Westinghouse Electric Corporation

Energy Systems

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DCP/NRC0743
Docket No.: STN-52-003

February 19, 1997

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

TO: T. R. QUAY

SUBJECT: RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION AND OPEN ITEMS ASSOCIATED WITH SSAR CHAPTER 3

Dear Mr. Quay:

In a letter dated February 7, 1997, the NRC provided additional requests for additional information and updates of status for several areas being reviewed by ECGB and EMEB for AP600. Attachment 1 to this letter provides information and responses to a number of these items. Many of the items are resolved and do not require a response. The responses to other items will be provided later. The responses are grouped by the enclosures of the NRC letter. Also attached are markups of SSAR revisions that will resolve a number of these items. These changes will be included in Revision 11 of the SSAR.

The resolution of the items addressed in the attachment will permit the NRC staff to provide input to the FSER for a number of the subsections.

If you have any questions please contact D. A. Lindgren at (412) 374-4856.

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Attachments

cc: D. Jackson, NRC (w/attachments)

E004/1

250146

NRC Letter Enclosure 2

3. Open Item 3.2.2-1 (OITS 564) - Classification of Emergency Core Cooling System (ECCS)
Action W

In a letter to Westinghouse dated August 20, 1996, this open item was reported by the staff as being resolved. However, before this issue is considered resolved, the staff needs the following information and/or clarifications in the SSAR:

a. The staff has identified the components and systems listed below as part of ECCS systems that are classified as AP600 Class C (ASME Class 3):

- In-containment refueling water storage tank (SSAR Fig. 6.3-2)
- Accumulator (SSAR Fig. 6.3-1)
- Accumulator injection piping to discharge check valve V-028 (SSAR Fig. 6.3-1)
- Containment recirculating piping and valves to in-containment refueling water storage tank (IRWST) injection check valve V-122 (SSAR Fig. 6.3-1)
- Piping from 1st, 2nd & 3rd stage automatic depressurization valves (ADV) to the IRWST, including depressurization spargers (SSAR Fig. 5.1-5 & 6.3-2)

Westinghouse is requested to verify in the SSAR Subsection 3.2.2.5, that all of the above components and systems and any other Class 3 ECCS not listed above are included in the commitment to random radiography for all ECCS.

b. It appears that SSAR Subsection 3.2.2.5 is the only place in the SSAR that contains the above commitment. Since this commitment is not stated in either Table 3.2-3 or applicable P&IDs, how can the staff be assured that it will be implemented on all AP600 plants?

Westinghouse Response

a. Information will be added to SSAR subsection 3.2.2.5 to list the portions of systems to which the augmented weld inspection applies. The IRWST is not fabricated as a free standing tank but is formed using portions of containment internals structural modules. Reference to the requirements in 3.8.3.6.2 for inspection of the structural modules that form the IRWST will be included in subsection 3.2.2.5. A markup of these additions is attached.

b. Design and fabrication requirements such as the need for these inspections are included in internal AP600 design documents. A reference to the requirements in subsection 3.2.2.5 will be included in SSAR subsection 6.3.2.3. A markup of this addition is attached.

7. RAI 210.219 (OITS 3509) - Table 3.2-3, Passive Cont. Cooling System
Action N

The response to this issue in the letter dated December 2, 1996, is being evaluated by the staff.

Westinghouse Response

SSAR did not include all of the SSAR revision included in the December 2, 1996 letter. These will be included in Revision 11 of the SSAR.

9. RAI 210.221 (OITS 3512)- Table 3.2-3, Reactor System
Action W

Revision 10 to the SSAR, Table 3.2-3 provides acceptable responses to RAI 210.221a through d. However, the response to 210.221e is not acceptable. This portion of the RAI requested the basis for the Core Barrel Nozzle to be Class D and non-seismic when the Core Barrel is Class B and Seismic Category I. In a letter dated December 2, 1996, the response to this request states that the seismic classification of the nozzle would be changed to Category II, and the safety classification would remain as Class D because the nozzle does not provide core support and does not have to be safety-related. The staff's position is that the nozzle is an integral part of the core barrel (which is a safety-related component), and therefore should have the same safety and seismic classifications as the barrel. Table 3.2-3 should be revised to change the nozzle to be AP600 Class B and Seismic Category I. Therefore, OITS 3512 remains open.

Westinghouse Response

The core barrel nozzle will be changed to have the same classification as the core barrel. Westinghouse has determined that based on the criteria in subsection 3.2.2.5, the appropriate classification is Equipment Class C and the core support structures will be changed from Class B to Class C. Class C is a safety-related class to which 10 CFR Part 50 Appendix B applies and the seismic category does not change. Also the guide tube assemblies will be changed from Class D, seismic Category II to Class C, seismic Category I. A markup of these changes is attached.

13. Open Item 3.6.2-1 (OITS 592) - Subcompartment Design
Action W

The response to this issue in the letter from McIntyre to Quay dated October 23, 1996, does not appear to contain the detailed information requested by the staff during the review meeting with Westinghouse on July 25 & 26, 1995. As stated in the DSER, Section 3.6.2, page 3-94, the staff's position is that a minimum subcompartment pressure which bounds the effects of a high energy pipe break (with consideration of leak-before-break (LBB) acceptance) must be determined. Specifically, the staff requests that for all subcompartments both inside and outside containment, SSAR Subsections 3.8.3.5 and 3.8.4.3.1.4 be revised to state that those compartments containing high energy piping are designed to the worst case of either the 5 psi load (the 7.5 psi load for the CVS room) or the double ended pipe rupture of the applicable high energy pipe.

Westinghouse Response

The subcompartment design pressure bounds the effects of postulated breaks. The SSAR will be revised to specify that.

SSAR Revisions

The sixth paragraph of subsection 3.8.3.5 will be revised as follows:

> The determination of pressure and temperature loads due to pipe breaks is described in subsections 3.6.1 and 6.2.1.2. Subcompartments inside containment containing high energy piping are designed for a pressurization load of 5 psi. The pipe tunnel in the CVS room (room 11209, Figure 1.2-6) is designed for a pressurization load of 7.5 psi. These subcompartment design pressures bound the pressurization effects due to postulated breaks in high energy pipe. The design for the effects of postulated pipe breaks is performed as described in subsection 3.6.2. Determination of pressure loads resulting from actuation of the automatic depressurization system is described in subsection 3.8.3.4.3.

In subsection 3.8.4.3.1.4 the paragraph under P_a will be revised as follows:

> The main steam isolation valve (MSIV) and steam generator blowdown valve compartments are designed for a pressurization load of 5 psi. The subcompartment design pressure bounds the pressurization effects due to postulated breaks in high energy pipe. Determination of subcompartment pressure loads is discussed in subsection 6.2.1.2.

15. RAI 210.40 (OITS 3702) - Break Exclusion in Steam Generator (SG) Blowdown, Startup FW, and Chemical and Volume Control System (CVS) Lines
Action W

In a letter from McIntyre to Quay dated October 23, 1996, and in the OITS 3702 report, Westinghouse states that additional information on the startup line, including the isometric drawings will be provided during a forthcoming meeting with the staff. This issue will be discussed during the next meeting or a telephone conference.

In addition, Revision 10 to SSAR Subsection 3.6.2.1.1.4 added portions of the Chemical and Volume Control System (CVS) to the list of break exclusion areas. These new areas include makeup piping from containment to the anchors upstream of the outside isolation valve and downstream of the inside isolation valve, including branch connections. Revision 10 did not revise SSAR Figure 3E-5 to identify these areas. Therefore, the staff requests more information relative to the exact location of the anchors, the length of piping from the inside and outside isolation valves to each anchor, and the location and lengths of all applicable branch lines.

Westinghouse Response

Westinghouse is revising the break exclusion area for the CVS makeup line to be from the outside containment isolation valve to the inside containment isolation valve. Westinghouse has provided copies of the isometric drawings for this line for review in the Westinghouse Rockville licensing

office. This change will add two additional break locations to be considered in the evaluation of high energy breaks. The fluid in this line is cold and a break will not result in pressurization. A copy of the SSAR markup showing changes for the revised break exclusion area are attached.

16. Open Item 3.6.2.3-1 (OITS 595) - Break Locations and Stress Summary
Action N

In Revision 10 to the SSAR contains a significant revision to Subsection 3.6.2.5 which provides additional information on the pipe break hazard analysis. The staff's preliminary evaluation of this submittal resulted in the following request:

> As discussed under Open Item 3.6.2.3-5 (OITS 599) below, Westinghouse has submitted a revision to SSAR Subsection 3.6.1.3.2 which refers to the pipe rupture hazards analysis. In addition to the information in Revision 10, (1) add a reference in SSAR Subsection 3.6.2.5, to the new information in Subsection 3.6.1.3.2 that is applicable to the hazards analysis, and (2) in Subsection 3.6.4.1, state that the as-built reconciliation of the hazards analysis will be in accordance with the criteria in SSAR Subsections 3.6.2.5 and 3.6.1.3.2.

Westinghouse Response

(1) A reference to the criteria in subsection 3.6.1.3.2 will be added to the paragraph under Essential Target Evaluation in subsection 3.6.2.5.
(2) A reference to the criteria in subsection 3.6.1.3.2 and 3.6.2.5 will be added to subsection 3.6.4.1.

A markup is attached.

18. Open Item 3.6.2.3-5 (OITS 599) - Separating Structures
Action W

In Revision 10 to SSAR Subsection 3.6.1.3.2, information was added which provides a basis for resolving this issue as a part of the pipe rupture hazards analysis. Based on a preliminary review of this submittal, the staff has no further requests for information except to repeat the request in this open item to delete the exception to the standard review plan (SRP) Section 3.6.2 BTP MEB 3-1, Section B.1.c.(4) in WCAP-13054, Revision 2.

Westinghouse Response

WCAP-13054 will be revised to remove the exception to criteria B.1.c.(4). A markup of the change is attached.

23. Open Item 3.9.3.3-2 (OITS 793) - Anchor Bolts for Pipe Supports
Action W

In a letter from McIntyre to Quay dated October 23, 1996, Westinghouse responded to this item by referencing Revision 9 to SSAR Subsection 3.9.3.4. Revisions 9 and 10 contain no change to this portion of Subsection 3.9.3.4. It still commits only to the baseplate flexibility requirements of IE Bulletin 79-02 and is silent on the factors of safety for concrete expansion anchor bolts. Since the factor of safety issue is being evaluated by the staff under DSER Open Item 3.8.4.2-2, Subsection 3.9.3.4 should contain a reference to the applicable portion of SSAR Subsection 3.8.4 for information relative to these factors of safety.

Westinghouse Response

Supplemental requirements for fastening anchor bolts to concrete are provided in subsection 3.8.4.5.1. reference to these requirements will be added to subsection 3.9.3.4. A markup of this addition is attached.

24. Open Item 3.10-1 (OITS 813) - Use of Seismic Experience Data
Action W

Revision 10 to SSAR Section 3.10.6 states that the COL applicant, as a part of the Combined License application, will identify equipment qualified based on experience and include details of the methodology and the corresponding experience data. This agrees with the staff's request on this item, and is acceptable. However, the exception to SRP 3.10 in Revision 2 to WCAP-13054 contains statements which either need to be deleted or clarified. The first two sentences imply that IEEE 344-1987 is acceptable relative to the use of experience data. Regulatory Guide (RG) 1.100, Revision 2 states that this method of qualification in IEEE 344-1987 will be evaluated by the staff on a case-by-case basis. It appears to the staff that the exception in the WCAP is relative to RG 1.100, Revision 2. These two sentences should be revised to reflect the position in RG 1.100, Rev. 2. In addition, the discussion relative to Generic Issue A-46 is not applicable to new plants. The staff's position is that A-46 is only used for verification of equipment in operating plants, and is not acceptable for qualification of equipment in advanced light water reactors (ALWRs). This discussion should either be deleted or revised.

