Consumers Power
POWERING MICHIGAN'S PROGRESS
Palisades Nuclear Plant: 27780 Blue Star Memorial Highway, Covert, MI 49043

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U. S. Nuclear Regulatory Commission
Document Control Desk
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DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT
TECHNICAL SPECIFICATION CHANGE REQUEST - REVISION OF ADMINISTRATIVE CONTROLS - RESOLUTION OF NRC COMMENTS

A request for a change to the Palisades Technical Specifications Administrative Controls was submitted on December 11, 1995, and supplemented on January 18, 1996. During NRC review of the proposed changes, several improvements and alterations were suggested. The proposed Technical Specifications have been revised to include these improvements and alterations.

Attachment 1 contains a revised list of proposed changes and a revised No Significant Hazards Analysis which address the changes proposed by our December 11, 1995 change request, by the January 18, 1996, supplement to that request, and by this letter. Attachment 2 contains a complete set of revised Technical Specifications pages. Attachment 3 contains Technical Specifications marked to show the differences (including those made by Amendment 171) between the revised pages of Attachment 1 and those proposed pages submitted in our December 11, 1995, and January 18, 1996, letters. It is requested that the Technical Specifications contained in the Facially Operating License DPR-20 for the Palisades Plant be changed accordingly.

A001 1/1

The following improvements and alterations have been made to the proposed Technical Specifications pages submitted earlier:

A) Pages 4-1, 4-3, and 4-6 have been updated to include those changes made by Amendment 171, which was approved on April 5, 1996. Since these items have already been changed in the current Technical Specifications by Amendment 171, they are not listed among the changes discussed in Attachment 1.

B) On page 4-14, the newly proposed SR for verification that fuel pool ventilation filter flow could be initiated from the control room was renumbered from Item 4 to Item 3.c. As it had been proposed, there was no frequency specified for that SR. This item is discussed under change 5.e in Attachment 1.

C) On page 4-22, sixth basis paragraph, a superscript has been added to direct the reader to references 5 and 6. In the initially proposed revision, references 5 and 6 were listed, but the basis text did not call them out. Since this item only makes an editorial clarification to the bases, it is not discussed in Attachment 1.

D) On page 6-6, paragraph 6.5.2 was revised to explicitly call out the Shutdown Cooling System as one of the subject systems. This item is discussed under change 8 in Attachment 1.

E) On page 6-8, paragraph 6.5.5, the date for Regulatory Guide 1.35, Revision 3, was corrected. This item is discussed under change 9 of Attachment 1.

F) On page 6-17, the second paragraph of the proposed Containment Leak Rate Testing Program was revised to more closely emulate the NEI model Technical Specifications suggested for implementation of 10 CFR 50, Appendix J, Option B. The revised paragraph also specifies the calculated P_a, as defined in Appendix J, rather than the containment design pressure which had been formerly proposed. This item is discussed under change 48 in Attachment 1.

G) On page 6-17, sub paragraph 6.5.14.b, the proposed air lock testing acceptance criteria has been revised to delete an explicit limit for the overall air lock leak rate. The existing Palisades license contains no explicit limitation on overall air lock leakage, so the overall containment limit of 0.60 L_a is the defacto limit. Since this limit is expressed in item 6.5.14.a, it is unnecessary to repeat it. This item is discussed under change 48 of Attachment 1.

H) On page 6-18, the Process Control Program has been included. In the current Technical Specifications requirements for the Process Control Program exist in two locations; a definition, which specifies the required content, in Section 1.2 (page 1-5) and limitations on the change process in Section 6.19 (page 6-35).

The December 11, 1995, and January 18, 1995, submittals for this change request had proposed moving the Process Control Program requirements to the Offsite Dose Calculation Manual (ODCM) and Administrative Procedures. Discussions with the NRC reviewer have indicated that such a change would not be acceptable. Therefore, the subject requirements have been retained in the Technical Specifications. The definition has been moved to the Administrative Controls Section emulating the movement of the ODCM definition. The existing reference to current Specification 6.10.2o has been changed to reference the Quality Program, CPC-2A. The requirements currently contained in Specification 6.10.2o have been relocated, by this change request, to CPC-2A. Therefore, the change relocating the Process Control Program requirements is no longer proposed. This item is discussed under change 1.c of Attachment 1.

l) On page 6-18, reference to the Process Control Program has been restored to the Radioactive Effluent Release Report, item 6.6.3. The reference had been deleted in the earlier submittal when it was proposed that the Process Control Program be relocated. Since this item is not a change to the current Technical Specifications, it is not discussed in Attachment 1.

In order to provide time for completion of the procedure changes associated with this Technical Specification change request, it is requested that full implementation of the associated amendment not be required until 60 days after approval.

SUMMARY OF COMMITMENTS

This letter establishes no new commitments and makes no revisions to existing commitments.

Thomas C. Bordine
Manager, Licensing

CC Administrator, Region III, USNRC
Project Manager, NRR, USNRC
NRC Resident Inspector - Palisades

Attachments

CONSUMERS POWER COMPANY

To the best of my knowledge, the contents of this Technical Specifications change request, which revises the Administrative Controls section to emulate STS and to allow use of 10 CFR 50, Appendix J, Option B testing, are truthful and complete.

By Thomas C. Bordine

Thomas C. Bordine
Manager, Licensing

Sworn and subscribed to before me this 3rd day of September 1996.

Alora M. Davis

Alora M. Davis, Notary Public
Berrien County, Michigan
(Acting in Van Buren County, Michigan)
My commission expires August 26, 1999

[SEAL]

ATTACHMENT 1

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

TECHNICAL SPECIFICATION CHANGE REQUEST
ADMINISTRATIVE CONTROLS
ADDITIONAL CHANGES

Revised List of Changes and NSH Analysis

25 Pages

The following list of proposed changes and No Significant Hazards Analysis applies to the composite of changes proposed by Consumers Power Company Technical Specifications (TS) change request of December 11, 1995; by the January 18, 1996, supplement to that change request, and by this letter.

The proposed changes are described below. Each change is classified as one of the following four categories:

ADMINISTRATIVE - A change which is editorial in nature, which only involves movement of requirements within the TS without affecting their technical content, or clarifies existing TS requirements. These changes are described generically in the No Significant Hazards Determination.

RELOCATED - A change which only moves requirements from the TS to the FSAR, to the Operating Requirements Manual, or to other documents controlled under 10 CFR parts 50.54 or 50.59. These changes are described generically in the No Significant Hazards Determination.

MORE RESTRICTIVE - A change which only adds new requirements, or which revised an existing requirement resulting in additional operational restriction. These changes are described generically in the No Significant Hazards Determination.

LESS RESTRICTIVE - A change which deletes any existing requirement, or which revises any existing requirement resulting in less operational restriction. These changes are described individually in the No Significant Hazards Determination.

The following abbreviations are used in the discussions below:

COLR	Core Operating Limits Report
ODCM	Offsite Dose Calculation Manual
RG	Regulatory Guide
SR	Surveillance Requirement
STS	Standard Technical Specifications (NUREG 1432)
TS	Technical Specifications

I. Changes Proposed:

General:

References, in several sections, to sub parts of the former 10 CFR 20 were replaced with the corresponding reference in the revised 10 CFR 20. These changes in Part 20 references are not discussed individually.

In addition, the specific titles of supervisory positions were replaced by the generic title to allow these titles to be changed without necessitating a change in these TS. Section 6.2.1a has been amended to require that the relationship between generic titles and plant specific titles will be included in the FSAR.

The entire Administrative Controls section was re-paginated to eliminate voids and unused pages. The proposed numbering scheme makes provision for features of the STS which are not yet included in the Palisades TS, a Fuel Oil Testing Program, a safety Functions Determinations Program, and a Pressure - Temperature Limits Report.

The list of modifying amendments, which appears at the bottom of each page, was rewritten to list those amendments which modified the subject matter appearing on each proposed page.

Because no requirements are being changed, these general changes are classified as Administrative.

1. Sub-sections 1.1, Operating Definitions, and 1.2, Miscellaneous Definitions have been combined as 1.0, Definitions.

   a. Definitions for Members of the Public, Process Control Program, Site Boundary, and Unrestricted Area (page 1-5) are deleted from the TS. Members of the Public, Site Boundary, and Unrestricted Area are terms defined in 10 CFR 20.1003. These definitions need not be restated in the TS. These definitions do not appear in STS.

      Where these defined terms appeared, in the balance of the TS, in capitalized text (to indicate a term defined in Section 1), they have been replaced with lower case text.

      Definitions do not comprise a requirement. The subject definitions are redundant to definitions provided in 10 CFR 20, therefore, the deletion of these definitions will have no effect on the interpretations of requirements included in TS. Because no requirements are being changed; deleted items are redundant to items contained in 10 CFR 20, Change 1.a is classified as Administrative.

b. The definition for Offsite Dose Calculation Manual (page 1-5) is moved to proposed Section 6.5.1 and reworded slightly to emulate STS. This change does not alter any TS requirement; it only moves a requirement within TS. Because no requirements are being changed, Change 1.b is classified as Administrative.

c. The definition for Process Control Program (page 1-5) is moved to proposed Section 6.5.15. This change does not alter any TS requirement; it only moves a requirement within TS. Because no requirements are being changed, Change 1.c is classified as Administrative.

d. The definition for Core Operating Limits Report (page 1-5), is arranged alphabetically with the remaining definitions from former Section 1.1. The reference to Specification 6.9.1, in the COLR definition, is revised to match the proposed numbering. Because no requirements are being changed, Change 1.d is classified as Administrative.

2. The requirement to shutdown the reactor following a Safety Limit violation is moved from Section 6.7.1.a (page 6-10) to Sections 2.1.1 and 2.2.1 (page 2-1), which formerly referenced Section 6.7. This change places the operating requirements directly into the Action Statement, as is done in STS, instead of incorporating it by reference. While this TS requirement is redundant to the noted Section of 10 CFR, it is retained in the TS to assure that operators are aware of this immediate action. Change 2 is classified as Administrative.

3. Section 3.17.4, Accident Monitoring Instrumentation (page 3-70), is revised to reflect the inclusion of an Accident Monitoring Report into the Administrative Controls section of TS, to emulate STS. Action 3.17.4.7 is revised to reference the reporting requirement of Specification 6.6.7, the proposed reporting requirement. The proposed reporting requirement is unchanged from that of the existing TS. Because no requirements are being changed, Change 3 is classified as Administrative.

4. Specification 4.0.5 (page 4-1) has been moved to the Administrative Controls section as Specification 6.5.7, emulating STS Specification 5.5.8. The wording for the proposed specification is taken

from STS. There is no change in requirements. The Basis for Section 4 has been reworded accordingly, deleting reference to, and explanation of, Section 4.0.5. References to Specification 4.0.5 in the balance of the TS have been revised to reference Specification 6.5.7. Because no requirements are being changed, Change 4 is classified as Administrative.

5. The requirements of Table 4.2.3 have been revised to incorporate a Ventilation Filter Testing Program, similar to that in STS, and to update the remaining requirements of existing Table 4.2.3.

   The more restrictive requirements proposed by changes 5.c, 5.d, and 5.e, are currently being maintained under administrative controls to provide assurance of ventilation system operability. The revision of Table 4.2.3 made in conjunction with revision of the Administrative Controls section provided a convenient opportunity to request an enhancement of the existing requirements.

   a. Ventilation filter testing requirements of Table 4.2.3 (page 4-14) have been replaced with a single requirement to "Perform required Control Room Ventilation and Fuel Storage Area filter testing in accordance with the Ventilation Filter Testing Program" emulating STS SR 3.7.11.2 and 3.7.14.2. The wording for the proposed program is taken from STS. Because no requirements are being changed, Change 5.a is classified as Administrative.

   b. Existing Item c.2 (proposed Item 2.a) was editorially revised to more closely agree with current plant terminology. Because no requirements are being changed, Change 5.b is classified as Administrative.

   c. Existing Item c.3 (proposed Item 2.b) was revised to increase the required differential pressure capability and to require control room pressure to be established with respect to the outside atmosphere and rather than the viewing gallery. Because a more restrictive requirement is being added, Change 5.c is classified as More Restrictive.

   d. The existing requirement to verify control room temperature is below 120°F each 12 hours, was revised to require that control room temperature is below 90°F. This change is proposed to assure that the Thermal Margin Monitor in the Reactor Protective System remains within its design temperature range. Because a more restrictive requirement is being added, Change 5.d is classified as More Restrictive.

e. An additional requirement was added to Table 4.2.3, proposed Item 2.c, which requires, each refueling cycle:

> *"Verifying that the Fuel Pool Ventilation System is OPERABLE by initiating flow through the HEPA filter and charcoal adsorbers from the control room."*

This Item is comparable to the verification that Control Room Ventilation system automatically switches to the emergency mode. The Fuel Pool Ventilation System is manually actuated. Because a new requirement is being added, Change 5.e is classified as More Restrictive.

