing to the ball valve is an improvement in safety as the stroke time for the ball valve is less than that of the globe valve. Additionally, the weight of the ball valve is approximately half of that of the globe valve which is a benefit in the seismic and piping analysis. Finally, with the ball valve design, Class VI shutoff may be achieved. There are no additional or degraded safety requirements resulting from this change.

10. CN46: Identification of Reactor Internals Materials
Question: The NRC asked what new material(s) is being introduced, what is the industry experience with that material and characterize the performance characteristics of the material.
Response: The only "new" material being introduced is the UNS U06990 material which was previously not identified for the flow skirt. However it has been listed in this section for the RV clevis inserts and is also used for the AP1000 RV head penetrations and SG tubing. So it is not really a "new" material, the only change is that we identified a "new" component (the flow skirt) as using this material. The UNS U06990 material has been previously used for the pressure boundary penetrations in the replacement Reactor Vessel Closure Heads since about 2004. This material has also been used for the replacement Steam Generator U-tubes for several years. The UNS U06990 material was chosen for these replacement components due to its greater resistance to Stress Corrosion Cracking (SCC) as compared to the original Inconel 600 material in the original components.

11. CN44: Spray Nozzle Safe – End Material
Question: The NRC asked if, with the change in spray nozzle safe-end material, the new material met code requirements.
Response: Table 5.2-1 of the DCD identifies the acceptable safe end material for use on the pressurizer spray nozzle as SB-163 (N06690). However the material from SB-163 is tubing material which is not available in the diameter nor thickness which has been determined analytically acceptable. An acceptable material form is SB564 which is a forging material. The 1998 Edition of the ASME Code permits the use of the SB-564 (N06690) forging material for pressure boundary.

12. CN10: Change from "Operability" to "Applicable" in Tech Spec Bases
Question: The NRC asked for a better explanation of the change from "operability" to "applicable" in Tech Spec B.3.2.5. Although this change was not made as part of CN10, Westinghouse agreed to pursue this.
Response: The specific Bases change is as follows: change from "The OPDMS monitoring of SDM must be OPERABLE in MODES 1 and 2" to "The OPDMS monitoring of SDM is applicable in MODES 1 and 2 with Keff ≥ 1.0." The OPERABLE to applicable change is needed to clarify that LCO 3.2.5 does not require OPDMS to be OPERABLE. Thus, there are no Actions that apply if OPDMS is inoperable. If OPDMS is OPERABLE, only then is the LCO Applicable for SDM monitoring. If OPDMS is inoperable, then the LCO 3.2.5 OPDMS monitoring of SDM is not Applicable.