Westinghouse Response

The discussion in WCAP-13054 on the Criteria 1 for SRP 3.10 will be revised to clarify that the exception is to the revision on Regulatory Guide 1.100 and IEEE 344 and that the AP600 is in conformance with Regulatory Guide 1.100, Revision 2. The requirement that the combined license applicant identify use of experience based data for equipment qualification and the methodology used is included in subsection 3.10.6. A draft markup of the exception in WCAP-13054 is attached.

NRC Letter Enclosure 5

1. DSER# 3.9.2.3-2 (783) - Flow-induced vibration prediction analysis
   Action W

In SSAR Revision 10, the first paragraph of Subsection 3.9.2.3 indicates that the flow-induced vibration assessment is documented in WCAP-14761, which is also included in the reference list in SSAR Section 3.9.9. This is acceptable. The WCAP-14761 is a replacement of previous report MI01-GER-001, which was submitted by Westinghouse and reviewed by the staff and found acceptable.

However, the reactor internals of the first AP600 plant is designated as the prototype as defined in SRP 3.9.2 and RG 1.20 for vibration assessment of AP600 reactor internals. Information of vibration assessment from reference plants, which include H. B. Robinson, DOEL 3 and 4, etc. may only be used in vibration prediction analysis for the prototype and should not be confused with the prototype. The wording in SSAR Sections 3.9.2.3 and 3.9.2.4 should be revised to avoid confusion between the "prototype" and the "reference plants."

Westinghouse Response

Subsections 3.9.2.3 and 3.9.2.4 will be revised to delete the portion that stated that prototype and reference were equivalent. A markup of the changes is attached.

3. DSER# 3.9.5-1, RAI 210.226, (OITS 3517) - 20% damping value for fuel assemblies
   Resolved

Information provided in Westinghouse letter NSD-NRC-97-4933, dated 1/8/97, indicates that the damping value is justified by testing and is consistent with evaluations for Westinghouse-designed fuel in operating nuclear power plants. This is acceptable. However, Westinghouse needs to provide a suitable reference in the SSAR.

Westinghouse Response

A paragraph will be added to subsection 3.9.2.6 to identify the damping for the fuel assemblies and reference WCAP-8236. A markup of the additions is attached.

NRC Letter Enclosure 1

2. RAI 210.227 - SSAR 3.9.6 (IST)
   Action W

Revise the SSAR to reflect correct reference of OM Standards, OMa-1988 or the 1990 Edition of the OM Codes. From the January 31, 1997, telephone conference, Westinghouse will send a letter requesting an exemption and revise the SSAR accordingly.

Westinghouse Response

AP600 used the 1990 Edition of the Code as the baseline for preparation of the inservice testing plan. Letter NSD-NRC-97-4986, dated February 14, 1997 requesting an exemption from the requirements in 10 CFR 50.55a for an earlier version of the Code has be sent to the NRC. The fourth paragraph of subsection 3.9.6 will be revised to remove the reference to ANSI in the name of the Code. A markup of the changes is attached.

3. RAI 210.228 - SSAR 3.9.6 (IST)
   Action W

The main feedwater (FW) check valves, SGS-V058A/B appear to have a safety function to close based on SSAR Subsection 10.4.7.1.1. Revision 10 still indicates that these valves have a safety-related function, but they are not included in the ISTP (IST Program). In the January 31, 1997, telephone conference, Westinghouse stated that these valves do not perform a safety function and will move the FW check valve description from SSAR Subsection 10.4.7.1.1 to SSAR Subsection 10.4.7.1.2.

Westinghouse Response

These valves do not perform a safety function. The description of the feedwater check valve function is located in an incorrect location and will be moved from SSAR Subsection 10.4.7.1.1 to SSAR Subsection 10.4.7.1.2. A markup of the changes is attached.

5. RAI 210.230 - Safety Relief Valve Test - SSAR 3.9.6 (IST)
   Action W

The SSAR (Rev. 10) was revised to include "5 years and 20% in 2 years. For Class 2/3, the SSAR should be revised for consistency to include "10 years and 20% in 4 years." From the January 31, 1997, telephone conference, Westinghouse will revise the SSAR.

Westinghouse Response

The column in Table 3.9-16 will be revised to include 10 years and 20% in 4 years for the appropriate valves. A markup of the changes is attached.

6. RAI 210.231 - SSAR 3.9.6 (IST)
   Action W

The discussion of Issue 87 in SSAR Section 1.9 should be revised to state that valves built to Section III are required to be tested in accordance with the ASME Code. Revision 10 still states that these valves may be tested in accordance with the OM Code. From the January 31, 1997, telephone conference, Westinghouse will revise the SSAR.

Westinghouse Response

The second paragraph of the response for Issue 87 in Section 1.9 will be revised to indicate that compliance with the ASME OM is required rather than being an option. Also, the response will be revised to make the response consistent with the position in subsection 3.9.6.2.2 on testing of valves. A markup of the changes is attached.

NRC Letter Enclosure 6

2. OITS Item No. 801
   Action W

Note: This is only a portion of this item. The balance of the question will be answered with items related to valve qualification.

Westinghouse had added SSAR Subsection 3.9.6.2.3 to address valve disassembly and inspection. The disassembly and inspection program must also be addressed in Section 3.9.8, as the COL will have to develop this program.

Westinghouse Response

Specific reference to the valve disassembly and inspection program will be added to subsection 3.9.8.4. A markup of the SSAR changes is attached.

3. OITS Item No. 805
   Action W

Valves RNS-V002A/B are CIVs and are Type C tested per SSAR Table 6.2.3-1. These valves should be leak tested in the ISTP, Table 3.9-16. In the January 31, 1997, telephone conference, Westinghouse will revise SSAR Table 6.2.3-1.

Westinghouse Response

Because of the function and lay out of the normal residual heat removal system these valves are not subject to a containment leak test. The notes for these valves in Table 6.2.3-1 will be revised to clarify this and be consistent with the information in Table 3.9-16. A markup of the changes is attached.

4. OITS Item No. 807
   Action W

Per NRC comments on the testing deferral, revise Note 4, Note 9, Note 11, and Note 21 in SSAR Table 3.9-16. In the January 31, 1997, telephone conference, Westinghouse stated they will provide additional justification on the use of solenoid operated valves for the head vent (Note 4); provide additional information on "sufficiently long" cold shutdown times (Note 9); delete RNS-PL-V046 from

Note 21 and the table (Note 21); provide additional information on the testing capabilities of VES-PL-V0008 A and B (Note 21). The staff will respond to the Westinghouse December 17, 1996, letter regarding Note 11.

Westinghouse Response

Note 4 will be revised to identify the potential for quarterly testing to cause valve leakage. Solenoid valves are chosen for this application because they are the best overall choice to meet the several disparate design requirements. These valves have safety-related functions to transfer open and to transfer close. Air operated valves are not well suited to such an application because they are normally capable of safety-related transfer in only one direction. To achieve safety-related transfer capability in two directions requires the use of a piston operator and a safety-related air supply. Motor-operated valves are also not well suited to this application because they are larger and heavier with an extended operator that makes them difficult to locate and support. Motor-operated valves are also less reliable. Both air operated valves and motor-operated valves have packing which is subject to leakage.

Note 9 will be revised to add a partial exercise testing of the accumulator check valves at longer cold shutdowns. Full stroke exercise testing during refueling shutdowns is maintained.

Note 11 will be revised to provide additional information on the test device used to open the recirculation check valve disks.

Reference to Note 21 will be deleted from the entry for valve RNS-PL-V046.

A markup of the changes is attached.

5. OITS Item No. 809
   Action W

Delete the first sentence of the second paragraph in SSAR Subsection 3.9.6.2, and move the second sentence to the end of the paragraph. Westinghouse will revise the SSAR.

Westinghouse Response

This paragraph will be revised to clarify the requirements for testing of valves with RTNSS important missions. A markup of the changes is attached.

6. OITS Item No. 1730
   Action W

Valve PCS-V014A is a normally closed stop check valve (P&ID Fig. 6.2.2.1 of the SSAR, Rev. 6 and Rev. 9) with a safety function to open. However, no check exercise is specified and the valve is still identified as a Category B valve in Revision 10. Westinghouse will revise the SSAR.

Westinghouse Response

A check exercise test will be included in Table 3.9-16 for this valve. A markup of the change is attached.

7. OITS Item No. 1731
   Action W

Westinghouse has stated that valves RCS-PL-150VA-D have an active function to move to the open position. However, SSAR Fig. 5.1-5 identifies these valves as failed closed. The IST Table should be revised such that these valves are subject to a fail-safe test, or the P&ID should be revised. Westinghouse will include the testing in the SSAR.

Westinghouse Response

Table 3.9-16 will be revised to include a safety function of Active-to-Fail for these valves.

13. RAI Q952-96
    Action N

In a letter response dated May 13, 1996, Westinghouse continues to state that the ADS valves will be tested at conditions determined with input from type selection testing. The qualification testing of the prototypical ADS valves should be performed under design basis conditions. From the January 31, 1997, telephone conference, Westinghouse stated the response did not apply to ISTP. The staff will review the issue further for acceptability.

Westinghouse Response

This question is in reference to a response to an RAI about ADS system testing. Westinghouse has been very specific that these tests were not valve qualification testing. Since this RAI response was not about valve qualification, there is no need to revise it to address a valve qualification question.

Issue 87 Failure of HPCI Steam Line Without Isolation

Discussion:

Generic Safety Issue 87 addresses the uncertainty regarding the operability of the motor-operated isolation valves for the steam supply lines of the high-pressure coolant injection (HPCI) system in boiling water reactors following a postulated break in the supply line. A break in the line could lead to high flow or high differential pressure that may inhibit closure of the isolation valve. These valves typically cannot be tested in-situ for the design flow rates and pressures. Although the AP600 does not have a high-pressure coolant injection system, it does have isolation valves designed to close against high flow or high pressure differential in the event of a postulated pipe break.

The issue of the operability of motor-operated valves has received considerable attention since Generic Safety Issue 87 was initiated. The NRC provided guidance for inservice testing of motor-operated, safety-related valves in Generic Letter 89-10. SECY-93-087 identifies the proposed position on inservice testing of safety-related valves for advance light water reactors. The guidance in these documents recommends that safety-related valves be tested under full flow under actual plant conditions where practical. EPRI has a program to demonstrate operation of motor-operated valves.

AP600 Response:

Safety-related valves must meet the requirements of ASME Code, Section III to provide pressure boundary integrity. Valves and valve operators are sized to provide operation under a full range of design basis flow and pressure drop conditions. For the AP600, safety-related motor-operated valve designs are subject to qualification testing to demonstrate the capability of the valve to open, close, and seat against maximum pressure differential and flow. The requirements for this testing are based on ANSI B16.41, "Functional Qualification Requirements for Power Operated Active Valve Assemblies for Nuclear Power Plants." See subsection 5.4.8 for an outline of AP600 valve requirements.