6. The remaining text of Specifications 4.2 through 4.6 (pages 4-15a through 4-41) was repaginated to eliminate blank pages. The list of modifying amendments, which appears at the bottom of each page, was rewritten to list only those amendments which modified the subject matter appearing on each proposed page.

   Existing pages 4-25 and 4-26 contain a notation at the bottom of the page that they were altered by "Change No. 16" and by "Amendment No. 12". These are two identifiers for the same TS revision; "Change No. 16" to the TS was issued by "Amendment No. 12" to the Facility Operating License on February 25 1975. The references to "Change No. 16" have been omitted. Because no requirements are being changed, Change 6 is classified as Administrative.

7. Specification 4.3 (page 4-16) was revised to remove requirements which are redundant to parts of 10 CFR 50 or the ASME codes.

   a. Specification 4.3e, a requirement to reevaluate the Inservice Inspection Program (page 4-16), was deleted; it is redundant to the referenced section of 10 CFR 50 (50.55a (g)(5)), and is included in the Inservice Inspection and Testing Program. This change does not result in a change in requirements. Because no requirements are being changed; deleted items are redundant to items in 10 CFR 50, Change 7.a is classified as Administrative.

   b. The Specification 4.3f requirement to inspect the regenerative heat exchanger is redundant to the ASME Boiler and Pressure Vessel

Code, Section XI, Category B-A. The frequency in the code is once every 10 years; the existing requirement is once every 5 years. There is no unusual feature of the regenerative heat exchanger to require more frequent testing. Past testing at the 5 year interval has disclosed no problems. Because the required surveillance frequency is reduced, Change 7.b is classified as Less Restrictive.

c. The Specification 4.3f primary coolant pump testing requirements were moved to proposed Specification 6.5.6, Primary Coolant Pump Flywheel Surveillance Program, emulating STS Specification 5.5.7. There is no change in primary coolant pump flywheel testing requirements. Because no requirements are being changed, Change 7.c is classified as Administrative.

8. Section 4.5.3, Recirculation Heat Removal Systems (page 4-28a), which contains testing requirements for systems outside the containment which could potentially contain highly radioactive fluids, has been combined with former Section 6.15, Systems Integrity, as the proposed Section 6.5.2, Primary Coolant Sources Outside Containment. This change places all requirements for such testing in one location emulating the STS treatment of these testing requirements. This combining of these requirements was suggested by the NRC in the December 1, 1982, letter which issued Amendment 71 to the Palisades Operating License. The text placed in paragraph 6.5.2 was revised to explicitly call out the Shutdown Cooling System as one of the subject systems; the basis discussion (page 4-35) for Section 4.5.3 was deleted. There is no change in testing requirements. Because no requirements are being changed, Change 8 is classified as Administrative.

9. Specifications 4.5.4, Surveillance for Prestressing System (page 4-29); 4.5.5, End Anchorage Concrete Surveillance (page 4-32); and 4.5.8, Dome Delamination Surveillance (page 4-32a) were replaced by proposed Surveillance Requirement 4.5.4, which requires verification of containment structural integrity in accordance with the Containment Structural Integrity Surveillance Program, and Specification 6.5.5, the Containment Structural Integrity Surveillance Program. These proposed specifications emulate the STS treatment of containment structural integrity surveillance requirements. The proposed program requires compliance with Regulatory Guide 1.35, as does the equivalent STS program. That guide contains the details of the testing requirements. The parts of Specification 4.5 Basis and References pertaining to containment structural testing were deleted. Because no requirements are being changed, Change 9 is classified as Administrative.

10. Specification 4.5.6, Containment Isolation Valves (page 4-32), was renumbered 4.5.3. Because no requirements are being changed, Change 10 is classified as Administrative.

11. Surveillance requirement 4.6.2b, verification that the containment spray nozzles are open (page 4-39), was deleted; Item 4.6.2c was renumbered 4.6.2b. The requirement to inspect the containment spray nozzles is redundant to the ASME Boiler and Pressure Vessel Code, Section XI, Subparagraph ICW-5222(d) and Table ICW-2500-1. The frequency in the code is once every 10 years; the existing requirement is once every 5 years. There is no unusual feature of the Palisades spray nozzles to require more frequent testing. Past testing at the 5 year interval has disclosed no problems. Because the required surveillance frequency is reduced, Change 11 is classified as Less Restrictive.

12. The details of Specification 4.14, Augmented Inservice Inspection Program for Steam Generators (page 4-66), were moved to the Administrative Controls section of TS, emulating STS. There are no changes in requirements. Parts 4.14.2 through 4.14.5 were moved to proposed Specification 6.5.8, Steam Generator Tube Surveillance Program; part 4.14.6 was moved to proposed Specification 6.6.9, Steam Generator Tube Surveillance Report.

    Section 4.14 was retitled "Steam Generator Surveillance" and existing part 4.14.1, which currently requires the steam generators to be demonstrated operable by performance of the specified testing, was reworded to reference the proposed program. The wording used for surveillance requirement 4.14.1 was taken from STS. The reference to the Inservice Inspection and Testing Program (former 4.0.5) was retained.

    Table 4.14-1 was eliminated, because the pre-service and first service inspections have been completed and the requirements are no longer applicable. The Table 4.14-1 footnote has been reworded as the first paragraph of the Program.

    The wording of the existing testing requirements was revised to eliminate redundancies and to remove requirements pertinent only to preservice and initial testing. The wording of the reporting requirements was revised to utilize parallel sentence structure.

    The proposed testing program and reporting requirements are equivalent, with exception of preservice and initial testing which has been completed, to the requirements of existing Section 4.14.

Because no requirements are being changed, Change 12 is classified as Administrative.

13. The record retention requirement of Section 4.16.1f (page 4-74) was revised to eliminate reference to the deleted Section 6.10. Because no requirements are being changed, Change 13 is classified as Administrative.

14. A requirement for the plant superintendent to approve tests, experiments, and modifications was added to Section 6.1.1 (page 6-1). The proposed wording was taken from STS. Because a new requirement is being added, Change 14 is classified as More Restrictive.

15. The wording of Section 6.1.2 (page 6-1) is revised to match STS. The proposed requirements are nearly identical to the existing requirements. Because no requirements are being changed, Change 15 is classified as Administrative.

16. Section 6.2.2, Plant Staff (page 6-2), was extensively revised to emulate the STS. The table which provides shift staffing requirements does not appear in STS and was deleted. The requirements of the table which are not redundant to requirements of 10 CFR 50 have been included as separate requirements. These changes are administrative, and do not involve any change in requirements.

    a. Item 6.2.2a currently requires each shift on duty to include the staff required by Table 6.2-1. The table contains requirements for Shift Supervisor (SS), Shift Engineer (SE) or Senior Reactor Operator (SRO), Reactor Operator (RO), Auxiliary Operator (AO), and Shift Technical Advisor (STA) positions. The SS, SE or SRO, and RO requirements duplicate the requirements of 10 CFR 50.54 (m)(2)(i) and need not be repeated in TS; the AO requirement of that table appears as the proposed 6.2.2a; and the STA requirements appear as the proposed 6.2.2g. Because no requirements are being changed, Change 16.a is classified as Administrative.

    b. The wording of Item 6.2.2b is revised to match STS. The proposed requirements are nearly identical to the existing requirements. Because no requirements are being changed, Change 16.b is classified as Administrative.

    c. A new Item 6.2.2c has been added. Its wording is taken from STS, and is nearly equivalent to the existing footnote in Table 6.2-1 which allows the shift crew to be "one less than" the required crew during unexpected absences. The proposed Item 6.2.2c allows the crew to be "less than" required. Because the proposed allowance is less restrictive than the existing allowance, Change 16.c is classified as Less Restrictive.

d. Existing Item 6.2.2c was renumbered 6.2.2d and reworded to match STS. The proposed Item 6.2.2d effectively combines the existing Item 6.2.2c and the associated footnote which appears at the bottom of the page. The footnote is deleted. Because no requirements are being changed, Change 16.d is classified as Administrative.

e. Existing Item 6.2.2d, which requires core alterations to be performed under supervision of an SRO, is deleted. There is no equivalent requirement in STS. It is redundant to the requirements of 10 CFR 50.54 (m)(2)(iv). Because no requirements are being changed, Change 16.e is classified as Administrative.

f. Existing requirement 6.2.2f is renumbered as 6.2.2e and reworded, slightly, to more closely match STS. (There is no existing requirement numbered 6.2.2e.) There is no reduction, nor any significant change, in requirements. Because no requirements are being changed, Change 16.f is classified as Administrative.

g. A new Item, 6.2.2f, has been added requiring either the operations manager or his assistant to hold an SRO license. This requirement is moved to 6.2.2 from existing 6.3.5 to match the wording of STS. Because no requirements are being changed, Change 16.g is classified as Administrative.

h. A new Item, 6.2.2g, describing the requirements for a Shift Technical Advisor has been added. It replaces the STA requirements formerly listed in deleted Table 6.2-1. The proposed wording is a combination of that from STS and the footnote from Table 6.2-1. Because no requirements are being changed, Change 16.h is classified as Administrative.

17. Reference to formerly deleted Item 6.2.3 was removed from page 6-2a. Its retention served no function. Because no requirements are being changed, Change 17 is classified as Administrative.

18. Section 6.3, Plant Staff Qualifications (page 6-3), was revised. The proposed changes limit the requirements of this section to qualifications, allow assignment of the individuals who perform 50.59 reviews to other

departments within the plant staff, and move one Item to Section 6.2.2, emulating its placement in STS.

a. Item 6.3.1 is unchanged.

b. Item 6.3.2 has been revised to delete the requirements that the radiation safety manager be designated by the Plant General Manager, and that the radiation safety manager shall have direct access to the plant manager. Neither of these requirements is germane to a section on qualifications. In addition, the associated footnote has been incorporated into the main paragraph. Because requirements have been deleted, Change 18.b is classified as Less Restrictive.

c. Item 6.3.3 is unchanged.

d. Item 6.3.4 was reworded to allow the assignment of the required 50.59 reviews to other portions of the plant staff. The qualification requirement for these individuals who perform the reviews is unchanged. The reference to Item 6.5.3 was reworded because this change request proposes deleting that section from TS. Because the qualification requirements of the persons performing the subject reviews has not been changed, but only the reference to their assignment in the plant organization, Change 18.d is classified as Administrative.

e. Item 6.3.5 was deleted; its requirements appear elsewhere. The first requirement, for an operations manager to hold an SRO license, appears as proposed 6.2.2f, as it does in STS. The second requirement, for meeting ANSI N18-1, is redundant to 6.3.1. The third requirement is also moved to 6.2.2f. Because no requirements are being changed, Change 18.e is classified as Administrative.

19. As discussed under change 16, above, Table 6.2-1 (page 6-4) has been deleted. Those requirements of Table 6.2-1 which do not appear in 10 CFR 50.54 have been added to the text of Section 6.2.2. This change emulates STS. Because no requirements are being changed, Change 19 is classified as Administrative.

20. Section 6.4 (page 6-5), formerly deleted, was retitled "Procedures" emulating Section 5.4 of STS. The procedure requirements have been moved to Section 6.4 from Section 6.8.1. Change Because no requirements are being changed, 20 is classified as Administrative.

21. a. The requirements of Section 6.5, Review and Audit, (pages 6-5 through 6-10) were deleted from the TS entirely. These requirements have been relocated to the Quality Program Description, CPA-2A. There are no comparable review and audit requirements in STS. Because items have been moved from TS to a document controlled under 50.54, Change 21.a is classified as Relocated.

    b. Section 6.5 was retitled "Programs and Manuals", emulating Section 5.5 of STS and replacing existing Section 6.8.4. Programs and Manuals currently required by Section 6, and a newly proposed TS basis control program (Change 47) are gathered in this section. Because no requirements are being changed, Change 21.b is classified as Administrative.