RAI 210.231

The in-service testing program for safety-related valves is discussed in subsection 3.9.6. ~~Where practical, m~~Motor-operated valves ~~and check valves~~ are to be operability tested as outlined in subsection 3.9.6.2.2. ~~under full flow under actual plant conditions.~~ Subsection 3.9.6.2.2 includes a discussion of the factors to be considered to determine which valves and the test conditions to used for oberability testing of power-operated valves. Sufficient flow is provided to fully open check valves during testing unless the maximum accident flows are not sufficient to fully open the check valve. The valves built to ASME Code, Section III are ~~may be~~ tested in compliance with the requirements found in the ASME code, "Code for Operation and Maintenance of Nuclear Power Plants." For additional information on inservice testing of safety-related valves, see subsection 3.9.6.

- Handle spent fuel, the failure of which could result in fuel damage such that significant quantities of radioactive material could be released from the fuel and results in offsite doses greater than normal limits (for example, new and spent fuel racks, the bridge, and the hoist)
- Maintain spent fuel sub-critical
- Monitor radioactive effluent to confirm that release rates or total releases are within limits established for normal operations and transient operation
- Monitor variables to indicate status of Class A, B or C structures, systems, and components required for post-accident mitigation
- Provide for functions defined in Class B where structures, systems, and components, or portions thereof are not within the scope of the ASME Code, Section III, Class 2.
- Provide provisions for connecting temporary equipment to extend the use of safety related systems. See subsection 1.9.5 for a discussion of actions required for an extended loss of onsite and offsite ac power sources.

OI 3.2.2-1

The components and portions of systems that provide emergency core cooling functions and are required to have radiography of a random sample of welds during construction include the following:

- Accumulators
- Injection piping from the accumulators to the reactor coolant system isolation check valves in the direct vessel injection line
- Piping from the in-containment refueling water storage tank (IRWST) and recirculation screens to the reactor coolant system isolation check valves in the direct vessel injection line
- Piping from the Stage 1, 2, and 3 automatic depressurization system valves to the IRWST including the spargers.

The IRWST is formed from portions of structural modules that are elements of the containment internal structures. The inspection requirements for the welds in these structural modules are provided in Subsection 3.8.3.6.2.

3.2.2.6 Equipment Class D

Class D is nonsafety-related with some additional requirements on procurement, inspection or monitoring.

Table 3.2-3 (Sheet 31 of 61)

AP600 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP600 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MI-01	Reactor Upper Internals	C~~B~~	I	ASME III, CS	
RXS-MI-02	Reactor Lower Internals	C~~B~~	I	ASME III, CS	
RXS-MI-10	Non-Threaded Fasteners	D	NS	ASME III, CS	
RXS-MI-11	Threaded Structural Fasteners	C~~B~~	I	ASME III, CS	
RXS-MI-20	Lower Core Support Plate	C~~B~~	I	ASME III, CS	
RXS-MI-21	Secondary Core Support	D	II	ASME III, CS	
RXS-MI-22	Vortex Suppression Plate	D	II	ASME III, CS	
RXS-MI-23	Radial Reflector Assembly	D	II	ASME III, CS	
RXS-MI-24	Radial Supports [4]	C~~B~~	I	ASME III, CS	
RXS-MI-25	Core Barrel	C~~B~~	I	ASME III, CS	
RXS-MI-26	Core Barrel Nozzle	C~~D~~	I~~II~~	ASME III, CS	
RXS-MI-27	Head and Vessel Pins	D	II	ASME III, CS	
RXS-MI-28	Lower Support Plate Fuel Alignment Pins	C~~B~~	I	ASME III, CS	
RXS-MI-29	Core Barrel Hold Down Spring	C~~B~~	I	ASME III, CS	
RXS-MI-50	Upper Support	C~~B~~	I	ASME III, CS	
RXS-MI-51	Upper Core Plate	C~~B~~	I	ASME III, CS	
RXS-MI-52	Support Columns [38]	C~~B~~	I	ASME III, CS	
RXS-MI-53	Guide Tube Assemblies [61]	C~~D~~	I~~II~~	ANSI B31.1	
RXS-MI-54	Upper Support Plate Fuel Alignment Pins	C~~B~~	I	ASME III, CS	
RXS-MI-55	Upper Core Plate Inserts	C~~B~~	I	ASME III, CS	
RXS-MI-56	Safety Injection Deflector	D	II	ANSI B31.1	
RXS-MI-57	Irradiation Specimen Guide Tubes	D	II	ANSI B31.1	
RXS-MI-58	Head Cooling Nozzles	D	II	ANSI B31.1	
RXS-MV-10	Reactor Integrated Head Package	C	I	AISC-690	
RXS-MV-10A	Integrated Head Package Shroud	C	I	ASME-NF	
RXS-MV-10B	Integrated Head Package Seismic Support Plate	C	I	ASME-NF	

- For evaluation of spray wetting, flooding, and subcompartment pressurization effects, longitudinal cracks (with crack flow areas of 1 square foot) are postulated in the main steam and main feedwater piping. The dynamic effects of pipe whip and jet impingement are not evaluated for these cracks. Locations having the greatest effect on essential equipment are chosen.

- Guard pipe assemblies for high-energy piping in the containment annulus region between the containment shell and shield building that are part of the containment boundary are designed according to the rules of Class MC, subsection NE, of the ASME Code. The following requirements also apply. The design pressure and temperature are equal to or greater than the maximum operating pressure and temperature of the enclosed process pipe under normal plant conditions. Level C service limits of the ASME Code, Section III, Paragraph NE-3221(c), are not exceeded by the loadings associated with containment design pressure and temperature in combination with a safe shutdown earthquake. The guard pipe assemblies are subjected to a pressure test performed at the maximum operating pressure of the enclosed process pipe.

Areas of system piping where no breaks, except as noted in subsections 3.6.1.3 and 3.6.1.2.2, are postulated are as follows:

- The main steam piping, from the containment penetration flued head outboard weld, to the upstream weld of the auxiliary building anchor downstream of the main steam isolation valves, including the main steam safety valves and the connecting branch piping

- The main feedwater piping, from the containment penetration flued head outboard weld, to the auxiliary building anchor upstream of the isolation valve~~, including branch connections~~

- The startup feedwater piping from the containment penetration to the auxiliary building anchor upstream of the isolation valve ~~including branch connections~~

- The steam generator blowdown piping from the containment to auxiliary building anchor downstream of the isolation valve

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- **The chemical and volume control system makeup piping from the containment to the ~~anchor upstream of the~~ outboard isolation valve ~~including branch connections~~.**

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The chemical and volume control system makeup piping from the containment to the ~~anchor downstream of the~~ inboard isolation valve ~~including branch connections~~.

All other fluid system containment penetrations are for moderate-energy systems or for pipe of 1-inch nominal diameter or smaller. See subsection 6.2.3 for a discussion of containment penetrations.

Essential Target Evaluation) BOL1

To complete the essential target evaluation jet parameters, volumetric area of affected compartments, plant layout, and separating structures are considered. Parameters that determine the shape of the jet and the magnitude of the jet and thrust loads include pressure, temperature, and friction losses between the break and the reservoir. The volumetric area affected is determined by considering jet shape and loads at the postulated location of the breaks. Where an initial evaluation of essential targets indicated adverse effects, layout may be changed to relocate the target or postulated break. If necessary, the location of whip restraints and jet shields is established to protect essential systems and components. Essential equipment protected by pipe whip restraints or jet shields is listed in Table 3.6-3. The criteria for the break location postulated for evaluation of separating structures is outlined in subsection 3.6.1.3.2. OI 3.6.2.3-1

Verification of the Pipe Break Hazard Analysis) BOL1

The ASME Code, Section III, requires that each plant have a Design Report for the piping system that includes as-built information. Included in the Design Reports are the loads and loading combinations used in the analysis. Where mechanistic pipe break requirements are used to eliminate the evaluation of dynamic effects of pipe rupture in ASME Code, Section III, Class 1, 2, and 3 piping system, the basis for the exclusion is documented in the Design Report.

As-built reconciliation of the pipe break hazard analysis is addressed by the Combined License applicant.

3.6.2.6 Evaluation of Flooding Effects from Pipe Failures) BOL1

The effect of flooding due to high and moderate energy pipe failures on essential systems and components is described in Section 3.4.

3.6.2.7 Evaluation of Spray Effects from High- and Moderate-Energy Through-Wall Cracks) BOL1

Essential systems and components are evaluated for the potential effects of spray from high- and moderate-energy through-wall cracks. Spray effects are assumed to be limited to the compartment where the pipe failure occurs. The spray is assumed to wet unprotected components in the compartment. It is further assumed the spray does not damage non-electrical passive components, including piping, ducts, valve bodies, or mechanical components of valve operators. Spray may cause failure of electrical components not designed to withstand wetting. Components protected by NEMA 4 or NEMA 12 enclosures are not affected by spray effects.

The safe shutdown components inside containment are subject to wetting from design basis events inside containment. These conditions bound the effects of spray from moderate energy cracks. Sensitive components are qualified for this environment as described in Section 3.11.

3.6.3.4 Documentation of Leak-before-Break Evaluations

The leak-before-break evaluation is used to support the elimination of dynamic effects of pipe breaks from the loading conditions for the piping analysis. An evaluation of leak-before-break using the as-built configuration of the piping system and supports is required as part of the Design Report of the as-built configuration required to meet ASME Code requirements. Appendix 3B contains a discussion of the bounding analysis methods for the leak-before-break evaluation.

The analysis methods, criteria, and loads used for evaluation of stress in piping systems are outlined in subsections 3.7.3 and 3.9.3. The seismic input bounds the soil design profiles outlined in subsection 3.7.1.4 and Appendicies 2A and 2B. The evaluation also bound soil profiles qualified using site specific evaluations as outlined in subsection 2.5.4.5.5

3.6.4 Combined License Information) BOLD

3.6.4.1 Pipe Break Hazard Analysis) BOLD

Combined License applicants referencing the AP600 certified design will address as built reconciliation of the pipe break hazards analysis in accordance with the criteria outlined in subsections 3.6.1.3.2 and 3.6.2.5. OI 3.6.2.3-1

3.6.4.2 Leak-before-Break Evaluation) BOLD

Combined License applicants referencing the AP600 certified design will address: 1) verification that the as-built stresses, diameter, wall thickness, material, welding process, pressure, and temperature in the piping excluded from consideration of the dynamic effects of pipe break are bounded by the leak-before-break bounding analysis; 2) a review of the Certified Material Test Reports or Certifications from the Material Manufacturer to verify that the ASME Code, Section III strength and Charpy toughness requirements are satisfied; and 3) complete the leak-before-break evaluation by comparing the results of the final piping stress analysis with the bounding analysis curves documented in Appendix 3B.

3.6.5 References) BOLD

1. NUREG/CR-2913, "Two-Phase Jet Loads," January 1983.

2. WCAP-8077, "Ice Condenser Containment Pressure Transient Analysis Methods," March 1977.

3. ASME/ANSI-B31.1, Code for Power Piping, 1989 Addenda to 1989 Edition.

4. ANSI/ANS-58.2-1988, "Design Bases for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture."

5. Moody, F. J., Fluid Reaction and Impingement Loads, paper presented at the ASCE Specialty Conference, Chicago, December 1973.

Table 3.6-3

ROOMS WITH HIGH ENERGY PIPE BREAKS AND POTENTIAL ESSENTIAL TARGET INTERACTION

Elevation	Room Numbers*	High Energy Break Source	Essential Equipment Protected by Whip Restraints or Jet Shields
66'-6"	None		
82'-6"	11201	RCS Press. Spray - Terminal End	RCS-ADS valves: V004A, V004C, V014A, V014C
	11204		None
	11209		None
96'-6"	11204		None
	11209		None
100'-0" and 107'-2"	11209 Pipe chase	SGS Blowdown Piping - Terminal End	CVS Makeup, CVS Letdown, CVS Hydrogen Supply, and SGS steam generator blowdown piping
	~~11209 Pipe chase~~	**CVS Makeup Piping - Terminal End**	**CVS Makeup valve V091**
	11300		None
	11301		None
	11303/ 11304	RCS Makeup Piping - Intermediate Break	~~RCS and~~ SGS sg blowdown and sg drain Piping, RCS pressurizer pressure and level instrumentation, and Pressurizer support steel
117'-6"	11400	SGS Start Up Feedwater Piping - Terminal end	None ~~Raceways and cables for Divisions A/C and B/D~~
	11401		~~None~~ RCS-ADS valves: V004A, V004C, V014A, V014C are protected from a break located in room 11403
	11402		Steam Generator supports are protected from a break located in room 11400

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	11403	RCS Press Spray - Terminal End	None
	11403	RCS Letdown - Intermediate Break	Raceways for Divisions A/C and B/D
	~~11403~~	RCS Press Spray - Intermediate Break	RCS-ADS valves: V004A, V004C, V014A, V014C
135'-3"	~~None~~ 11503	RCS Press Spray - Terminal End	RCS-ADS valves: lower tier platform support steel
160'-6" and 153'-0"	11601	SGS Start Up Feedwater Piping - Terminal end SGS Main Feedwater Piping- Terminal End	RCS head vent piping SGS level instrumentation piping
	11602	SGS Main Feedwater Piping- Terminal End	~~None~~ SGS level instrumentation piping
	11603	RCS ADS Stage 1 Piping - Teminal End	RCS piping and ADS valves 002B, 003B, 012B, & 013B Raceways and cables for Divisions A/C and B/D
	11703	RCS ADS Stage 1 Piping - Teminal End	RCS piping and ADS valves 002A, 003A, 012A, & 013A Raceways and cables for Division ~~B/D~~ A/C
	12244	**CVS Makeup Piping - Terminal End**	**CVS Makeup valve V090**

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* See Figures 1.2-1 through 1.2-8, 1.2-10, and 1.2-11 for room numbers

- Heat exchangers
- Filters
- Passive valves

Dynamic analysis without testing is used to qualify heavy machinery too large to be tested. For active equipment, it is verified that deformations due to seismic loadings do not cause binding of moving parts to the extent that the component cannot perform its required safety function.

Dynamic Testing

Dynamic testing is used for components with mechanisms that must change position in order to perform the required safety function. Section 3.10 discusses the seismic qualification of electrical equipment and combinations of valves and valve operators. Such components include the following:

- Electric motor valve operators
- Valve position sensors
- Similar appurtenances for other active valves

Combinations of Analysis with Testing

Combinations of analysis, static testing, and dynamic testing are used for seismic qualification of complex valves. Section 3.10 discusses the requirements for these combinations for equipment, which includes the following:

- Main steam and main feedwater isolation valves
- Other active valves

3.9.2.3 Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions

The vibration characteristics and behavior due to flow-induced excitation are complex and not readily ascertained by analytical means alone. Assessment of vibrational response is done using a combination of analysis and testing. Comparisons of results obtained from reference plant vibration measurement programs have been used to confirm the validity of scale model tests and other prediction methods as well to confirm the adequacy of reference plant internals regarding flow induced vibration. ~~In the following discussion the term "reference plant" is equivalent to the term prototype as used and defined in Standard Review Plan 3.9.2 and Regulatory Guide 1.20 for vibration assessment of reactor internals.~~ The flow-induced vibration assessment is documented in WCAP-14761 (Reference 18). OI 3.9.3.3-2

Reactor components are excited by flowing coolant, which causes oscillatory pressures on the surfaces. The integration of these pressures over the applied area provides the forcing functions to be used in the dynamic analysis of the structures. In view of the complexities of the geometries and the random character of the pressure oscillations, a closed form

The reactor coolant canned motor pumps of the AP600, have the same rotational speed and the same number of impeller blades as in previous plants. Therefore, a significant change in vibration is not expected. The forcing function frequencies are similar to previous plants. For calculation of pump induced pulsations acting on the AP600 reactor internals, the pulsation level at the pumps is taken to be the same as the level previous shaft seal pumps. Since the horsepower of the AP600 pumps is lower than in shaft seal pumps, the shaft seal pulsation is a conservative analysis basis for the AP600

3.9.2.4 Pre-operational Flow-Induced Vibration Testing of Reactor Internals

The pre-operational vibration test program for the reactor internals of the AP600 conducted on the first AP600 is consistent with the guidelines of Regulatory Guide 1.20 for a comprehensive vibration assessment program. Design features that have not previously been tested in the reference plants or subsequent testing are tested to verify the vibration analysis. Conformance with Regulatory Guide 1.20 is summarized in Section 1.9.1.

The program is directed toward confirming the long-term, steady-state vibration response of the reactor internals for operating conditions. The three aspects of this evaluation are the following: a prediction of the vibrations of the reactor internals, a preoperational vibration test program of the internals of the first plant, and a correlation of the analysis and test results.

With respect to the reactor internals preoperational test program, the first AP600 plant reactor vessel internals are classified as prototype as defined in Regulatory Guide 1.20. The AP600 reactor vessel internals do not represent a first-of-a-kind or unique design based on the arrangement, design, size, or operating conditions. The units referenced in the subsection 3.9.2.3 as supporting the AP600 reactor vessel internals design features and configuration have successfully completed vibration assessment programs including vibration measurement programs. These units have subsequently demonstrated extended satisfactory inservice operation.

The ~~prototype~~ reference plant for the AP600 is H. B. Robinson that has substantially the same size and operating conditions as the AP600. Structural differences include modifications resulting from the use of 17x17 fuel, the removal of the thermal shield and the change to the inverted top hat upper internals support assembly. These design changes were incorporated into the Doel 3 and Doel 4 reactor internals as well as the AP600. OI 3.9.3.3-2

The effects of these design evolutions from the reference plant were shown by instrumented preoperational testing at the Doel 3 (upper internals) and Doel 4 (lower internals) plants. The vibrational responses of the AP600 reactor internals are characterized by the Doel 3 and 4 vibration measurement programs.

The pre-operational test program of the first AP600 plant includes a limited vibration measurement program and a pre- and post-hot functional inspection program. This program satisfies the guidelines for a Regulatory Guide 1.20 Prototype Category plant. The AP600 reactor internals design does not require supplemental testing including component vibration

considering the maximum stresses for each condition and combining them with square root of the sum of the squares method.

The system seismic analysis of the reactor vessel and its internals is either performed by a response spectrum analysis method or by a time-history integration method. Both of these analysis techniques are consistent with guidelines in the Standard Review Plan.

For certain systems or components, when time dependent seismic response is desired, the nonlinear time history analysis is used. The seismic time-history analysis technique is essentially the same as that for the pipe rupture analysis, except that in seismic analysis time history accelerations are used as the forcing function. The seismic response is combined with the pipe rupture response, as outlined in subsection 3.9.3, in order to obtain the maximum stresses and deflections.

Reactor internals components are within acceptable stress and deflection limits for the postulated pipe rupture combined with the safe shutdown earthquake condition.

3.9.2.5.3 Control Rod Insertion

During full power plant operation, rod cluster control assemblies and the corresponding drive rod assemblies are held at a fully withdrawn position by their respective control rod drive mechanisms. During certain accident conditions, such as small break loss of coolant accident or a safe shutdown earthquake condition or both, control assemblies are assumed to drop to their fully inserted position. The guide tubes are evaluated to demonstrate the function of the control rods for a break size of 144 inches and smaller.

No credit for the function of the control rods is assumed for large breaks in the safety analyses outlined in Chapter 15. However, for break sizes consistent with use of the leak-before-break criteria, the design of the guide tubes permits control rod insertion at each control rod position.

3.9.2.6 Correlation of Reactor Internals Vibration Tests with the Analytical Results

The results of dynamic analysis of reactor internals have been compared to the results of preoperational testing in reference plants. This comparison verifies that the analytical model used provides appropriate results.

OI 3.9.5-1

The damping for fuel assemblies of 20% for the evaluation of the response during a safe shutdown earthquake is for the fundamental mode. This damping value is supported by test results presented in WCAP-8236 (Reference 21).

The preoperational vibration test program for the reactor vessel internals of the AP600 conducted on the first plant, conforms to the intent of the guidelines in Regulatory Guide 1.20 for a comprehensive vibration assessment program. This program includes a correlation of the analysis and test results. This comparison provides additional verification for the analytical model.

Use of baseplates with concrete expansion anchors is minimized in the AP600. Concrete expansion anchors may be used for pipe supports. For these pipe support baseplate designs, the baseplate flexibility requirements of IE Bulletin 79-02, Revision 2, dated November 8, 1979 are met by accounting for the baseplate flexibility in the calculation of anchor bolt loads. Supplemental requirements for fastening anchor bolts to concrete are outlined in subsection 3.8.4.5.1. OI 3.9.3.3-2

Friction forces induced by the pipe on the support must be considered in the analysis of sliding type supports, such as guides or box supports, when the resultant unrestrained thermal motion is greater than 1/16 inch. The friction force is equal to the coefficient of friction times the pipe load, and acts in the direction of pipe movement. A coefficient of friction of 0.35 for steel-on-steel sliding surfaces shall be used. If a self-lubricated bearing plate is used, a 0.15 coefficient of friction shall be used. The pipe load from which the friction force is developed includes only deadweight and thermal loads. The friction force can not be greater than the product of the pipe movement and the stiffness of the pipe support in the direction of movement.

Small gaps are provided for frame type supports built around the pipe. These gaps allow for radial thermal expansion of the pipe as well as allowing for pipe rotation. The minimum gap (total of opposing sides) between the pipe and the support is equal to the diametral expansion of the pipe due to temperature and pressure. The maximum gap is equal to the diametral expansion of the pipe due to temperature and pressure plus 1/8 inch.

For standard component pipe supports, the manufacturer's functional limitations for example, travel limits and sway angles, should be followed. This criterion is applicable to limit stops, snubbers, rods, hangers and sway struts. Snubber settings should be chosen such that pipe movement occurs over the mid range of the snubber travel. Some margin should be provided between the expected pipe movement and the maximum or minimum snubber-stroke to accommodate construction tolerance.

3.9.3.4.1 ASME Code Class 1 Component Supports

The load combinations and allowable stresses for ASME Code Class 1 component supports are given in Tables 3.9-8 and 3.9-9.

3.9.3.4.1.1 Class 1 Component Supports Models and Methods

The static and dynamic structural analyses employ the matrix method and normal mode theory for the solution of lumped-parameter, multimass structural models. The equipment support structure models are dual-purpose, since they represent quantitatively the elastic restraints that the supports impose upon the component, and represent the individual support member stresses due to the forces imposed upon the supports by the component.

A description of the supports for the reactor pressure vessel, steam generator, and pressurizer is found in subsection 5.4.10. The supports are modeled using elements such as beams, plates, and springs where applicable.

The instrument columns ~~tubes~~ housing the in-core detector provide a protective path for the detectors during installation, reactor operation, and removal at refueling outages.