22. As mentioned in change 2, above, Item 6.7.1a, the Safety Limit violation shutdown requirement, was moved to Sections 2.1.1 and 2.2.1 which formerly referenced Section 6.7. Item 6.7.1.b, 1 hour notification, and Items 6.7.1c and 6.7.1d, written reporting, (page 6-10) were deleted. Item 6.7.1.b is redundant to 10 CFR 50.72(b)(1)(i)(A); Items 6.7.1.c and 6.7.1.d are redundant to 10 CFR 50.36(c)(1)(i)(A) and to 10 CFR 50.73(a)(2)(i)(A), (i)(B), and (ii)(B).

    This change would extend the required time for submitting a Safety Limit violation report from 14 to 30 days. Since plant operation may not resume until authorized by the Commission, a slight extension of reporting time would not have any effect on safety. Because of the extension in required reporting time, Change 22 is classified as Less Restrictive.

23. Section 6.8.1 (page 6-11) was retitled "Procedures" and renumbered 6.4, emulating STS Section 5.4. Item 6.8.1a was reworded slightly to emulate STS wording. Items 6.8.1d and 6.8.1e, which require having written procedures for the security and emergency plans, were deleted in accordance with the recommendations of Generic Letter 93-07. There are no comparable requirements in STS. Parts 50 and 73 of Title 10 of the Code of Federal Regulations include provisions that are sufficient to address these requirements. Because no requirements are being changed; deleted items are redundant to items in 10 CFR parts 50 and 73, Change 23 is classified as Administrative.

24. Sections 6.8.2 and 6.8.3 (page 6-11), which describe requirements for the procedure change process, have been relocated to the Quality Program Description, CPC-2A. This level of detail does not appear in STS. Because

TS requirements have been deleted and replaced with requirements controlled under 50.54, Change 24 is classified as Relocated.

25. Item 6.8.4a, Radioactive Effluent Controls Program (page 6-12), was renumbered 6.5.4, emulating the STS Item 5.5.4 of the same title. Editorial revisions were made to make the wording closer to that in STS. Because no requirements are being changed, Change 25 is classified as Administrative.

26. Item 6.8.4b, Radiological Environmental Monitoring Program (page 6-13), was deleted. No comparable item exists in STS. The program is contained within the ODCM, as required by proposed Specification 6.5.1. Changes to the ODCM are required to be submitted to the NRC as part of, or concurrent with, the Radioactive Effluent Release Report as required by proposed Specifications 6.5.1 and 6.6.3. Because no requirements are being changed, Change 26 is classified as Administrative.

27. Item 6.9, Reporting Requirements (page 6-14) was renumbered 6.6, emulating STS Section 5.6 of the same title. Reporting requirements in existing TS Section 6 were collected in this section. Because no requirements are being changed, Change 27 is classified as Administrative.

28. Item 6.9.1a, Startup Report (page 6-14), was deleted. No comparable report is required by STS. The existing TS require a summary of plant start-up and power escalation testing shall be submitted within 90 days following completion of the start-up test program following "amendment to the license involving a planned increase in power level, installation of fuel that has a different design or has been manufactured by a different fuel supplier and, modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant".

    This reporting requirement has been judged as unnecessary for inclusion in the STS. It simply summarizes the complete records which are part of the permanent plant records and are thereby available for NRC review. The 90 days of operation allowed before the report submittal, and the lack of any required approval, imply that the report is not intended to be used for NRC safety decisions. Because a requirement is being deleted, Change 28 is classified as Less Restrictive.

29. Item 6.9.1b, Annual Report (page 6-14), was renumbered 6.6.1 and retitled "Occupational Radiation Exposure Report" emulating STS Item 5.6.1. The option to base radiation exposure reporting on electronic dosimeters was added. The wording was revised, editorially, to more closely match STS.

Because no requirements are being changed, Change 29 is classified as Administrative.

30. Item 6.9.1c, Monthly Operating Report (page 6-15), was renumbered 6.6.4 emulating STS Item 5.6.4. Because no requirements are being changed, Change 30 is classified as Administrative.

31. Item 6.9.1d, Radioactive Effluent Release Report (page 6-15), was renumbered 6.6.3 emulating STS Item 5.6.3. Because no requirements are being changed, Change 31 is classified as Administrative.

32. Item 6.9.1e, Radiological Environmental Operating Report (page 6-15), was renumbered 6.6.2 emulating STS Item 5.6.2. Because no requirements are being changed, Change 32 is classified as Administrative.

33. Item 6.9.1f, Core Operating Limits Report (page 6-15), was renumbered 6.6.5 emulating STS Item 5.6.5. Because no requirements are being changed, Change 33 is classified as Administrative.

34. Item 6.9.2, Reportable Events (page 6-17), was deleted. There is no comparable requirement in STS. It is redundant to 10 CFR 50.73. The existing requirement is:

    "The Commission shall be notified of Reportable Events and a report submitted pursuant to the requirements of 10 CFR 50.73."

    Because no requirements are being changed; the deleted item is redundant to 10 CFR 50.73, Change 34 is classified as Administrative.

35. Item 6.9.3, Nonroutine Reports (page 6-17), was deleted. There is no comparable requirement in STS. The existing requirement is:

    "A report shall be submitted in the event that (a) the Radiological Environmental Monitoring Programs are not substantially conducted as described in the ODCM or (b) an unusual or important event occurs from plant operation that causes a significant environmental impact or affects a potential environmental impact. Reports shall be submitted within 30 days."

    a. Part (a) of this requirement is not redundant to any other requirement, but plant administrative procedures require initiation of a Condition

Report for such an event. Condition Reports are available for NRC audit and are often reviewed by the NRC resident inspector. Because a requirement is being deleted, Change 35.a is classified as Less Restrictive.

b. Part (b) of the existing requirement is redundant to Item 4.1 of the Palisades Environmental Protection Plan, which is Appendix B to the Facility Operating License. That requirement is:

> *"Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to plant operation shall be recorded and promptly reported to the NRC within 24 hours by telephone, telegraph, or facsimile transmissions followed by a written report per Subsection 5.4.2."*

In addition, an event that which could or did result in a significant environmental impact would probably involve a notification of other governmental agencies and thereby are subject to the reporting requirements of 10 CFR 50.72(b)(2)(iv)(A&B). Because no requirements are being changed, Change 35.b is classified as Administrative.

36. Item 6.9.4a, under Special Reports (page 6-26), was reworded and renumbered 6.6.8 and retitled "Containment Structural Integrity Surveillance Report" emulating STS Item 5.6.9. Because no requirements are being changed, Change 36 is classified as Administrative.

37. Item 6.9.4.b (page 6-26) was deleted. There is no comparable requirement in STS. It is redundant to 10 CFR 50.4. The existing requirement is:

> *"Special reports shall be submitted in accordance with 10 CFR 50.4, within the time period specified for each report."*

Because no requirements are being changed; the deleted item being redundant to an item in 10 CFR 50, Change 37 is classified as Administrative.

38. The requirements of Section 6.10, Record Retention (page 6-26 through 6-28), were deleted from the TS entirely. These requirements are relocated to the Quality Program Description, CPC-2A. There are no comparable

Record Retention requirements in STS. Because TS requirements are deleted and replaced by requirements in a document controlled by 10 CFR 50.54, Change 38 is classified as Relocated.

39. Item 6.11, Radiation Protection Program (page 6-28), has been deleted. It is redundant to proposed Item 6.4a (existing 6.8.1a), in that it contains requirements for procedures for personnel radiation protection. Procedures for personnel radiation protection are listed in Regulatory Guide 1.33, and maintaining the procedures recommended by RG 1.33 is required by proposed Item 6.4a (existing 6.8.1a). The existing requirement is:

    "Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20, and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure."

    Because no requirements are being changed, Change 39 is classified as Administrative.

40. Section 6.12, High Radiation Area (page 6-28), was renumbered 6.7, emulating STS Section 5.7. The proposed section has been revised to use the wording from the CE Standard Technical Specifications, NUREG 1432, Revision 1 (STS). The major change in requirements due to this change is the addition of paragraph 6.7.3 (5.7.3 in STS) which allows high radiation areas, greater than 1000 mrem per hour, inside large areas such as the containment, where no enclosure exists for purposes of locking, to be controlled by barricading and posting, and identified with a flashing light. The current Technical Specifications require that high radiation areas be locked. Construction of lockable enclosures, solely for the purpose of bounding high radiation areas, incurs both significant cost and personnel radiation exposure. Items 6.12.1 (page 6-28) and 6.12.2 were renumbered as 6.7.1 and 6.7.2 and clarified with respect to the measurement of radiation dose rate by addition of the 10 CFR 20 phrase "at 30 cm from the radiation source or from any surface which the radiation penetrates." Because the proposed requirements are less restrictive than the ones which they replace, Change 40 has been classified as Less Restrictive.

41. Section 6.15, Systems Integrity (page 6-33), was combined with Section 4.5.3 (see change 8, above) and retitled "Primary Coolant Sources Outside Containment" and renumbered 6.5.2, emulating STS Section 5.5.2. Because no requirements are being changed, Change 41 is classified as Administrative.

42. Sections 6.16, Iodine Monitoring (page 6-33), and 6.17, Post Accident Sampling (page 6-34), were combined, retitled "Post Accident Sampling Program", and renumbered 6.5.3, emulating STS Section 5.5.3. Because no requirements are being changed, Change 42 is classified as Administrative.

43. Section 6.18, Offsite Dose Calculation Manual (page 6-35), was combined with its definition from existing Section 1.1 and renumbered 6.5.1, emulating STS Section 5.5.1 of the same title. The wording was revised, editorially, to more closely match STS. Because no requirements are being changed, Change 43 is classified as Administrative.

44. Section 6.19, Process Control Program (page 6-35), was combined with its definition from existing Section 1.1 and renumbered as 6.5.15. Section 6.19 deals solely with changes to the Process Control Program. The existing requirement for PRC review and approval, prior to changes to the Process Control Program becoming effective, has been relocated to the Quality Program Description, CPC-2A, along with other PRC review requirements currently in TS section 6.5.1. Because a requirement was deleted and replaced by a requirement in a document controlled by 10 CFR 50.54, Change 44 is classified as Relocated.

45. Section 6.21, Sealed Source Contamination Program (page 6-37), is being relocated to the ODCM. Section 6.21 was added to the Palisades Technical Specifications by Amendment 98 on October 19, 1986, to emulate Section 3/4.7.10 of the former CE STS, NUREG 0212. The subject requirements were not retained in the current STS. Because a TS requirement was deleted and replaced by a requirement in a Program required by the TS, Change 45 is classified as Relocated.

46. Section 6.22, Secondary Water Chemistry (page 6-38), was retitled "Secondary Water Chemistry Program" and renumbered 6.5.9, emulating STS Section 5.5.10 of that title. Because no requirements are being changed, Change 46 is classified as Administrative.

47. A new Section, 6.5.12 "Technical Specification Bases Control Program" was added. That proposed Section is copied from STS Section 5.5.14. Because a new requirement is being added, Change 47 is classified as More Restrictive.

48. Changes have been proposed to allow containment Type A leak rate testing to be performed in accordance with 10 CFR 50, Appendix J, Option B.

Surveillance requirement 4.5.1, Integrated Leakage Rate Tests, has been revised to require that Type A containment testing to be performed in accordance with a newly added "Containment Leak Rate Testing Program".

The Containment Leak Rate Testing Program has been added to the Administrative Controls section as item 6.5.14. The wording of the proposed program is very similar to that contained in the model Technical Specifications developed by the NRC and NEI. The proposed wording specifies that Type A testing be performed in accordance with Option B, and Type B and C testing be performed in accordance with Option A.

These changes are administrative in nature, and involve no change in testing methodology. They simply implement the newly approved Option B testing frequency for Type A testing. This option is desired for Type A testing because it avoids unnecessary testing and thereby affords a reduction in costs, outage time, and personnel radiation exposure. The Type B and C testing will continue to be performed in accordance with Option A because no evaluation has yet been performed to determine the impact of changing these tests to Option B. Therefore, Change 48 is classified as Administrative.

II. Analysis of No Significant Hazards Consideration

Consumers Power Company finds that this proposed Technical Specifications change involves no significant hazards and, accordingly, that a no significant hazards determination per 10 CFR 50.92© is justified.

As discussed in Section I, the each proposed change has been classified as Administrative, Relocated, More Restrictive, or Less Restrictive. Administrative, Relocated, and More Restrictive changes are discussed generically; Less Restrictive changes are discussed individually.

Seven of the proposed changes are considered "Less Restrictive":

7.b Deleting the SR 4.3f requirement to inspect the regenerative heat exchanger each 5 years, and relying on the ASME code requirement to perform this inspection each 10 years.

11 Deleting the SR 4.6.2b requirement to inspect the containment spray nozzles each 5 years, and relying on the ASME code requirement to perform this inspection each 10 years.

16.c Replaces existing allowance for a crew to be "one less than" required with the STS wording allowing a crew to be "less than" required.

18.b Deleting the 6.3.2 requirements that the Radiation Safety Manager be designated by the Plant General Manager and have direct access to the Plant General Manager.

22 Deleting 6.7.1 requirements to provide a written report of any Safety Limit Violation to the NRC within 14 days, and relying on the 10 CFR 36 and 10 CFR 50.73 requirement to submit an LER within 30 days.

28 Deleting the 6.9.1a requirement for submittal of a Startup Report.

35 Deleting the 6.9.3 requirement for submittal of special reports describing environmental reports and reliance upon the administrative procedure requirements for Condition Reporting and on part 4.1 of Appendix B to the Facility Operating License.

40 Deleting the requirement for all high radiation areas to be locked, and allowing certain high radiation areas to be controlled by barricades and postings.

Five of the proposed changes are considered "Relocated":

21.a Relocation of the Section 6.5 requirements for Review and Audit to the Quality Program Description, CPC-2A, which is controlled under 50.54(a).

24 Relocation of the Section 6.8 requirements for controls on procedure changes to the Quality Program Description, CPC-2A, which is controlled under 50.54(a).

38 Relocation of the Section 6.10 requirements for Record Retention to the Quality Program Description, CPC-2A, which is controlled under 50.54(a).

44 Relocation of the Section 6.19 requirements for PRC review and approval of changes to the Process Control Program to the Quality Program Description, CPC-2A, which is controlled under 50.54(a).

45 Relocation of the Section 6.21 requirements for a Sealed Source Contamination Program to the ODCM, which is referenced in the TS and FSAR and is controlled under 50.59.

Five of the proposed changes are considered "More Restrictive":

5.c Increasing the Table 4.2.3 SR c.3 minimum differential pressure.

5.d Reducing the Table 4.3.2 SR g maximum control room temperature.

5.e Adding a new requirement to verify Fuel Pool Ventilation filter flow from the control room.

14 Adding an Administrative requirement for the plant superintendent to approve tests, experiments, and modifications.

47 Adding an Administrative requirement to have a TS Bases Control Program.

The remainder of the proposed changes are considered "Administrative".

<u>Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?</u>

"LESS RESTRICTIVE" CHANGES:

Changes 7.b and 11:

These changes are LESS RESTRICTIVE only in their allowance of a longer surveillance testing interval. The proposed requirements emulate the STS in the reliance on ASME Code requirements to accomplish this testing. The resulting times are those in the ASME Code. Changing a surveillance interval, alone, does not alter any plant design, operating conditions, operating practices, equipment settings, or equipment capabilities. Since these items are unchanged, changing an AOT or a surveillance interval would not increase the probability of any accident previously evaluated.

Excessively extending a surveillance interval could affect the probability that a piece of equipment will function properly upon demand. An overly restrictive surveillance interval could also affect the ability of the equipment to mitigate an accident by imposing unnecessary testing wear, equipment manipulations, and system transients on the plant, and thereby affect the consequences of an accident. The current TS surveillance intervals were based on the operating experience available when they were added to the TS. Typically this was done during the initial plant licensing, circa 1970. In each of these changes where it is proposed that a surveillance interval be extended, the time proposed is that stipulated in the ASME Code, as is done in STS. The surveillance intervals stipulated in the STS are based on a much larger accumulation of operating experience and have been judged by the NRC and by the industry to be appropriate for typical situations. There are no special features of the Palisades plant which would invalidate those judgements for these changes. Therefore, operation of the facility in accordance with the requirements proposed by these changes does not involve a significant increase in the probability of an accident previously evaluated.

Changes 16.c, 18, 22, 28, 35, and 40:

These changes delete administrative requirements and do not affect plant operation or equipment. Change 40 alters the method of control for certain High Radiation Areas in a manner intended to reduce overall radiation dose. These changes do not alter any plant design, operating conditions, operating practices, equipment settings, or equipment capabilities. Since these items are unchanged, deleting the subject administrative requirements would not increase the probability of any accident previously evaluated.

"RELOCATED" changes only move requirements from the TS to documents controlled under 10 CFR parts 50.54(a) or 50.59. There would be no effect on plant operations from moving a requirement from one controlled document to another. However, a requirement in a document controlled under 50.54 or 50.59 could be changed without prior NRC approval.

Documents controlled under 50.54 may only be changed, without prior NRC approval, in a manner which does not reduce the commitments therein. Therefore the requirements relocated to a document controlled under 50.54 might be made more restrictive, but not less restrictive, without prior NRC approval. Changes which make requirements More Restrictive would not involve a significant increase in the probability or consequences of an accident previously evaluated, as discussed below. Changes which make requirements less restrictive would not involve a significant increase in the probability or consequences of an accident previously evaluated because they must receive prior NRC review.

10 CFR 50.59 specifically prohibits changes to the facility as described in the safety analysis report, and to procedures described in the safety analysis report "if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased". Since the conditions which limit changes performed under 50.59 are more restrictive than the conditions which define changes considered to involve a significant hazards consideration, relocation of a requirement from the TS to the FSAR or to documents which are referenced by the FSAR cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

Therefore, changes classified as "Relocated" cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

"MORE RESTRICTIVE" changes only add new requirements, or revise existing requirements to result in additional operational restrictions. Since the TS, with all "MORE RESTRICTIVE" changes incorporated, will still contain all of the requirements which existed prior to the changes; "MORE RESTRICTIVE" changes cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

"ADMINISTRATIVE" changes move or clarify requirements within the TS without affecting their technical content. Since "ADMINISTRATIVE" changes do not alter the technical content of any requirements, they do not alter plant operation or configuration at all. Therefore, they cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

<u>Do the proposed changes create the possibility of a new or different kind of accident from any previously evaluated?</u>

"LESS RESTRICTIVE" changes:

Changes 7.b and 11:

These changes are LESS RESTRICTIVE only in their allowance of a longer surveillance testing interval. The proposed requirements emulate the STS in the reliance on ASME Code requirements to accomplish this testing. The resulting times are those on the ASME Code. Changing a surveillance interval, alone, cannot alter any plant operating conditions, operating practices, equipment settings, or equipment capabilities. Therefore, changing a surveillance interval would not create the possibility of a new or different kind of accident from any previously evaluated.

Changes 16.c, 18, 22, 28, 35, and 40:

Changes delete administrative requirements and do not affect plant operation or equipment. Change 40 alters the method of control for certain High Radiation Areas in a manner intended to reduce overall radiation dose. These changes do not alter any plant design, operating conditions, operating practices, equipment settings, or equipment capabilities. Since these items are unchanged, deleting the subject administrative requirements would not create the possibility of a new or different kind of accident from any previously evaluated.

"RELOCATED" changes only move requirements from the TS to documents controlled under 10 CFR parts 50.54(a) or 50.59. There would be no effect on plant operations from moving a requirement from one controlled document to another. However, a requirement in a document controlled under 50.54 or 50.59 could be changed without prior NRC approval.

Documents controlled under 50.54 may only be changed, without prior NRC approval, in a manner which does not reduce the commitments therein. Therefore the requirements relocated to a document controlled under 50.54 might be made more restrictive, but not less restrictive, without prior NRC approval. Changes which make requirements More Restrictive would not create the possibility of a new or different kind of accident from any previously evaluated, as discussed below. Changes which make requirements less restrictive would not create the possibility of a new or different kind of accident from any previously evaluated because they must receive prior NRC review.

10 CFR 50.59 specifically prohibits changes to the facility as described in the safety analysis report, and to procedures described in the safety analysis report "if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created". Since the conditions which limit changes performed under 50.59 are more restrictive than the conditions which define changes considered to involve a significant hazards consideration, relocation of a requirement from the TS to the FSAR or to documents which are referenced by the FSAR cannot create the possibility of a new or different kind of accident from any previously evaluated.

Therefore, changes classified as "Relocated" cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

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<u>Do the proposed changes involve a significant reduction in a margin of safety?</u>

"LESS RESTRICTIVE" changes:

Changes 7.b and 11:

These changes are LESS RESTRICTIVE only in their allowance of an extension to a surveillance testing interval. Extending a surveillance interval, alone, cannot alter any plant operating conditions, operating practices, equipment settings, or equipment capabilities.

An excessive surveillance interval extension could reduce the margin of safety by reducing assurance that required equipment will function as designed or that parameters are within the required limits. An overly restrictive surveillance interval could also reduce the margin of safety by imposing unnecessary testing wear, equipment manipulations, and system transients on the plant.

The current TS surveillance intervals were based on the operating experience available when they were added to the TS. Typically this was done during the initial plant licensing, circa 1970. In each of these changes where it is proposed that a surveillance interval be extended, the time proposed is that stipulated in the ASME Code, as is done in STS. The surveillance intervals stipulated in the STS and the ASME Code are based on a much larger accumulation of operating experience and have been judged by the NRC and by the industry to be appropriate for typical situations. There are no special features of the Palisades plant which would invalidate those judgements for these changes. Therefore, operation of the facility in accordance with the requirements proposed by these changes does not involve a significant reduction in a margin of safety.

Changes 16.c, 18, 22, 28, 35, and 40:

These changes delete administrative requirements and do not affect plant operation or equipment. Change 40 alters the method of control for certain High Radiation Areas in a manner intended to reduce overall radiation dose. These changes do not alter any plant design, operating conditions, operating practices, equipment settings, or equipment capabilities. Since these items are unchanged, deleting the subject administrative requirements would not involve a significant reduction in a margin of safety.

"RELOCATED" changes only move requirements from the TS to documents controlled under 10 CFR parts 50.54(a) or 50.59. There would be no effect on plant operations from moving a requirement from one controlled document to another. However, a requirement in a document controlled under 50.54 or 50.59 could be changed without prior NRC approval.

Documents controlled under 50.54 may only be changed, without prior NRC approval, in a manner which does not reduce the commitments therein. Therefore the requirements relocated to a document controlled under 50.54 might be made more restrictive, but not less restrictive, without prior NRC approval. Changes which make requirements More Restrictive would not involve a significant reduction in a margin of safety, as discussed below. Changes which make requirements less restrictive would not involve a significant reduction in a margin of safety because they must receive prior NRC review.

10 CFR 50.59 specifically prohibits changes to the facility as described in the safety analysis report, and to procedures described in the safety analysis report "if the margin of safety as defined in the basis for any technical specification is reduced". Since the conditions which limit changes performed under 50.59 are more restrictive than the conditions which define changes considered to involve a significant hazards consideration, relocation of a requirement from the TS to the FSAR or to documents which are referenced by the FSAR cannot involve a significant reduction in a margin of safety.

Therefore, changes classified as "Relocated" cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

"MORE RESTRICTIVE" changes only add new requirements, or revise existing requirements to result in additional operational restrictions. Since the TS, with all "MORE RESTRICTIVE" changes incorporated, will still contain all of the requirements which existed prior to the changes; "MORE RESTRICTIVE" changes cannot involve a significant reduction in a margin of safety.

"ADMINISTRATIVE" changes move or clarify requirements without affecting their technical content. Since "ADMINISTRATIVE" changes do not alter the technical content of any requirements, they do not alter plant operation or configuration at all. Therefore, they cannot involve a significant reduction in a margin of safety.

III. Conclusion

The Palisades Plant Review Committee has reviewed this Technical Specifications Change Request and has determined that proposing this change does not involve an unreviewed safety question. Further, the change involves no significant hazards consideration. This change has been reviewed by the Nuclear Performance Assessment Department.

ATTACHMENT 2

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

TECHNICAL SPECIFICATION CHANGE REQUEST
ADMINISTRATIVE CONTROLS
ADDITIONAL CHANGES

Replacement Proposed Pages

75 Pages

PALISADES PLANT TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

1.0 DEFINITIONS

The following terms are defined for uniform interpretation of these Technical Specifications.