The guide tube assemblies sheath and guide the control rod drive shafts and control rods. The guide tubes are fastened to the upper support and are restrained by pins in the upper core plate for proper orientation and support.

The upper core support assembly is positioned in its proper orientation, with respect to the lower core support assembly, by flat-sided pins in the core barrel flange. Four equally spaced flat-sided pins are located at an elevation in the core barrel where the upper core plate is positioned. Four mating sets of inserts are located in the upper core plate at the same positions. As the upper support assembly is lowered into the lower support assembly, the inserts engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design.

Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. This system of locating pins and guidance arrangement provides proper alignment of the lower core support assembly, the upper core support assembly, the fuel assemblies, and control rods.

The upper and lower core support assemblies are preloaded by a large circumferential spring, which rests between the upper barrel flange and the upper core support assembly. This spring is compressed by installation of the reactor vessel head.

Vertical loads from weight, earthquake acceleration, hydraulic loads, and fuel assembly preload are transmitted through the upper core plate via the support columns, to the upper support, and then into the reactor vessel head. Transverse loads from coolant cross-flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the upper support and upper core plate. The upper support plate is particularly stiff to minimize deflection.

3.9.5.1.3 Radial Reflector

The radial reflector is between the lower core barrel and core, surrounding the core and forming the core cavity. The reflector is manufactured of solid rings of stainless steel with holes bored vertically for water cooling. The stainless steel reflects fast neutrons back to the core regions. This results in lower neutron loss from the core and decreased fluence on the reactor pressure vessel. Each reflector ring is sized in height so that adjoining sections meet at a fuel grid elevation.

3.9.5.1.4 Reactor Internals Interface Arrangement

Figure 3.9-8 shows the arrangement of reactor internals components shown in Figures 3.9-5 and 3.9-6 and their relative position in the reactor vessel. As shown in the figure, the lower reactor internal (Figure 3.9-5) rests on the vessel ledge. The upper core support structure (Figure 3.9-6) also rests at the same location on the top of a large compression spring (hold

3.9.6 Inservice Testing of Pumps and Valves

Inservice testing of ASME Code, Section III, Class 1, 2, and 3 pumps and valves is performed in accordance with Section XI of the ASME Code and applicable addenda, as required by 10 CFR 50.55a(f), except where specific relief has been granted by the NRC in accordance with 10 CFR 50.55a(f). The Code includes requirements for leak tests and functional tests for active components.

The requirements for system pressure tests are defined in the ASME Code, Section XI, IWA-5000. These tests verify the pressure boundary integrity and are part of the inservice inspection program, not part of the inservice test program.

Testing requirements for components constructed to the ASME Code are in several parts of the ASME OM Code (Reference 2). The ASME OM Code used to develop the inservice testing plan for the AP600 Design Certification is the 1990 Edition. The edition and addenda to be used for the inservice testing program are administratively controlled by the Combined License applicant.

The specific ASME Code requirements for functional testing of pumps are found in the ASME~~/ANSI~~ OM Code, Subsection ITSB. The specific ASME Code requirements for functional testing of valves are found in the ASME~~/ANSI~~ OM Code, Subsection ISTC. The functional tests are required for pumps and valves that have an active safety-related function. RAI 210.227

The AP600 inservice test plan does not include testing of pumps and valves in nonsafety-related systems unless they perform safety-related missions, such as containment isolation. This is based on the AP600 implementation of the regulatory treatment of nonsafety-related systems (RTNSS) process (WCAP-13856, Reference 15). Fluid systems with RTNSS important missions are shown to be available by operation of the system.

The AP600 inservice test plan includes periodic systems level tests and inspections that demonstrate the capability of safety-related features to perform their safety-related functions such as passing flow or transferring heat. The test and inspection frequency is once every 10 years. Staggering of the tests of redundant components is not required. These tests may be performed in conjunction with inservice tests conducted to exercise check valves or to perform power-operated valve operability tests. Alternate means of performing these tests and inspections that provide equivalent demonstration may be developed by the Combined License applicant in the inservice test program. Table 3.9-17 identifies the system inservice tests.

A preservice test program, which identifies the required functional testing, is to be submitted to the NRC by the Combined License applicant prior to performing the tests and following the start of construction. The inservice test program, which identifies requirements for functional testing, is to be submitted to the NRC prior to the anticipated date of commercial operation by Combined License applicant. Table 3.9-16 identifies the components subject to the preservice and the inservice test program. This table also identifies the method, extent, and frequency of preservice and inservice testing.

3.9.6.1 Inservice Testing of Pumps

Safety-related pumps are subject to operational readiness testing. The only safety-related mission performed by an AP600 pump is the coast down of the reactor coolant pumps. As a result, the AP600 inservice test plan does not include any pumps.

The AP600 inservice test plan does not include testing of pumps in nonsafety-related systems unless they perform safety-related missions. Systems containing pumps with RTNSS important missions have the capability during operation to measure the flow rate, the pump head, and pump vibration to confirm availability of the pumps. These measurements may be made with temporary instruments or test devices. The AP600 inservice test plan does not include testing of nonsafety-related pumps (including RTNSS important pumps) because they do not perform safety-related missions.

3.9.6.2 Inservice Testing of Valves

Safety-related valves are subject to operational readiness testing. Inservice testing of valves assesses operational readiness including actuating and position indicating systems. The valves that are subject to inservice testing include those valves that perform a specific function in shutting down the reactor to a safe shutdown condition, in maintaining a safe shutdown condition, or in mitigating the consequences of an accident. The AP600 safe shutdown condition includes conditions other than the cold shutdown mode. Safe shutdown conditions are discussed in subsection 7.4.1. In addition, pressure relief devices used for protecting systems or portions of systems that perform a function in shutting down the reactor to a safe shutdown condition, in maintaining a safe shutdown condition, or in mitigating the consequences of an accident, are subject to inservice testing.

~~The AP600 inservice test plan does not include testing of valves in nonsafety-related systems unless they perform safety-related missions. This testing may use temporary instruments or test devices.~~ The AP600 inservice test plan does not include testing of nonsafety-related valves (including RTNSS important valves) because they do not perform safety-related missions. Valves that are identified as having RTNSS important missions have provisions to allow testing ~~and~~ but are not included in the inservice test plan unless ~~if~~ inservice testing is identified as part of the regulatory oversight required for RTNSS. This testing may use temporary instruments or test devices. OITS 809

The valve test program is controlled administratively by the Combined License holder and is based on the plan outlined in this subsection. Valves (including relief valves) subject to inservice testing in accordance with ~~Section XI of~~ the ASME Code are indicated in Table 3.9-16. This table includes the type of testing to be performed and the frequency at which the testing should be performed. The test program conforms to the requirements of ASME OM, Subsection ISTC, to the extent practical. The guidance in NRC Generic Letters, AEOD reports, and industry and utility guidelines (including NRC Generic Letter 89-04) is also considered in developing the test program. Inservice testing incorporates the use of nonintrusive techniques to periodically assess degradation and performance of selected valves.

3.9.8 Combined License Information

3.9.8.1 Reactor Internals Vibration Response

Information including predicted vibration response and allowable response will be provided prior to the preoperational vibration testing of the first AP600 consistent with the guidance of Regulatory Guide 1.20.

3.9.8.2 Design Specifications and Reports

Combined License applicants referencing the AP600 design will have available for NRC audit the design specifications and design reports prepared for ASME Section III components.

3.9.8.3 Snubber Operability Testing

Combined License applicants referencing the AP600 design will develop a program to verify operability of essential snubbers as outline in subsection 3.9.3.4.3.

3.9.8.4 Nonintrusive Valve Inservice Testing

Combined License applicants referencing the AP600 design will develop an inservice test program in conformance with the valve inservice test requirements outlined in subsection 3.9.6 and Table 3.9-16. This program will include provisions for nonintrusive check valve testing methods and the program for valve disassembly and inspection outlined in subsection 3.9.6.2.3. OITS 801

3.9.8.5 Feedwater Line Thermal Monitoring

A monitoring program will be implemented by the Combined License holder at the first AP600 to record temperature distributions and thermal displacements of the feedwater line piping as outlined in subsection 3.9.3.1.2.

14. NRC BULLETIN NO. 88-11: Pressurizer Surge Line Thermal Stratification, December 20, 1988.

15. "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," WCAP-13856 September 1993.

16. NRC IE Bulletin 79-13, "Cracking in Feedwater System Piping," June 25, 1979 and Revisions 1 and 2, dated August 30, 1979 and November 16, 1979.

17. "Investigation of Feedwater Line Cracking in Pressurized Water Reactor Plants," (Proprietary) WCAP-9693 June 1980,

18. "AP600 Reactor Internals Flow-Induced Vibration Assessment Program," WCAP-14761, March, 1996.

19. "Functional Capability Criteria for Essential Mark II Piping," General Electric Company, NEDO-21985, 78NED174, E. C. Rodabaugh, September, 1978.

20. "Functional Capability of ASME Class 2/3 Stainless Steel Bends and Elbows," ASME 83- PVP-66, T. H. Liu, E. R. Johnson, K. C Chang.

21. "Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident," WCAP-8236 (Proprietary), WCAP-8238 (non-proprietary) OI 3.9.5-1

- The portion of the feedwater system to be constructed in accordance with ASME Code, Section III, Class 2 requirements is provided with access to welds and removable insulation for inservice inspection, in accordance with ASME Code, Section XI. The portion of the feedwater system to be constructed in accordance with ASME Code, Section III, Class 3 requirements is also designed and configured to accommodate inservice inspection in accordance with ASME Code, Section XI.

- The condensate and feedwater system classification is described in Section 3.2. The control functions and power supplies are described in Chapters 7 and 8.

- For a main feedwater or main steam line break (MSLB) inside the containment, the condensate and feedwater system is designed to limit high energy fluid to the broken loop. High energy line break for piping not qualified for leak before break (LBB) criteria is discussed in subsection 3.6.3.

- Double valve main feedwater isolation is provided via the main feedwater control valve (MFCV) and main feedwater isolation valve (MFIV). Valves fail closed on loss of actuating fluid. Both valves are designed to close automatically on main feedwater isolation signals, an appropriate engineered safety features (ESF) isolation signal, within the time established within the Technical Specification, Section 16.1.

RAI 210.228

- ~~A check valve, which acts on reverse pressure differential, is provided in the main feedwater line to each steam generator between the MFIV and the containment penetration. The check valve is designed to withstand the forces encountered when closing after a main feedwater line rupture. The valves prevent blowdown from more than one steam generator during feedline break while the appropriate ESF signal is generated to isolate using the MFIV and MFCV. During normal or upset conditions, the function of these check valves is to prevent reverse flow from the steam generators whenever the feedwater system is not in operation.~~

- The MFCVs provide backup isolation to their respective containment isolation valves in order to terminate feedwater flow. The MFCVs are located in the auxiliary building in piping designed to ASME Code, Section III, Class 3 seismic Category I requirements. These valves are components of the steam generator system (SGS).

- For a steam generator tube rupture event, positive and redundant isolation is provided for the main feedwater system (MFIV and MFCV) with ESF isolation signals generated by the protection and safety monitoring system.

10.4.7.1.2 Power Generation Design Basis

- The condensate and feedwater system provides a continuous feedwater supply to the two steam generators at the required pressures and temperatures for steady-state and anticipated transient conditions.

- Plant operation is possible at 100-percent power with one condensate pump out of service, and at 70-percent power with one booster/main feedwater pump assembly out of service.