ASSEMBLY RADIAL PEAKING FACTOR - F_r^A

ASSEMBLY RADIAL PEAKING FACTOR shall be the maximum ratio of the power generated in any fuel assembly, to the average fuel assembly power. (Each of these power terms shall be integrated over core height and shall include tilt.)

AVERAGE DISINTEGRATION ENERGY - $\bar{E}$

AVERAGE DISINTEGRATION ENERGY shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

AXIAL OFFSET or AXIAL SHAPE INDEX - AO or ASI

AXIAL OFFSET or AXIAL SHAPE INDEX shall be the ratio of the power generated in the lower half of the core minus the power generated in the upper half of the core, to the sum of those powers.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor, alarm, interlock, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated. Neutron detectors may be excluded from CHANNEL CALIBRATIONS.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and status with other indications and status derived from independent instrument channels measuring the same parameter. A CHANNEL CHECK shall include verification that the monitored parameter is within limits imposed by the Technical Specifications.

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel to verify that it is OPERABLE, including any alarm and trip initiating function.

COLD SHUTDOWN

The COLD SHUTDOWN condition shall be when the primary coolant is at SHUTDOWN BORON CONCENTRATION and T_{ave} is less than 210°F.

CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY is defined to exist when all the following are true:

a. All nonautomatic containment isolation valves and blind flanges are closed (OPERABLE) except as noted in Table 3.6.1.

b. The equipment hatch is properly closed and sealed.

c. At least one door in each personnel air lock is properly closed and sealed.

d. All automatic containment isolation valves are OPERABLE (as demonstrated by satisfying isolation times specified in Table 3.6.1 and leakage criterion in Specification 4.5.2) or are locked closed.

e. The uncontrolled containment leakage satisfies Specification 4.5.

CONTROL RODS

CONTROL RODS shall be all full-length shutdown and regulating rods.

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.6.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

HOT SHUTDOWN

The HOT SHUTDOWN condition shall be when the reactor is subcritical by an amount greater than or equal to the margin as specified in Technical Specification 3.10 and T_{ave} is greater than 525°F.

HOT STANDBY

The HOT STANDBY condition shall be when T_{ave} is greater than 525°F and any of the CONTROL RODS are withdrawn and the neutron flux power range instrumentation indicates less than 2% of RATED POWER.

LOW POWER PHYSICS TESTING

LOW POWER PHYSICS TESTING shall be testing performed under approved written procedures to determine CONTROL ROD worths and other core nuclear properties. Reactor power during these tests shall not exceed 2% of RATED POWER, not including decay heat and PCS T_{ave} and PCS pressure shall be in the range of 371°F to 538°F and 415 psia to 2150 psia, respectively. Certain deviations from normal operating practice which are necessary to enable performing some of these tests are permitted in accordance with the specific provisions in these Technical Specifications.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE, or have OPERABILITY, when it is capable of performing its specified functions, and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication, or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified functions are also capable of performing their related support functions.

POWER OPERATION

The POWER OPERATION condition shall be when the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of RATED POWER.

QUADRANT POWER TILT - T_q

QUADRANT POWER TILT shall be the algebraic ratio of quadrant power minus average quadrant power, to average quadrant power.

RATED POWER

RATED POWER shall be a steady state reactor core output of 2530 MW_t.

1.0 DEFINITIONS (continued)

REACTOR CRITICAL

The reactor is considered critical for purposes of administrative control when the neutron flux wide range channel instrumentation indicates greater than 10^{-4}% of RATED POWER.

REFUELING BORON CONCENTRATION

REFUELING BORON CONCENTRATION shall be a Primary Coolant System boron concentration of at least 1720 ppm AND sufficient to assure the reactor is subcritical by $\geq$ 5% $\Delta\rho$ with all CONTROL RODS withdrawn.

REFUELING OPERATION

A REFUELING OPERATION shall be any operation involving movement of core components (except for incore detectors) when the reactor vessel head is untensioned or removed with fuel in the reactor vessel.

REFUELING SHUTDOWN

The REFUELING SHUTDOWN condition shall be when the primary coolant is at REFUELING BORON CONCENTRATION and T_{ave} is less than 210°F.

SHUTDOWN BORON CONCENTRATION

SHUTDOWN BORON CONCENTRATION shall be a Primary Coolant System boron concentration sufficient to assure the reactor is subcritical by $\geq$ 2% $\Delta\rho$ with all CONTROL RODS in the core and the highest worth CONTROL ROD fully withdrawn.

SHUTDOWN MARGIN

SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming that all CONTROL RODS are fully inserted except for the single highest worth CONTROL ROD which is assumed to be withdrawn.

TOTAL RADIAL PEAKING FACTOR - F_r^T

The TOTAL RADIAL PEAKING FACTOR shall be the maximum product of the ratio of individual assembly power to core average assembly power, times the highest local peaking factor integrated over the total core height, including tilt. Local peaking factor is defined as the maximum ratio of an individual fuel rod power to the assembly average rod power.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit - Reactor Core

The Minimum DNBR of the reactor core shall be maintained greater than or equal to the DNB correlation safety limit.

Correlation	Safety Limit
XNB	1.17
ANFP	1.154
HTP	1.141

Applicability

Safety Limit 2.1 is applicable in HOT STANDBY and POWER OPERATION.

Action

2.1.1 If a Safety Limit is exceeded, the reactor shall be shut down immediately and not restarted until the Commission authorizes resumption of operation in accordance with 10 CFR 50.36(c)(1)(i)(A).

2.2 Safety Limit - Primary Coolant System Pressure (PCS)

The PCS Pressure shall not exceed 2750 psia.

Applicability

Safety Limit 2.2 is applicable when there is fuel in the reactor.

Action

2.2.1 If a Safety Limit is exceeded, the reactor shall be shut down immediately and not restarted until the Commission authorizes resumption of operation in accordance with 10 CFR 50.36(c)(1)(i)(A).

2.3 Limiting Safety System Settings - Reactor Protective System (RPS)

The RPS trip setting limits shall be as stated in Table 2.3.1.

Applicability

Limiting Safety System Settings of Table 2.3.1 are applicable when the associated RPS channels are required to be OPERABLE by Specification 3.17.1.

Action

2.3.1 If an RPS instrument setting is not within the allowable settings of Table 2.3.1, immediately declare the instrument inoperable and complete corrective action as directed by Specification 3.17.1.

3.17 INSTRUMENTATION SYSTEMS

Specification

3.17.4 The Accident Monitoring Instruments listed in Table 3.17.4 shall be OPERABLE. (Specifications 3.0.3, 3.0.4, and 4.0.4 do not apply.)

Applicability

Specification 3.17.4 applies when the PCS temperature is > 300°F.

Action

3.17.4.1 With one required channel of functions 1 through 14 inoperable for one or more functions:

a. Restore channel to OPERABLE status within 7 days.

3.17.4.2 With two required channels of functions 1 through 14 inoperable for one or more functions:

a. Restore one channel to OPERABLE status within 48 hours.

3.17.4.3 With position indication inoperable for one or more Containment Isolation Valves:

a. Restore the indication to OPERABLE status or lock the associated valves in the closed position within 7 days.

3.17.4.4 If any action required by 3.17.4.1 through 3.17.4.3 is not met AND the associated completion time has expired,

a. The reactor shall be placed in HOT SHUTDOWN within 12 hours, and

b. The reactor shall be placed in a condition where the affected equipment is not required, within 48 hours.

3.17.4.5 With one channel of functions 16 through 21 inoperable for one or more functions:

a. Restore the channel to OPERABLE status within 7 days.

3.17.4.6 With two required channels of functions 16 through 21 inoperable for one or more functions:

a. Restore one channel to OPERABLE status within 48 hours.

3.17.4.7 If any action required by 3.17.4.5 or 3.17.4.6 is not met AND the associated completion time has expired:

a. With two CETs in any one quadrant inoperable, complete Action 3.17.4.4 in lieu of Action 3.17.4.7 c),

b. With two RVWL channels inoperable, initiate alternate monitoring within 48 hours,

c. Submit a report to the NRC in accordance with Specification 6.6.7.

d. Restore the channels to OPERABLE status prior to startup from the next refueling.

4.0 SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance requirements shall be applicable during the reactor operating conditions associated with individual Limiting Conditions for Operation unless otherwise stated in an individual surveillance requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the operability requirements for a Limiting Condition for Operation. The time limits of the action requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The action requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the action requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into a reactor operating condition or other specified condition shall not be made unless the Surveillance Requirements associated with a Limiting Condition of Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to plant conditions as required to comply with action requirements.

4.0 SURVEILLANCE REQUIREMENT (Continued)

4.0.5 Deleted

Amendment No. ~~130~~, ~~162~~,

Specifications 4.0.1 through 4.0.4 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance requirements stated in the code of Federal Regulations, 10 CFR 50.36(c)(3):

> "Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during reactor operating conditions or other conditions for which the requirements of the Limiting Conditions for Operation apply, unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a reactor operating condition or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational condition for which the requirements of the associated Limiting Condition for Operation do not apply, unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception the requirements of a specification.

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend the surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the operability requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be operable when Surveillance Requirements have

Specification 4.0.4 establishes the requirement that all applicable surveillances must be met before entry into a reactor operating condition or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component operability requirements or parameter limits are met before entry into an operational condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in reactor operating conditions or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with action requirements, the provisions of Specification 4.0.4 do not apply because this would delay placing the facility in a lower operational condition.

4.1 OVERPRESSURE PROTECTION SYSTEM TESTS

Surveillance Requirements

In addition to the requirements of The Inservice Inspection and Testing Program, Specification 6.5.7, each PORV flow path shall be demonstrated OPERABLE by:

1. Testing the PORVs in accordance with the inservice inspection requirements for ASME Boiler and Pressure Vessel Code, Section XI, Section IWV, Category B valves.

2. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months.

3. When the PORV flow path is required to be OPERABLE by Specification 3.1.8.1:

   (a. Performing a complete cycle of the PORV with the plant above COLD SHUTDOWN at least once per 18 months.

   (b. Performing a complete cycle of the block valve prior to heatup from COLD SHUTDOWN, if not cycled within 92 days.

4. When the PORV flow path is required to be OPERABLE by Specification 3.1.8.2:

   (a. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days.

   (b. Verifying the associated block valve is open at least once per 72 hours.

5. Both High Pressure Safety Injection pumps shall be verified incapable of injection into the PCS at least once per 12 hours, unless the reactor head is removed, when either PCS cold leg temperature is < 300°F, or when both shutdown cooling suction valves, MO-3015 and MO-3016, are open.

Basis

With the reactor vessel head installed when the PCS cold leg temperature is less than 300°F, or if the shutdown cooling system isolation valves MO-3015 and MO-3016 are open, the start of one HPSI pump could cause the Appendix G or the shutdown cooling system pressure limits to be exceeded; therefore, both pumps are rendered inoperable.

Table 4.2.3

VENTILATION SYSTEM TESTS

The Control Room Ventilation and Isolation System and the Fuel Storage Area HEPA/Charcoal Exhaust System shall be demonstrated to be OPERABLE by the following tests:

1. Performing required Control Room Ventilation and Fuel Storage Area filter testing in accordance with the Ventilation Filter Testing Program.

2. At least once per refueling cycle by:

   a. Verifying that on a containment high-pressure and high-radiation test signal, the Control Room Ventilation system automatically switches into the emergency mode of operation with flow through the HEPA filter and charcoal adsorber bank.

   b. Verifying that the Control Room Ventilation system maintains the Control Room at a positive pressure ≥ 1/8 inch WG relative to the outside atmosphere during system emergency mode operation.

   c. Verifying that the Fuel Pool Ventilation System is OPERABLE by initiating flow through the HEPA filter and charcoal adsorbers from the control room.

3. Verifying that the Control Room temperature is ≤ 90°F; once per 12 hours.

Basis - Table 4.2.2 Item 12 - Trisodium Phosphate (TSP) Tests

Item 12.a - TSP quantity verification

Verification of the quantity of TSP in the baskets ensures that neither leakage nor other water sources in the containment reduce the basket content below the required minimum. This requirement ensures that there is an adequate quantity of TSP to adjust the pH of the post LOCA sump solution to a value between 7.0 and 8.0.

Item 12.b - TSP quality verification

Periodic testing is performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. Satisfactory completion of this test assures that the TSP in the baskets is "active" as required by Specification 3.19.

Adequate solubility is verified by submerging a representative sample of TSP from one of the baskets in containment in un-agitated borated water heated to a temperature representing post-LOCA conditions; the TSP must completely dissolve within a 4 hour period. The test time of 4 hours is specified to allow time for the dissolved TSP to naturally diffuse through the un-agitated test solution. Agitation of the test solution during the solubility verification is prohibited, since an adequate standard for the agitation intensity (other than no agitation) cannot be specified. The flow and turbulence in the containment sump during recirculation would significantly decrease the time required for the TSP to dissolve.

Adequate buffering capability is verified by a measured pH of the sample solution, following the solubility verification, between 7 and 8. The sample is cooled and thoroughly mixed prior to measuring pH.

The quantity of the TSP sample, and quantity and boron concentration of the water are chosen to be representative of post-LOCA conditions.

4.3 SYSTEMS SURVEILLANCE

APPLICABILITY

Applies to preoperational and inservice structural surveillance of the reactor vessel and other Class 1, Class 2 and Class 3 system components.

OBJECTIVE

To insure the integrity of the Class 1, Class 2 and Class 3 piping systems and components.

SPECIFICATIONS

a,b,c,d,e,f - Deleted

g. A surveillance program to monitor radiation induced changes in the mechanical and impact properties of the reactor vessel materials shall be maintained as described in Section 4.5.3 of the FSAR.

h. Periodic leakage testing[a],[b] on each check valve listed in Table 4.3.1 shall be accomplished prior to returning to the Power Operation Condition after every time the plant has been placed in the Refueling Shutdown Condition, or the Cold Shutdown Condition for more than 72 hours if such testing has not been accomplished within the previous 9 months, and prior to returning the check valves to service after maintenance, repair or replacement work is performed on the valves.

i. Whenever integrity of a pressure isolation valve listed in Table 4.3.1 cannot be demonstrated and credit is being taken for compliance with Specification 3.3.3.b, the integrity of the remaining check valve in each high pressure line having a leaking valve shall be determined and recorded daily and the position of the other closed valve located in that pressure line shall be recorded daily.

j. Following each use of the LPSI system for shutdown cooling, the reactor shall not be made critical until the LPSI check valves (CK-3103, CK-3118, CK-3133 and CK-3148) have been verified closed.

[a] To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

[b] Reduced pressure testing is acceptable (see footnote 5, Table 4.3.1). Minimum test differential pressure shall not be less than 150 psid.

Basis

The inspection program specified places major emphasis on the areas of highest stress concentration as determined by general design evaluation and experience with similar systems.[1] In addition, that portion of the reactor vessel shell welds which will be subjected to a fast neutron dose sufficient to change ductility properties will be inspected. The inspections will rely primarily on ultrasonic methods utilizing up-to-date analyzing equipment and trained personnel. To the extent applicable, based upon the existing design and construction of the plant, the requirements of Section XI of the Code shall be complied with. Significant exceptions are detailed in the requests for relief which have received NRC approval and are contained in the Class 1, Class 2 and Class 3 Long-Term Inspection Plans.

Valve Testing

To ensure the continued integrity of selected check valves which are relied upon to preclude a potential LOCA outside containment, special requirements for periodic leak tests are specified. In addition a valve disk position check for the LPSI check valves is specified following each use of the LPSI system for shutdown cooling. This position check ensures that the four LPSI check valves have reclosed upon cessation of shutdown cooling flow.

References

(1) FSAR, Section 4.5.6

(2) Deleted

(3) Systematic Evaluation Program Topic V-II.A, NRC letter to the licensee transmitting the final topic evaluation dated November 9, 1981.

TABLE 4.3.1

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum [a] Allowable Leakage
High Pressure Safety Injection		
Loop 1A, Cold Leg	3101	5.0 gpm
	3104	5.0 gpm
Loop 1B, Cold Leg	3116	5.0 gpm
	3119	5.0 gpm
Loop 2A, Cold Leg	3131	5.0 gpm
	3134	5.0 gpm
Loop 2B, Cold Leg	3146	5.0 gpm
	3149	5.0 gpm
Low Pressure Safety Injection		
Loop 1A, Cold Leg	3103	5.0gpm
Loop 1B, Cold Leg	3118	5.0gpm
Loop 2A, Cold Leg	3133	5.0gpm
Loop 2B, Cold Leg	3148	5.0gpm

Footnote:

[a]

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.
2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
4. Leakage rates greater than 5.0 gpm are considered unacceptable.
5. Measured leakage rates must be adjusted for test pressures less than the maximum potential pressure differential across the valve by assuming leakage to be directly proportional to the pressure differential to the one-half power.

4.4 Deleted

4.5 CONTAINMENT TESTS

4.5.1 Integrated Leakage Rate Tests

The containment integrated leak rate testing shall be performed in accordance with the Containment Leak Rate Testing Program.

4.5.2 Local Leak Detection Tests

a. Test

(1) Local leak rate tests shall be performed at a pressure of not less than 55 psig.

(2) Local leak rate tests for checking air lock door seals within 72 hours of each door opening shall be performed at a pressure of not less than 10 psig.

(3) Acceptable methods of testing are halogen gas detection, soap bubble, pressure decay, or equivalent.

(4) The local leak rate shall be measured for each of the following components:

(a) Containment penetrations that employ resilient seal gaskets, sealant compounds, or bellows.

(b) Air lock and equipment door seals.

(c) Fuel transfer tube.

(d) Isolation valves on the testable fluid systems' lines penetrating the containment.

(e) Other containment components which require leak repair in order to meet the acceptance criterion for any integrated leak rate test.

b. Acceptance Criteria

(1) The total leakage from all penetrations and isolation valves shall not exceed 0.60 L_a.

(2) The leakage for an air lock door seal test shall not exceed 0.023 L_a.

4.5.2 Local Leak Detection Tests (continued)

c. Corrective Action

(1) If at any time it is determined that 0.60 L_a is exceeded, repairs shall be initiated immediately. If repairs are not completed and conformance to the acceptance criterion of 4.5.2.b(1) is not demonstrated with 48 hours, the Plant shall be placed in at least hot shutdown within the next 6 hours and in at least cold shutdown within the following 30 hours.

(2) If at any time it is determined that total containment leakage exceeds L_a, within one hour action shall be initiated to bring the Plant to hot shutdown within the next six (6) hours and cold shutdown within the following thirty (30) hours.

(3) If air lock door seal leakage is greater than 0.023 L_a, repairs shall be initiated immediately to restore the door to less than specification 4.5.2.b(2). In the event repairs cannot be completed within 7 days, the Plant shall be brought to a hot shutdown condition within the next six (6) hours and cold shutdown within the following thirty (30) hours.

If air lock door seal leakage results in one (1) door causing total containment leakage to exceed 0.60 L_a, the door shall be declared inoperable and the remaining operable door shall be immediately locked closed and tested within four (4) hours. As long as the remaining door is found to be operable, the provisions of 4.5.2.c(2) do not apply. Repairs shall be initiated immediately to establish conformance with specification 4.5.2.b(1). In the event conformance to this specification cannot be established within 48 hours the Plant shall be brought to a hot shutdown within the next 6 hours and cold shutdown within the following 30 hours.

4.5 CONTAINMENT TESTS

4.5.2 Local Leak Detection Tests (continued)

d. Test Frequency

(1) Individual penetrations and containment isolation valves shall be leak rate tested at a frequency of at least every six months prior to the first postoperational integrated leak rate test and at a frequency of at least every refueling thereafter, not exceeding a two-year interval, except as specified in (a) and (b) below:

(a) The containment equipment hatch and the fuel transfer tube shall be tested at each refueling shutdown or after each time used, if that be sooner.

(b) A full air lock penetration test shall be performed at six-month intervals. During the period between the six-month tests when containment integrity is required, a reduced pressure test for the door seals or a full air lock penetration test shall be performed within 72 hours after either each air lock door opening or the first of a series of openings.

(2) Each three months the isolation valves must be stroked to the position required to fulfill their safety function unless it is established that such operation is not practical during plant operation. The latter valves shall be full-stroked during each cold shutdown.

4.5.3 Containment Isolation Valves

a. The isolation valves shall be demonstrated operable by performance of a cycling test and verification of isolation time for auto isolation valves prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit.

b. Each isolation valve shall be demonstrated operable by verifying that on each containment isolation right channel or left channel test signal, applicable isolation valves actuate to their required position during cold shutdown or at least once per refueling cycle.

c. The isolation time of each power operated or automatic valve shall be determined to be within its limit as specified in Table 3.6.1 when tested in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

4.5.4 Surveillance for Prestressing System

Verify containment structural integrity in accordance with the Containment Structural Integrity Surveillance Program.

Basis

The containment is designed for an accident pressure of 55 psig.[1] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a temperature of about 104°F. With these initial conditions, following a LOCA, the temperature of the steam-air mixture at the peak accident pressure of 55 psig is 283°F.

Prior to initial operation, the containment was strength-tested at 63 psig and then leak rate tested. The design objective of this preoperational leak rate test was established as 0.1% by weight per 24 hours at 55 psig. This leakage rate is consistent with the construction of the containment,[2] which is equipped with independent leak-testable penetrations and contains channels over all unaccessible containment liner welds, which were independently leak-tested during construction.

Accident analyses have been performed on the basis of a leakage rate of 0.1% by weight per 24 hours. With this leakage rate and with a reactor power level of 2530 MWt, the potential public exposure would be below 10 CFR 100 guideline values in the event of the Maximum Hypothetical Accident.[3]

The performance of a periodic integrated leak rate test during plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, this periodic leak rate test is to be performed without preliminary leak detection surveys or leak repairs and containment isolation valves are to be closed in the normal manner.

This normal manner is a coincident two-of-four high radiation or two-of-four high containment pressure signals which will close all containment isolation valves not required for engineered safety features except the component cooling lines' valves which are closed by CHP only. The control system is designed on a two-channel (right and left) concept with redundancy and physical separation. Each channel is capable of initiating containment isolation.[4]

The Type A test requirements including pretest test methods, test pressure, acceptance criteria, and reporting requirements are in accordance with the Containment Leak Rate Testing Program.[5,6]

The frequency of the periodic integrated leak rate test is keyed to the refueling schedule for the reactor because these tests can best be performed during refueling shutdowns. The specified frequency is based on three major considerations. First is the low probability of leaks in the liner because of (a) the test of the leak tightness of the welds during erection; (b) conformance of the complete containment to a low leak rate at 55 psig during preoperational testing which in consistent with 0.1% leakage at design basis accident (DBA) conditions: and (c) absence of any significant stresses in the liner during reactor operation.

4.5 CONTAINMENT TESTS

Basis (continued)

Second is the more frequent testing, at the full accident pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value ($0.60L_a$) of the total leakage that is specified as acceptable from penetrations and isolation valves. Third is the Containment Structural Integrity Surveillance Program which provides assurance that an important part, of the structural integrity of the containment is maintained.

The basis for specification of a total leakage rate of $0.60\ L_a$ from penetrations and isolation valves is specified to provide assurance that the integrated leak rate would remain within the specified limits during the intervals between integrated leak rate tests. This value allows for possible deterioration in the intervals between tests.

The basis for specification of an airlock door seal leakage rate of $0.023\ L_a$ is to provide assurance that the failure of a single airlock door will not result in the total containment leakage exceeding $0.6\ L_a$. The seven (7) day LCO specified for exceeding the airlock door leakage limit is acceptable since it requires that the total containment leakage limit is not exceeded.

References

(1) Updated FSAR Section 5.8.2.

(2) Updated FSAR Section 5.8.8

(3) Updated FSAR 14.22

(4) Updated FSAR Section 8.5.1.2

(5) 10 CFR Part 50, Appendix J.

(6) Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", September 1995.

4.6 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEMS TESTS

Surveillance Requirements

4.6.1 Safety Injection System

a. System tests shall be performed at each reactor refueling interval. A test safety injection signal will be applied to initiate operation of the system. The safety injection and shutdown cooling system pump motors may be de-energized for this test. The system will be considered satisfactory if control board indication and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing (ie, the appropriate pump breakers shall have opened and closed, and all valves shall have completed their travel).