- Plant operation is possible at greater than 70-percent power with one feedwater heater string out of service.

- The feedwater and condensate pumps and pump control system are designed so that loss of one booster/main feedwater pump assembly or one condensate pump does not result in trip of the turbine-generator or reactor.

- The pumps and other system components are designed so that the condensate, feedwater booster and feedwater pumps are protected from running with very low net positive suction heads without tripping on short transient low levels in a hotwell or deaerator tank.

- The condenser hotwell is designed to store, at the normal operating water level, an amount of condensate equivalent to at least three minutes of full-load condensate system operating flow.

- The system is able to accommodate ten-percent step or five-percent per minute ramp load changes without significant deviation from the programmed water levels in the steam generators or major effect on the feedwater system.

- The system has the capability of accommodating the necessary changes in feedwater flow to the steam generators with the steam pressure increase resulting from a 100-percent load rejection.

- The booster/main feedwater pumps are tripped simultaneously with the feedwater isolation signal to close the main feedwater isolation valves. In addition, the same isolation signal closes the isolation valve in the cross connect line between the main feedwater pump discharge header and the startup feedwater pump discharge header.

RAI 210.228

- A check valve, which acts on reverse pressure differential, is provided in the main feedwater line to each steam generator between the MFIV and the containment penetration. The check valve is designed to withstand the forces encountered when closing after a main feedwater line rupture. The valves perform no safety-related function but will serve to prevent blowdown from more than one steam generator during feedline break while the appropriate engineered safety features signal is generated to isolate using the MFIV and MFCV. During normal or upset conditions, the function of these check valves is to prevent reverse flow from the steam generators whenever the feedwater system is not in operation.

Criteria Section	Referenced Criteria	AP600 Position	Comments/Summary of Exceptions
B.1.c.(1)(a)		Acceptable	
B.1.c.(1)(b)		Acceptable	
B.1.c.(1)(c)		Acceptable	
B.1.c.(2)(a)		Acceptable	
B.1.c.(2)(b)	ASME Code, NC-3653	Acceptable	
B.1.c.(3)		Exception	The locations of postulated breaks in non-ASME pipe are based on ANSI/ANS-58.2-1988, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture", October, 1988. Stress value is calculated using equations in Section 104.8 of ANSI/ASME B31.1, Power Piping Code considering normal and upset plant conditions and compared to 0.8 (X+Y) where X and Y are the allowable stress values for Equations 12 and 13 of B31.1. OBE is not used to determine pipe break locations in the AP600.
B.1.c.(4)		~~Exception~~ ACCEPTABLE	~~When leak-before-break is not applied, separating structures are designed for postulated terminal end and high stress break locations.~~ OI 3.6.2.3-5
B.1.c.(5)		Acceptable	
B.1.d.		Acceptable	
B.1.e.(1)	ASME Code NB-3653	Acceptable	

Criteria Section	Referenced Criteria	AP600 Position	Comments/Summary of Exceptions
2.	ASME XI Section IWV	Exception	The IWV Section of the ASME Section XI has been replaced by ANSI/ASME-OM Part 10. The AP600 valve test program will meet the requirements of OM-10 and incorporate appropriate requirements from NRC Generic Letter 89-10. Generic Letter 89-04 will also be reviewed for applicable guidance. See SSAR Table 3.9-16 for a description of AP600 Inservice Test Requirements.
3.	ASME XI	Acceptable	

OI 3.10-1

SRP § 3.10 - Seismic and Dynamic Qualification of Mechanical and Electrical Equipment (Rev. 2, 7/81)

Criteria Section	Referenced Criteria	AP600 Position	Comments/Summary of Exceptions
1.	IEEE 344-1975 R.G. 1.100, REV. 2 ~~10 CFR Part 100~~ ~~GDC 1, 2, 4~~ REGULATORY GUIDE 1.100, REVISION 2, DATED JUNE 1988 ACCEPTS USE OF IEEE 344-1987	Exception	SRP 3.10 ~~ENTER~~ REFERENCES REG. GUIDE 1.100, REV. 1 WHICH ENDORSES IEEE 344-1975. The AP600 references qualification standards IEEE 323-1974 and IEEE 344-1987. As noted in IEEE 344-1987, safety related equipment may be qualified based on new testing and/or analysis or based on properly documented past test and experience data (Section 9.0 of IEEE 344-1987). The concept of using properly documented experience data is cost effective as evidenced by its proposed use in the resolution of the A-46 problem (NUREG-1030). The choice of qualification method is based upon many factors including practicality, complexity of the equipment, economics, and availability of previous qualification and experience data. If experience data is used, the COL applicant will identify the specific equipment and include details of the methodology and the corresponding experience data for each piece of equipment. Structural integrity and pressure retaining capability will be demonstrated by analysis using appropriate design codes such as the codes issued by the American Institute of Steel Construction (AISC) and American Society of Mechanical Engineers Boiler and Pressure Vessel Code (Section III).

Table 3.9-16 (Shee

VALVE INSERVICE TEST

Valve Tag Number	Description(1)	Valve Type	Safety-Related Missions	Safe
CVS-PL-V100	Makeup Line Containment Isolation Relief	Check	Maintain Close Transfer Close Transfer Open	Active Containm Safety Se
CVS-PL-V136A	Demineralized Water System Isolation	Remote	Maintain Close Transfer Close	Active-to Remote I
CVS-PL-V136B	Demineralized Water System Isolation	Remote	Maintain Close Transfer Close	Active-to Remote I
DWS-PL-V244	Demineralized Water Supply Containment Isolation - Outside	Manual	Maintain Close	Containm Safety Se
DWS-PL-V245	Demineralized Water Supply Containment Isolation - Inside	Check	Maintain Close	Containm Safety Se
FPS-PL-V050	Fire Water Containment Supply Isolation	Manual	Maintain Close	Containm Safety Se
FPS-PL-V052	Fire Water Containment Supply Isolation -Inside	Check	Maintain Close	Containm Safety Se
PCS-PL-V001A	PCCWST Isolation	Remote	Maintain Open Transfer Open	Active-to Remote
PCS-PL-V001B	PCCWST Isolation	Remote	Maintain Open Transfer Open	Active-to Remote
PCS-PL-V002A	PCCWST Series Isolation	Remote	Maintain Open Transfer Open	Active Remote
PCS-PL-V002B	PCCWST Series Isolation	Remote	Maintain Open Transfer Open	Active Remote
PCS-PL-V005	PCCWST Supply to Fire Protection Service Isolation	Manual	Maintain Close Transfer Close	Active
PCS-PL-V014A	Post-72 Hour Water Source Isolation	Manual Check	Transfer Open	Active
PCS-PL-V023	PCS Recirculation Return Isolation	Manual	Maintain Close Transfer Close	Active
PSS-PL-V008	Containment Air Sample Containment Isolation IRC	Remote	Maintain Close Transfer Close	Active-to Containm Safety S Remote

REQUIREMENTS

Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
t Isolation Leakage	AC	Containment Isolation Leak Test/2 Years Check Exercise/Refueling Shutdown	23, 27
ailed ition	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly	
ailed ition	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly	
t Isolation Leakage	A	Containment Isolation Leak Test (See Notes)	27
t Isolation Leakage	AC	Containment Isolation Leak Test (See Notes)	27
t Isolation Leakage	A	Containment Isolation Leak Test (See Notes)	27
t Isolation Leakage	AC	Containment Isolation Leak Test (See Notes)	27
ailed sition	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly	
ailed sition	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly	
sition	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly	
sition	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly	
	B	Exercise Full Stroke/Quarterly	13
	B	Exercise Full Stroke/Quarterly ~~Refueling Shutdown~~ Check Exercise/Refueling	21
	B	Exercise Full Stroke/Quarterly	13
ailed t Isolation Leakage sition	A	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test (See Notes) Exercise Full Stroke/Quarterly	27

OITS 1730

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Table 3.9-16 (Shee

VALVE INSERVICE TEST

Valve Tag Number	Description(1)	Valve Type	Safety-Related Missions	Safe
PXS-PL-V016B	Core Makeup Tank B Discharge Check	Check	Maintain Open Transfer Open Transfer Close	Active Remote
PXS-PL-V017A	Core Makeup Tank A Discharge Check	Check	Maintain Open Transfer Open Transfer Close	Active Remote
PXS-PL-V017B	Core Makeup Tank B Discharge Check	Check	Maintain Open Transfer Open Transfer Close	Active Remote
PXS-PL-V022A	Accumulator A Pressure Relief	Relief	Maintain Close Transfer Open Transfer Close	Active
PXS-PL-V022B	Accumulator B Pressure Relief	Relief	Maintain Close Transfer Open Transfer Close	Active
PXS-PL-V027A	Accumulator A Discharge Isolation	Remote	Maintain Open	Remote
PXS-PL-V027B	Accumulator B Discharge Isolation	Remote	Maintain Open	Remote
PXS-PL-V028A	Accumulator A Discharge Check	Check	Maintain Close Transfer Open	Active RCS Pre Remote
PXS-PL-V028B	Accumulator B Discharge Check	Check	Maintain Close Transfer Open	Active RCS Pre Remote
PXS-PL-V029A	Accumulator A Discharge Check	Check	Maintain Close Transfer Open	Active RCS Pre Remote
PXS-PL-V029B	Accumulator B Discharge Check	Check	Maintain Close Transfer Open	Active RCS Pre Remote
PXS-PL-V042	Nitrogen Supply Containment Isolation ORC	Remote	Maintain Close Transfer Close	Active-to Containm Safety S Remote
PXS-PL-V043	Nitrogen Supply Containment Isolation IRC	Check	Maintain Close Transfer Close	Active Containm Safety S Remote

REQUIREMENTS

Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
sition	BC	Remote Position Indication, Exercise/2 Years Check Exercise/Refueling Shutdown	10
sition	BC	Remote Position Indication, Exercise/2 Years Check Exercise/Refueling Shutdown	10
sition	BC	Remote Position Indication, Exercise/2 Years Check Exercise/Refueling Shutdown	10
	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
sition	B	Remote Position Indication, Exercise/2 Years	
sition	B	Remote Position Indication, Exercise/2 Years	
re Boundary sition	BC	Remote Position Indication, Exercise/2 Years Check Exercise/Refueling Shutdown	9
re Boundary sition	BC	Remote Position Indication, Exercise/2 Years Check Exercise/Refueling Shutdown	9
re Boundary sition	BC	Remote Position Indication, Exercise/2 Years Check Exercise/Refueling Shutdown	9
re Boundary sition	BC	Remote Position Indication, Exercise/2 Years Check Exercise/Refueling Shutdown	9
ailed t Isolation Leakage ition	A	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test (See Notes) Exercise Full Stroke/Quarterly	27
t Isolation Leakage ition	AC	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test (See Notes) Check Exercise/Quarterly	27