4.6.2 Containment Spray System

a. System test shall be performed at each reactor refueling interval. The test shall be performed with the isolation valves in the spray supply lines at the containment blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

b. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

4.6.3 Pumps

a. The safety injection pumps, shutdown cooling pumps, and containment spray pumps shall be started at intervals not to exceed three months. Alternate manual starting between control room console and the local breaker shall be practiced in the test program.

b. Acceptable levels of performance shall be that the pumps start, reach their rated heads on recirculation flow, and operate for at least fifteen minutes.

4.6.4 Valves

a. Each Safety Injection Tank flow path shall be verified OPERABLE within 7 days prior to each reactor startup by verifying each motor operated isolation valve is open by observing valve position indication and valve itself, and locking open the associated circuit breakers.

b. The Low Pressure Safety Injection flow path shall be verified OPERABLE within 7 days prior to each reactor startup by verifying flow control valve CV-3006 is open, and its air supply is isolated.

Surveillance Requirements (continued)

4.6.4 Valves (continued)

c. The safety injection recirculation path shall be verified OPERABLE within 7 days prior to each reactor startup by verifying valves CV-3027 and 3056 are open and their switches HS-3027A, HS-3027B, HS-3056A, and HS-3056B are open.

d. Each Containment Spray Valve manual control shall be verified to be OPERABLE at least once each refueling by cycling each valve from the control room while observing valve operation at least each 18 months.

4.6.5 Containment Air Cooling System

a. Emergency mode automatic valve and fan operation will be checked for OPERABILITY during each refueling shutdown.

b. Each fan and valve required to function during accident conditions will be exercised at intervals not to exceed three months.

4.6 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEMS TESTS

Basis

The safety injection system and the containment spray system are principal plant safety features that are normally inoperative during reactor operation.

Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes containment isolation and a containment spray system test requires the system to be temporarily disabled. The method of assuring OPERABILITY of these systems is therefore, to combine systems tests to be performed during annual plant shutdowns, with more frequent component tests, which can be performed during reactor operation.

The refueling interval systems tests demonstrate proper automatic operation of the safety injection and containment spray systems. A test signal is applied to initiate automatic action and verification made that the components receive the Safety Injection Signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.[1,2]

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked each shift and the initiating circuits are tested monthly. In addition, the active components (pumps and valves) are to be tested every three months to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval of three months is based on the judgment that more frequent testing would not significantly increase the reliability (ie, the probability that the component would operate when required), yet more frequent test would result in increased wear over a long period of time.

Other systems that are also important to the emergency cooling function are the SI tanks, the component cooling system, the service water system and the containment air coolers. The SI tanks are a passive safety feature. In accordance with the specifications, the water volume and pressure in the SI tanks are checked periodically. The other systems mentioned operate when the reactor is in operation and by these means are continuously monitored for satisfactory performance.

References

(1) FSAR, Section 6.1.3.
(2) FSAR, Section 6.2.3.

(Next Page is 4-42)

4.14 STEAM GENERATORS SURVEILLANCE

4.14.1 Verify Steam Generator tube integrity is acceptable in accordance with the Inservice Inspection and Testing Program, Specification 6.5.7, and the Steam Generator Tube Surveillance Program, Specification 6.5.8.

(Next page is 4-70)

4.16 INSERVICE INSPECTION PROGRAM FOR SHOCK SUPPRESSORS (Snubbers)

Applicability

Applies to periodic surveillance of safety-related snubbers as described per Specification 3.20.

4.16.1 Specifications

Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to the requirements of Specification 6.5.7. As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

a. Visual Inspection

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the following paragraph:

If one or more unacceptable snubbers are found, the next inspection interval shall be 2/3 (-25%) of the previous interval. If no unacceptable snubbers are found, the next interval may be doubled (-25%), but not to exceed 48 months. The interval extension provisions of Technical Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

Inspections performed before the interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections, performed before the original required time interval has elapsed (nominal time less 25%), may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

b. Visual Inspection Acceptance Criteria

Visual inspection shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers, irrespective of type, that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Technical Specification 4.16.1d or 4.16.1e, as applicable. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval.

4.16 INSERVICE INSPECTION PROGRAM FOR SHOCK SUPPRESSORS (Snubbers)

4.16.1 f. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each safety related snubber in use in the plant shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, replacement or reconditioning shall be indicated in the records.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The plant superintendent shall be responsible for overall plant operation and shall delegate in writing the succession for this responsibility during his absence.

The plant superintendent or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

6.1.2 The Shift Supervisor (SS) shall be responsible for the control room command function. During any absence of the SS from the control room while the plant is above COLD SHUTDOWN, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SS from the control room while the plant is in COLD SHUTDOWN, an individual with an active SRO license or Reactor Operator (RO) license shall be designated to assume the control room command function.

6.2 ORGANIZATION

6.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the Palisades plant.

a. Lines of authority, responsibility and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented, and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key positions, or in equivalent forms of documentation. These requirements and the plant specific equivalent of those titles referred to in these Technical Specifications shall be documented in the FSAR.

b. The plant superintendent shall be responsible for overall plant safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate executive shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out radiation safety and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 Plant Staff

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating above COLD SHUTDOWN.

b. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is above COLD SHUTDOWN, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i), and 6.2.2.a and 6.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the requirements.

d. A radiation safety technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. Administrative procedures shall be developed and implemented to limit the working hours of plant staff who perform safety-related functions (e.g., licensed SROs, licensed ROs, radiation safety personnel, auxiliary operators, and key maintenance personnel).

In the event that overtime is used, the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

6.2.2.e Plant Staff (Continued)

Any deviations from the overtime guidelines shall be authorized in advance by the plant superintendent or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the plant superintendent or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

f. The operations manager or an assistant operations manager shall hold an SRO license. The individual holding the SRO license shall be responsible for directing the activities of the licensed operators.

g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. If either SRO on shift satisfies the Shift Engineer qualification requirements, then the STA does not need to be stationed.

6.3 PLANT STAFF QUALIFICATIONS

6.3.1 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions.

6.3.2 The radiation safety manager shall meet the qualifications of a Radiation Protection Manager as defined in Regulatory Guide 1.8, September 1975. For the purpose of this section, "Equivalent," as utilized in Regulatory Guide 1.8 for the bachelor's degree requirement, may be met with four years of any one or combination of the following: (a) Formal schooling in science or engineering, or (b) operational or technical experience and training in nuclear power.

6.3.3 The Shift Technical Advisor shall have a bachelor's degree or equivalent and the Shift Engineer shall have a bachelor's degree in a scientific or engineering discipline. Specific training for both the Shift Technical Advisor and the Shift Engineer shall include plant design, operations, and response and analysis of the plant for transients and accidents. The Shift Engineer shall hold a Senior Reactor Operator license.

6.3.4 The plant staff who perform reviews which ensure compliance with 10 CFR 50.59 shall meet or exceed the minimum qualifications of ANS 3.1-1987, Section 4.7.1 and 4.7.2. A Senior Reactor Operator license or certification shall be considered equivalent to a bachelors degree for the purpose of this specification.

6.4 PROCEDURES

Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, Appendix A, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Site Fire Protection Program implementation.

e. All programs specified in Specification 6.5.

6.5 PROGRAMS AND MANUALS

The following programs shall be established, implemented, and maintained:

6.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain (1) the radioactive effluent controls and radiological environmental monitoring activities and (2) descriptions of the information that should be included in the Radiological Environmental Operating Report, and Radioactive Effluent Release Report required by Specification 6.6.2. and Specification 6.6.3.

c. Changes to ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

    a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the changes, and

    b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after approval by the plant superintendent.

3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

6.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage to the engineered safeguards rooms, from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident, to as low as practical. The systems include the Containment Spray System, the Safety Injection System, the Shutdown Cooling System, and the containment sump suction piping. This program shall include the following:

a. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

c. The portion of the shutdown cooling system that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 255 psig.

d. Piping from valves CV-3029 and CV-3030 to the discharge of the safety injection pumps and containment spray pumps shall be hydrostatically tested at no less than 100 psig.

e. The maximum allowable leakage from the recirculation heat removal systems' components (which include valve stems, flanges and pump seals) shall not exceed 0.2 gallon per minute under the normal hydrostatic head from the SIRW tank (approximately 44 psig).

6.5.3 Post Accident Sampling Program

This program provides controls which will ensure the capability to accurately determine the airborne iodine concentration in vital areas and which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. This program shall include the following:

a. Training of personnel,

b. Procedures for sampling and analysis, and

c. Provisions for maintenance of sampling and analytic equipment.

6.5.4 Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the Offsite Dose Calculation Manual (ODCM), (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas conforming to 10 CFR 20, Appendix B, Table 2, Column 2.

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

d. Limitation on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas conforming to 10 CFR 50, Appendix I,

e. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the doses associated with 10 CFR 20, Appendix B, Table 2, Column 1.

f. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary conforming to 10 CFR 50, Appendix I,

g. Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary conforming to 10 CFR 50, Appendix I,

h. Limitations on the annual doses or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR 190.

6.5.5 Containment Structural Integrity Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Containment Structural Integrity Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 3, July 1990.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Containment Structural Integrity Surveillance Program inspection frequencies.

6.5.6 Primary Coolant Pump Flywheel Surveillance Program

Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the upper flywheels each refueling.

6.5.7 Inservice Inspection and Testing Program

This program provides controls for inservice inspection and testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda (B&PV Code) as follows:

B&PV Code terminology for inservice testing activities	Required interval for performing inservice testing activities
Weekly	≤ 7 days
Monthly	≤ 31 days
Quarterly or every 3 months	≤ 92 days
Semiannually or every 6 months	≤ 184 days
Every 9 months	≤ 276 days
Yearly or annually	≤ 366 days
Biennially or every 2 years	≤ 731 days

b. The provisions of Surveillance Requirement 4.0.2 are applicable to the above required intervals for performing inservice testing activities;

c. The provisions of Surveillance Requirement 4.0.3 are applicable to inservice testing activities; and

d. Nothing in the B&PV Code shall be construed to supersede the requirements of any Technical Specification.

6.5.8 Steam Generator Tube Surveillance Program

This program provides controls for surveillance testing of the Steam Generator (SG) tubes to ensure that the structural integrity of this portion of the Primary Coolant System (PCS) is maintained. The program shall contain controls to ensure:

a. Steam Generator Tube Sample Selection and Inspection

The inservice inspection may be limited to one SG on a rotating schedule encompassing 6% of the tubes if the results of previous inspections indicate that both SGs are performing in a like manner. If the operating conditions in one SG are found to be more severe than those in the other SG, the sample sequence shall be modified to inspect the most severe conditions.

The SG tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 6.5.8-1. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all SGs; the tubes selected for these inspections shall be selected on a random basis except:

1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

2. The first sample of tubes selected for each inservice inspection of each SG shall include:

a) All nonplugged tubes that previously had detectable wall penetrations greater than 20%.

b) Tubes in those areas where experience has indicated potential problems.

c) A tube inspection shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

6.5.8 Steam Generator Tube Surveillance Program (continued)

3. The tubes selected as the second and third samples (if required by Table 6.5.8-1) during each inservice inspection may be subjected to a partial tube inspection provided:

   a) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

   b) The inspections include those portions of the tubes where imperfections were previously found.

4. The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.
Note:	In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

b. Inspection Frequencies

The above required inservice inspection of SG tubes shall be performed at the following frequencies:

1. Inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspections results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

6.5.8 Steam Generator Tube Surveillance Program (continued)

2. If the results of the inservice inspection of a SG conducted in accordance with Table 6.5.8-1 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 6.5.8.b.1; the interval may then be extended to a maximum of once per 40 months.