Table 3.9-16 (Shee

VALVE INSERVICE TEST

Valve Tag Number	Description(1)	Valve Type	Safety-Related Missions	Safe
RCS-PL-V004A	Fourth Stage Automatic Depressurization System	Squib	Maintain Open Maintain Close Transfer Open	Active RCS Pres Remote P
RCS-PL-V004B	Fourth Stage Automatic Depressurization System	Squib	Maintain Open Maintain Close Transfer Open	Active RCS Pres Remote P
RCS-PL-V004C	Fourth Stage Automatic Depressurization System	Squib	Maintain Open Maintain Close Transfer Open	Active RCS Pres Remote P
RCS-PL-V004D	Fourth Stage Automatic Depressurization System	Squib	Maintain Open Maintain Close Transfer Open	Active RCS Pres Remote P
RCS-PL-V005A	Pressurizer Safety Valve	Relief	Maintain Close Transfer Open Transfer Close	Active RCS Pres Remote P
RCS-PL-V005B	Pressurizer Safety Valve	Relief	Maintain Close Transfer Open Transfer Close	Active RCS Pres Remote P
RCS-PL-V010A	Automatic Depressurization System Discharge Header A Vacuum Relief	Relief	Transfer Open	Active
RCS-PL-V010B	Automatic Depressurization System Discharge Header B Vacuum Relief	Relief	Transfer Open	Active
RCS-PL-V011A	First Stage Automatic Depressurization System Isolation	Remote	Maintain Open Maintain Close Transfer Open	Active RCS Pres Remote P
RCS-PL-V011B	First Stage Automatic Depressurization System Isolation	Remote	Maintain Open Maintain Close Transfer Open	Active RCS Pres Remote P
RCS-PL-V012A	Second Stage Automatic Depressurization System Isolation	Remote	Maintain Open Maintain Close Transfer Open	Active RCS Pres Remote P
RCS-PL-V012B	Second Stage Automatic Depressurization System Isolation	Remote	Maintain Open Maintain Close Transfer Open	Active RCS Pres Remote P
RCS-PL-V013A	Third Stage Automatic Depressurization System Isolation	Remote	Maintain Open Maintain Close Transfer Open	Active RCS Pres Remote P

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REQUIREMENTS

Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
ure Boundary sition	D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5
ure Boundary sition	D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5
ure Boundary sition	D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5
ure Boundary sition	D	Remote Position Indication, Alternate/2 Years Charge Test Fire/20% in 2 Years	5
ure Boundary sition	BC	Remote Position Indication, Alternate/2 Years Class 1 Relief Valve Tests/5 Years and 20% in 2 Years	7
ure Boundary sition	BC	Remote Position Indication, Alternate/2 Years Class 1 Relief Valve Tests/5 Years and 20% in 2 Years	7
	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
ure Boundary sition	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke (See Notes) Operability Test/10 Years	3
ure Boundary sition	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke (See Notes) Operability Test/10 Years	3
ure Boundary sition	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke (See Notes) Operability Test/10 Years	3
ure Boundary sition	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke (See Notes) Operability Test/10 Years	3
ure Boundary sition	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke (See Notes) Operability Test/10 Years	3

Table 3.9-16 (Shee

VALVE INSERVICE TEST

Valve Tag Number	Description(1)	Valve Type	Safety-Related Missions	Safet
RCS-PL-V013B	Third Stage Automatic Depressurization System Isolation	Remote	Maintain Open Maintain Close Transfer Open	Active RCS Pres Remote F
RCS-PL-V014A	Fourth Stage Automatic Depressurization System Isolation	Remote	Maintain Open	Remote F
RCS-PL-V014B	Fourth Stage Automatic Depressurization System Isolation	Remote	Maintain Open	Remote F
RCS-PL-V014C	Fourth Stage Automatic Depressurization System Isolation	Remote	Maintain Open	Remote F
RCS-PL-V014D	Fourth Stage Automatic Depressurization System Isolation	Remote	Maintain Open	Remote F
RCS-PL-V150A	Reactor Vessel Head Vent	Remote	Maintain Open Maintain Close Transfer Open	Active-to RCS Pres Remote F
RCS-PL-V150B	Reactor Vessel Head Vent	Remote	Maintain Open Maintain Close Transfer Open	Active-to RCS Pres Remote F
RCS-PL-V150C	Reactor Vessel Head Vent	Remote	Maintain Open Maintain Close Transfer Open	Active-to RCS Pres Remote F
RCS-PL-V150D	Reactor Vessel Head Vent	Remote	Maintain Open Maintain Close Transfer Open	Active-to RCS Pres Remote F
RCS-K03	Safety Valve Discharge Chamber Rupture Disk	Relief	Transfer Open	Active
RCS-K04	Safety Valve Discharge Chamber Rupture Disk	Relief	Transfer Open	Active
RNS-PL-V001A	RNS Hot Leg Suction Isolation - Inner	Remote	Maintain Close Transfer Close	Active RCS Pres Remote F
RNS-PL-V001B	RNS Hot Leg Suction Isolation - Inner	Remote	Maintain Close Transfer Close	Active RCS Pres Remote F
RNS-PL-V002A	RNS Hot Leg Suction and Containment Isolation - Outer	Remote	Maintain Close Transfer Close	Active RCS Pres Containm Remote F
RNS-PL-V002B	RNS Hot Leg Suction and Containment Isolation - Outer	Remote	Maintain Close Transfer Close	Active RCS Pres Containm Remote F

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EQUIREMENTS

unctions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes	
e Boundary tion	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke (See Notes) Operability Test/10 Years	3	
tion	B	Remote Position Indication, Exercise/2 Years		
tion	B	Remote Position Indication, Exercise/2 Years		
tion	B	Remote Position Indication, Exercise/2 Years		
tion	B	Remote Position Indication, Exercise/2 Years		
iled e Boundary tion	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Operability Test/10 Years	4	OITS 1731
iled e Boundary tion	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Operability Test/10 Years	4	11
iled e Boundary tion	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Operability Test/10 Years	4	11
iled e Boundary tion	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Operability Test/10 Years	4	11
	BC	Inspect and Replace/5 Years		
	BC	Inspect and Replace/5 Years		
e Boundary tion	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown	15	
e Boundary tion	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown	15	
e Boundary Isolation tion	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown	15, 16	
e Boundary Isolation tion	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown	15, 16	

Table 3.9-16 (Sheet

VALVE INSERVICE TEST

Valve Tag Number	Description(1)	Valve Type	Safety-Related Missions	Safet
RNS-PL-V003A	RCS Pressure Boundary Valve Thermal Relief	Check	Maintain Close Transfer Open Transfer Close	Active RCS Pres
RNS-PL-V003B	RCS Pressure Boundary Valve Thermal Relief	Check	Maintain Close Transfer Open Transfer Close	Active RCS Pres
RNS-PL-V011	RNS Discharge Containment Isolation Valve - ORC	Remote	Maintain Close Transfer Close	Active Containm Safety Se Remote F
RNS-PL-V013	RNS Discharge Containment Isolation - IRC	Check	Maintain Close Transfer Open Transfer Close	Active Containm Safety Se
RNS-PL-V015A	RNS Discharge RCS Pressure Boundary	Check	Maintain Close Transfer Close	Active RCS Pres
RNS-PL-V015B	RNS Discharge RCS Pressure Boundary	Check	Maintain Close Transfer Close	Active RCS Pres
RNS-PL-V017A	RNS Discharge RCS Pressure Boundary	Check	Maintain Close Transfer Open Transfer Close	Active RCS Pres
RNS-PL-V017B	RNS Discharge RCS Pressure Boundary	Check	Maintain Close Transfer Open Transfer Close	Active RCS Pres
RNS-PL-V021	RNS Hot Leg Suction Pressure Relief	Relief	Maintain Close Transfer Open Transfer Close	Active Containm Safety Se
RNS-PL-V022	RNS Suction Header Containment Isolation - ORC	Remote	Maintain Close Transfer Close	Active Containm Safety Se Remote F
RNS-PL-V023	RNS Suction from IRWST - Containment Isolation	Remote	Maintain Close Transfer Close	Active Containm Safety Se Remote F
RNS-PL-V045	RNS Pump Discharge Relief	Relief	Maintain Close Transfer Open Transfer Close	Active
RNS-PL-V046	RNS Heat Exchanger A Channel Head Drain Isolation	Manual	Maintain Open Transfer Open	Active

REQUIREMENTS

Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
re Boundary	BC	Check Exercise/Refueling Shutdown	23
re Boundary	BC	Check Exercise/Refueling Shutdown	23
t Isolation Leakage ition	A	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test (See Notes) Exercise Full Stroke/Quarterly	27
t Isolation Leakage	AC	Containment Isolation Leak Test (See Notes) Check Exercise/Quarterly	27
re Boundary	BC	Check Exercise/Refueling Shutdown	24
re Boundary	BC	Check Exercise/Refueling Shutdown	24
re Boundary	BC	Check Exercise/Refueling Shutdown	24
re Boundary	BC	Check Exercise/Refueling Shutdown	24
t Isolation Leakage	AC	Containment Isolation Leak Test/2 Years Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	17, 27
t Isolation Leakage ition	A	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test (See Notes) Exercise Full Stroke/Quarterly	27
t Isolation Leakage ition	A	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test (See Notes) Exercise Full Stroke/Quarterly	17, 27
	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
	B	Exercise Full Stroke/Quarterly ~~Refueling Shutdown~~	~~21~~

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ANSTEC APERTURE CARD

Also Available on Aperture Card

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Table 3.9-16 (Shee

VALVE INSERVICE TEST

Valve Tag Number	Description(1)	Valve Type	Safety-Related Missions	Safe
SGS-PL-V255B	Startup Feedwater Control	Remote	Maintain Close Transfer Close	Active-to Remote F
VES-PL-V002A	Pressure Regulating Valve A	Press. Reg.	Throttle Flow	Active
VES-PL-V002B	Pressure Regulating Valve B	Press. Reg.	Throttle Flow	Active
VES-PL-V005A	Air Delivery Isolation Valve A	Remote	Maintain Open Transfer Open	Active-to
VES-PL-V005B	Air Delivery Isolation Valve B	Remote	Maintain Open Transfer Open	Active-to
VES-PL-V008A	Refill Check Valve A	Check	Maintain Close Transfer Open Transfer Close	Active
VES-PL-V008B	Refill Check Valve B	Check	Maintain Close Transfer Open Transfer Close	Active
VES-PL-V022A	Pressure Relief Isolation Valve A	Remote	Maintain Open Transfer Open	Active-to
VES-PL-V022B	Pressure Relief Isolation Valve B	Remote	Maintain Open Transfer Open	Active-to
VES-PL-V040A	Air Tank Safety Relief Valve A	Relief	Maintain Close Transfer Open	Active
VES-PL-V040B	Air Tank Safety Relief Valve B	Relief	Maintain Close Transfer Open	Active
VES-PL-V041A	Air Tank Safety Relief Valve A	Relief	Maintain Close Transfer Open	Active
VES-PL-V041B	Air Tank Safety Relief Valve B	Relief	Maintain Close Transfer Open	Active
VES-PL-V042	Refill Line Vent Isolation Valve	Manual	Maintain Close Transfer Close	Active

REQUIREMENTS

Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
Failed Position	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test/10 Years	
	B	Exercise Full Stroke/Quarterly Operability Test/10 Years	
	B	Exercise Full Stroke/Quarterly Operability Test/10 Years	
Failed	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test/10 Years	
Failed	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test/10 Years	
	BC	Check Exercise/Refueling Shutdown	21
	BC	Check Exercise/Refueling Shutdown	21
Failed	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test/10 Years	
Failed	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test/10 Years	
	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
	B	Exercise Full Stroke/Quarterly	

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.

3. ADS stage 1/2/3 valves (RCS-V001A/B, V002A/B, V003A/B, V011A/B, V012A/B, V013A/B) will be full stroke tested every 6 months on a staggered basis. This is a relief request from the ASME code. Exercise testing of these valves represents a risk of loss of reactor coolant and depressurization of the RCS if the proper test sequence is not followed. For this reason, the frequency of this valve exercise testing should be minimized. Conversely, the PRA assumes that valve reliability for these valves is a function of test frequency. The recommended frequency has been incorporated into the AP600 PRA. The PRA results show that the AP600 meets its safety goals and that the results are not overly dependent on the ADS stage 1/2/3 valves.

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 reactor coolant pressure boundary isolation valves (CVS-V001, V002, 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). 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 ~~from the accumulators~~ 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 in not available to provide a detectable obtuator movement. ~~These valves are not exercised during cold shutdowns. Prior to performing such a test the accumulator pressure must be significantly reduced, control power must be restored to the closed accumulator motor operated valve and the valve must be opened. Following the test the accumulator pressure must be increased, the accumulator water level increased, and the motor operated valve closed and its power removed. In addition, this test increases the chance of significantly perturbating the pressurizer level or lifting the RNS relief valve if the test is not conducted properly. Due to the complexity of the test, exercising these valves during cold shutdowns is not considered practical.~~ Full stroke ~~E~~exercise testing of these valves is conducted during a refueling shutdowns. ~~outage when it is practical.~~

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.

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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 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 squib valves are in line with the check valve which blocks the flow path for water that would be used in the test.~~ 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.

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. PCS stop check valves (PCS-V005, V023) have inservice testing on their stop function. Their check valve function does not provide active functions and is not subject to inservice testing.

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

17. Normal residual heat removal system containment penetration relief valve (RNS-V021) and containment isolation motor-operated valve (RNS-V023) are subjected to containment leak testing by pressurizing the lines in the reverse direction

to the flow which accompanies a containment leak in this path.

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 partially stroked on a quarterly basis and 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 verifies the valves operability status and that the valve is not mechanically bound.

21. Post-72 hour ~~manual or~~ 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-V014A 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 valve (SGS-V250A/B). The valves are not quarterly stroke tested since full stroke testing would result in a plant transient during power operation. Normal feedwater 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 (WLS-V071A/B/C, 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.

In the incontainment refueling water storage tank injection lines, the squib valves are in series with normally closed check valves. In the containment recirculation lines, the squib valves are in series with normally closed check valves in two lines and with normally closed motor operated valves in the other two lines. As a result, inadvertent opening of these squib valves will not result in loss of reactor coolant or in draining of the incontainment refueling water storage tank.

The type of squib valve used in these applications provides zero leakage in both directions. It also allows flow in both directions. A valve open position sensor is provided for these valves.

Squib valves are also used to isolate the fourth stage automatic depressurization system lines. These squib valves are in series with normally open motor operated gate valves. Redundant-series controllers are provided to prevent spuriously opening of these squib valves. The type of squib valve used in this application provides zero leakage of reactor coolant out of the reactor coolant system. The reactor coolant pressure acts to open the valve. A valve open position sensor is provided for these valves.

6.3.2.3 Applicable Codes and Classifications

Sections 5.2 and 3.2 list the equipment ASME Code and seismic classification for the passive core cooling system. Most of the piping and components of the passive core cooling system within containment are AP600 Equipment Class A, B, or C and are designed to meet seismic Category I requirements. Components and piping that provides an emergency core cooling function that is Equipment Class C has augmented weld inspection requirements, see subsection 3.2.2.5. Some system piping and components that do not perform safety-related functions are nonsafety-related.

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The requirements for the control, actuation, and Class 1E devices are presented in Chapters 7 and 8.

6.3.2.4 Material Specifications and Compatibility

Materials used for engineered safety feature components are given in Section 6.1. Materials for passive core cooling system components are selected to the meet the applicable material requirements of the codes in Section 5.2, as well as the following additional requirements:

- Parts of components in contact with borated water are fabricated of, or clad with, austenitic stainless steel or an equivalent corrosion-resistant material.

- Internal parts of components in contact with containment emergency sump solution during recirculation are fabricated of austenitic stainless steel or an equivalent corrosion resistant material.

- Valve seating surfaces are hard-faced to prevent failure and to reduce wear.

Table 6.2.3-1 (

Containment Mechanical Pen

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 SSAR 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
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 steamline isolation
LSL:	Low steamline 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

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 containment. These systems are not part of the reactor coolant pressure boundary and do not open directly to t generators is vented to the atmosphere outside containment to ensure that full test differential pressure is applie

3. The central chilled water system remains water-filled and operational during the Type A test in order to mainta

4. The containment isolation valves for this penetration are open during the Type A test to facilitate testing. Thei

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 ve ~~post accident operation.~~

7. The inboard globe valve is tested in the reverse direction. The test is conservative since the test pressure tends

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trations and Isolation Valves

Closure Time:
Required valve closure stroke time
- std: Industry standard for valve type
- 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

ampling piping from the steam generators to the containment penetration, is considered an extension of the e containment atmosphere during post-accident conditions. During Type A tests, the secondary side of the steam l to this boundary.

n stable containment atmospheric conditions.

leak rates are measured separately.

ted during local leak rate test ~~LLRT~~ to retain double isolation of RCS at elevated pressure. Valve is flooded during

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o unseat the valve disc, whereas containment pressure would tend to seat the disc.

Table 6.2.3-1

Containment Mechanical Pen

System	Containment Penetration			Valve/Hatch Identification	SSAR Subsectio
	Line	Flow	Closed Sys IRC		
CAS	Service air in	In	No	CAS-PL-V204 CAS-PL-V205	9.3.1
	Instrument air in	In	No	CAS-PL-V014 CAS-PLV015	9.3.1
CCS	IRC loads in	In	No	CCS-PL-V200 CCS-PLV201	9.2.2
	IRC loads out	Out	No	CCS-PLV208 CCS-PLV207	9.2.2
CVS	Spent resin flush out	Out	No	CVS-PL-V041 CVS-PL-V040 CVS-PL-V042	9.3.6
	Letdown	Out	No	CVS-PL-V047 CVS-PL-V045	9.3.6
	Charging	In	No	CVS-PL-V090 CVS-PL-V091 CVS-PL-V100 ~~V089~~	9.3.6
	H_2 injection to RCS	In	No	CVS-PL-V092 CVS-PL-V094	9.3.6
DWS	Demin. water supply	In	No	DWS-PL-V244 DWS-PL-V245	9.2.4
FHS	Fuel transfer	N/A	No	FHS-FT-01 ~~PL-V001~~	6.2.5
FPS	Fire protection standpipe sys.	In	No	FPS-PL-V050 FPS-PL-V052	9.5.1
PSS	RCS/PSX/CVS samples out	Out	No	PSS-PL-V011 PSS-PL-V010A,B	9.3.3
	Cont. air samples out	Out	No	PSS-PL-V046 PSS-PL-V008	9.3.3
	RCS/Cont. air sample return	In	No	PSS-PL-V023 PSS-PL-V024	9.3.3

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tions and Isolation Valves

Isolation Device			Test		
Position N-S-A	Signal	Closure Times	Type[1] & Note	Medium	Direction
C-O-C C-O-C	None None	N/A N/A	C,5	Air	Forward
O-O-C O-O-C	T None	std. N/A	C,5	Air	Forward
O-O-C O-O-C	S None	std. N/A	C,5	Air	Forward
O-O-C O-O-C	S S	std. std.	C,5	Air	Forward
C-C-C C-C-C C-C-C	None None None	N/A N/A N/A	C	Air	Forward
C-O-C C-O-C	T T	std. std.	C	Air	Forward
C-O-C C-O-C C-C-C	HR,PL2, S+PL1, SGL HR,PL2, S+PL1, SGL None	std. std. N/A	C	Air	Forward
C-C-C C-C-C	T None	std. N/A	C	Air	Forward
C-O-C C-O-C	None None	N/A N/A	C,5	Air	Forward
C-O-C	None	N/A	B	Air	Forward
C-C-C C-C-C	None None	N/A N/A	C,5	Air	Forward
C-C-C C-C-C	T T	std. std.	C	Air	Forward
O-C-C O-C-C	T T	std. std.	C	Air	Forward
O-C-C O-C-C	T None	std. N/A	C	Air	Forward

Table 6.2.3-1

Containment Mechanical Pen

System	Containment Penetration				
	Line	Flow	Closed Sys IRC	Valve/Hatch Identification	SSAR Subsectio
PXS	N_2 to accumulators	In	No	PXS-PL-V042 PXS-PL-V043	6.3
RNS	RCS to RHR pump	Out	No	RNS-PL-V002A/B RNS-PL-V023 RNS-PL-V022 RNS-PL-V021 RNS-PL-V061 PXS-PL-V208A	5.4.7 5.4.7 5.4.7 5.4.7 5.4.7 6.3
	RHR pump to RCS	In	No	RNS-PL-V011 RNS-PL-V013	5.4.7
SFS	IRWST/Ref. cav. SFP pump discharge	In	No	SFS-PL-V038 SFS-PL-V037	9.1.3
	IRWST/Ref. cav. purif. out	Out	No	SFS-PL-V035 SFS-PL-V034	9.1.3
SGS	Main steamline 01	Out	Yes	SGS-PL-V040A SGS-PL-V027A SGS-PL-V030A,31A,32A SGS-PL-V036A SGS-PL-V240A	10.3
	Main steamline 02	Out	Yes	SGS-PL-V040B SGS-PL-V027B SGS-PL-V030B,31B,32B SGS-PL-V036B SGS-PL-V240B	10.3
	Main feedwater 01	In	Yes	SGS-PL-V057A	10.3
	Main feedwater 02	In	Yes	SGS-PL-V057B	10.3
	SG blowdown 01	Out	Yes	SGS-PL-V074A	10.3
	SG blowdown 02	Out	Yes	SGS-PL-V074B	10.3
	Startup feedwater 01	In	Yes	SGS-PL-V067A	10.3
	Startup feedwater 02	In	Yes	SGS-PL-V067B	10.3

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rations and Isolation Valves

Isolation Device			Test		
Position N-S-A	Signal	Closure Times	Type[1] & Note	Medium	Direction
O-O-C	T	std.	C	Air	Forward
C-C-C	None	N/A			
C-O-C	HR	std.	C,4,6	Air	–
C-O-C	HR	std.	C		Reverse
C-O-C	HR	std.	C,4		Forward
C-C-C	None	N/A	C		Reverse
C-O-C	T	std.	C		Forward
C-C-C	None	N/A	C		Forward
C-O-C	HR	std.	C,4	Air	Forward
C-O-C	None	N/A	C,4		
C-O-C	T	std.	C,5	Air	Forward
C-O-C	None	N/A			
C-O-C	T	std.	C,5	Air	Forward
C-O-C	T	std.			
O-C-C	MS	5 sec	A,2	N_2	Forward
O-O-C	LSL	std.			
C-C-C	None	N/A			
O-O-C	MS	std.			
C-C-C	MS	std.			
O-C-C	MS	5 sec	A,2	N_2	Forward
O-O-C	LSL	std.			
C-C-C	None	N/A			
O-O-C	MS	std.			
C-C-C	MS	std.			
O-C-C	MF	5 sec	A,2	H_20	Forward
O-C-C	MF	5 sec	A,2	H_20	Forward
O-O-C	PRHR	std.	A,2	H_20	Forward
O-O-C	PRHR	std.	A,2	H_20	Forward
C-O-C	LTC, SGL	std.	A,2	H_20	Forward
C-O-C	LTC, SGL	std.	A,2	H_2O	Forward

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