3. Additional, unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 6.5.8-1 during the shutdown subsequent to any of the following conditions:

a) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.1.5.

b) A seismic occurrence greater than the Operating Basis Earthquake.

c) A loss-of-coolant accident resulting in initiation of flow of the engineered safeguards.

d) A main steam line or main feedwater line break.

c. Acceptance Criteria

1. As used in this Specification:

a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

c) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

6.5.8 Steam Generator Tube Surveillance Program (continued)

e) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

f) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.

g) Unserviceable described the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 6.5.8.b.3, above.

h) Tube Inspection means an inspection of the SG tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

i) Preservice Inspection means an inspection of the full length of each tube in SG performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the shop hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

2. The SG shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 6.5.8-1.

TABL[illegible].8-1
STEAM GENERATO[illegible]UBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first Sample
			C-3	Perform action for C-3 result of first Sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. 24 hour verbal notification to NRC with written follow up within next 30 days	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes each S.G. and plug defective tubes.	N/A	N/A

S = 6/n %   Where n is the number of steam generators inspected during an inspection

6.5.9 Secondary Water Chemistry Program

A program shall be established, implemented and maintained for monitoring of secondary water chemistry to inhibit steam generator tube degradation and shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables,

b. Identification of the procedures used to measure the values of the critical variables,

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

d. Procedures for the recording and management of data,

e. Procedures defining corrective actions for all off-control point chemistry conditions, and

f. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

6.5.10 Ventilation Filter Testing Program

A program shall be established to implement the following required testing of Control Room Ventilation (CRV) and Fuel Pool Ventilation (FPV) systems at the frequencies specified in Regulatory Guide 1.52, Revision 2 (RG 1.52), and in accordance with RG 1.52 and ASME N510-1989, at the system flowrates and tolerances specified below*:

a. Demonstrate for each of the ventilation systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% for the CRV and < 1.00% for the FPV when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Flowrate (CFM)
V-8A or V-8B	7300 ± 20%
V-8A and V-8B	10,000 ± 20%
V-95 or V-96	12,500 ± 10%

b. Demonstrate for each of the ventilation systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% for the CRV and < 1.00% for the FPV when tested in accordance with RG 1.52 and ASME N510-1989.

Ventilation System	Flowrate (CFM)
V-8A and V-8B	10,000 ± 20%
V-26A and V-26B	3200 +10% -5%

c. Demonstrate for each of the ventilation systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in RG 1.52 shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and equal to the relative humidity specified as follows:

Ventilation System	Penetration	Relative Humidity
VF-66	6.00%	95%
VFC-26A and VFC-26B	0.157%	70%

d. For each of the ventilation systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Delta P (In H_2O)	Flowrate (CFM)
V-8A and V-8B	6.0	10,000 ± 20%
VF-26A and VF-26B	8.0	3200 +10% -5%

e. Demonstrate that the heaters for each of the ventilation systems dissipate the following specified value ± 20% when tested in accordance with ASME N510-1989:

Ventilation System	Wattage
VHX-26A and VHX-26B	15 kW

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Ventilation Filter Testing Program frequencies.

* Should the 720-hour limitation on charcoal adsorber operation occur during a plant operation requiring the use of the charcoal adsorber - such as refueling - testing may be delayed until the completion of the plant operation or up to 1,500 hours of filter operation; whichever occurs first.

6.5.11 Reserved

6.5.12 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

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

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

 2. A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

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

d. Proposed changes that meet the criteria of Specification 6.5.12 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

6.5.13 Reserved

6.5.14 Containment Leak Rate Testing Program

Programs shall be established to implement the leak rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The Type A test program shall meet the requirements of 10 CFR 50, Appendix J, Option B and shall be in accordance with the guidelines of Regulatory Guide 1.163, "Performance-Based Containment Leakage-Test Program, dated September 1995." The Type B and Type C test program shall meet the requirements of 10 CFR 50, Appendix J, Option A, as modified by the exemption from certain requirements of 10 CFR 50 Appendix J which was granted in an NRC letter to Consumers Power Company dated December 6, 1989.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.64 psig (FSAR Table 14.18.1-4).

The maximum allowable containment leak rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

Leak rate acceptance criteria are:

a. Containment leak rate acceptance criteria is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leak rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock leak rate acceptance criteria is $\leq 0.023\ L_a$ for each door, when pressurized to ≥ 10 psig.

The Surveillance interval extensions of LCO 4.0.2 <u>are not</u> applicable to the Containment Leak Rate Testing Program requirements.

The provisions of LCO 4.0.3 <u>are</u> applicable to the Containment Leak Rate Testing Program requirements.

6.5.15 Process Control Program

a. The Process Control Program shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 10 CFR 71, Federal and State regulations, and other requirements governing the disposal of the radioactive waste.

b. Changes to the Process Control Program:

1. Shall be documented and records of reviews performed shall be retained as required by the Quality Program, CPC-2A. This documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluation justifying the change(s) and

   b) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after approval by the plant superintendent.

6.6 REPORTING REQUIREMENTS

The following reports shall be submitted in accordance with 10 CFR 50.4.

6.6.1 Occupational Radiation Exposure Report

This report shall include a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignment to various duty functions may be estimates based on pocket dosimeter, electronic dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions. The report shall be submitted by April 30 of each year.

6.6.2 Radiological Environmental Operating Report

The Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

6.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual and Process Control Program, and shall be in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

6.6.4 Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the NRC to arrive no later than the fifteenth of each month following the calendar month covered by the report.

6.6.5 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

3.1.1	ASI Limits.
3.10.5	Regulating Group Insertion Limits
3.23.1	Linear Heat Rate (LHR) Limits
3.23.2	Radial Peaking Factor Limits

b. The analytical methods used to determine the core operating limits shall be those approved by the NRC, specifically those described in the latest approved revision of the following documents:

1. XN-75-27(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," and Supplements 1(A), 2(A), 3(P)(A), 4(P)(A), and 5(P)(A); Exxon Nuclear Company. (LCOs 3.1.1, 3.10.1, 3.10.5, 3.23.1, & 3.23.2)

2. ANF-84-73(P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," and Appendix B(P)(A) and Supplements 1(P)(A), 2(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.1, 3.10.5, 3.23.1, & 3.23.2)

3. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company. (LCOs 3.1.1, 3.23.1, & 3.23.2)

4. ANF-84-093(P)(A), "Steamline Break Methodology for PWRs," and Supplement 1(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.10.1, 3.10.5, 3.23.1, & 3.23.2)

5. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company. (LCOs 3.1.1, 3.10.5, 3.23.1, & 3.23.2)

6. EXEM PWR Large Break LOCA Model as defined by: (LCOs 3.10.5, 3.23.1, & 3.23.2)

   a) XN-NF-82-20(A), "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company.

   b) XN-NF-82-07(P)(A), "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company.

   c) XN-NF-81-58(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company.

6.6.5 COLR (continued)

d) XN-NF-85-16(A), "PWR 17x17 Fuel Cooling Tests Program," Volume 1 and Supplements 1(P)(A), 2(P)(A), and 3(P)(A), and Volume 2 and Supplement 1(P)(A); Exxon Nuclear Company.

e) XN-NF-85-105(A), "Scaling of FCTF Based Reflood Heat Transfer Correlation for other Bundle Designs," and Supplement 1(P)(A); Exxon Nuclear Company.

7. XN-NF-78-44(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company. (LCOs 3.10.5, 3.23.1, & 3.23.2)

8. ANF-1224(P)(A), "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," and Supplement 1(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.1, 3.23.1, & 3.23.2)

9. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation. (LCOs 3.1.1, 3.10.5, 3.23.1, & 3.23.2)

10. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation. (LCOs 3.1.1, 3.23.1, & 3.23.2)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC.

6.6.6 Reserved

6.6.7 Accident Monitoring Instrument Report

When a report is required by Condition 3.17.4.7c, "Accident Monitoring Instrumentation," a report shall be submitted within the following 30 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to OPERABLE status.

6.6.8 Containment Structural Integrity Surveillance Report

Reports shall be submitted to the NRC covering Prestressing, Anchorage, and Liner and Penetration tests within 90 days after completion of the tests.

6.6.9 Steam Generator Tube Surveillance Report

The following reports shall be submitted to the Commission following each inservice inspection of steam generator tubes:

a. The number of tubes plugged in each steam generator shall be reported to the Commission within 15 days following the completion of each inspection, and

b. The complete results of the steam generator tube inservice inspection shall be reported to the Commission within 12 months following completion of the inspection. This report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of an imperfection.

   3. Identification of tubes plugged.

c. Results of steam generator tube inspections that fall into Category C-3 shall require 24 hour verbal notification to the NRC prior to resumption of plant operation. A written followup within the next 30 days shall provide a description of investigations and corrective measures taken to prevent recurrence.

6.7 HIGH RADIATION AREA

6.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., health physics technicians) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates < 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Work Request.

6.7.2 In addition to the requirements of Specification 6.7.1, except as allowed by 6.7.3, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Supervisor on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

6.7.3 For individual high radiation areas with radiation levels of ≥ 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

ATTACHMENT 3

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

TECHNICAL SPECIFICATION CHANGE REQUEST
ADMINISTRATIVE CONTROLS
ADDITIONAL CHANGES

Revised Pages Marked to Show Changes From Prior Submittal

10 Pages

4.0 SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance requirements shall be applicable during the reactor operating conditions associated with individual Limiting Conditions for Operation unless otherwise stated in an individual surveillance requirement.

~~4.0.2 Unless otherwise specified, each surveillance requirement shall be performed within the specified time interval with:~~

~~a. A maximum allowable extension not to exceed 25% of the surveillance interval, and~~

~~b. A total maximum combined interval time for any three consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.~~

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the surveillance interval. **[Changed by Amendment 171]**

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the operability requirements for a Limiting Condition for Operation. The time limits of the action requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The action requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the action requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into a reactor operating condition or other specified condition shall not be made unless the Surveillance Requirements associated with a Limiting Condition of Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to plant conditions as required to comply with action requirements.

Specifications 4.0.1 through 4.0.4 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance requirements stated in the code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during reactor operating conditions or other conditions for which the requirements of the Limiting Conditions for Operation apply, unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a reactor operating condition or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational condition for which the requirements of the associated Limiting Condition for Operation do not apply, unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception the requirements of a specification.

~~Specification 4.0.2 establishes the conditions under which the specified time interval for Surveillance Requirements may be extended. Item a. permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. Item b. limits the use of the provisions of Item a. to ensure that it is not used repeatedly to extend the surveillance interval beyond that specified. The limits of Specification 4.0.2 are based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. These provisions are sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.~~

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend the surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval. **[Changed by Amendment 171]**

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the operability requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be operable when Surveillance Requirements have

4.1 OVERPRESSURE PROTECTION SYSTEM TESTS

Surveillance Requirements

In addition to the requirements of The Inservice Inspection and Testing Program, Specification 6.5.7, each PORV flow path shall be demonstrated OPERABLE by:

1. Testing the PORVs in accordance with the inservice inspection requirements for ASME Boiler and Pressure Vessel Code, Section XI, Section IWV, Category B valves.

2. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months.* [Changed by Amendment 171]

3. When the PORV flow path is required to be OPERABLE by Specification 3.1.8.1:

   (a. Performing a complete cycle of the PORV with the plant above COLD SHUTDOWN at least once per 18 months.

   (b. Performing a complete cycle of the block valve prior to heatup from COLD SHUTDOWN, if not cycled within 92 days.

4. When the PORV flow path is required to be OPERABLE by Specification 3.1.8.2:

   (a. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days.

   (b. Verifying the associated block valve is open at least once per 72 hours.

5. Both High Pressure Safety Injection pumps shall be verified incapable of injection into the PCS at least once per 12 hours, unless the reactor head is removed, when either PCS cold leg temperature is < 300°F, or when both shutdown cooling suction valves, MO-3015 and MO-3016, are open.

Basis

With the reactor vessel head installed when the PCS cold leg temperature is less than 300°F, or if the shutdown cooling system isolation valves MO-3015 and MO-3016 are open, the start of one HPSI pump could cause the Appendix G or the shutdown cooling system pressure limits to be exceeded; therefore, both pumps are rendered inoperable.

* ~~For Cycle 11 only, this surveillance need not be performed until prior to startup for Cycle 12.~~ **[Changed by Amendment 171]**

4.2 EQUIPMENT SAMPLING AND TESTS

Table 4.2.3

VENTILATION SYSTEM TESTS

The Control Room Ventilation and Isolation System and the Fuel Storage Area HEPA/Charcoal Exhaust System shall be demonstrated to be OPERABLE by the following tests:

1. Performing required Control Room Ventilation and Fuel Storage Area filter testing in accordance with the Ventilation Filter Testing Program.

2. At least once per refueling cycle by:

   a. Verifying that on a containment high-pressure and high-radiation test signal, the Control Room Ventilation system automatically switches into the emergency mode of operation with flow through the HEPA filter and charcoal adsorber bank.

   b. Verifying that the Control Room Ventilation system maintains the Control Room at a positive pressure ≥ 1/8 inch WG relative to the outside atmosphere during system emergency mode operation.

   c. Verifying that the Fuel Pool Ventilation System is OPERABLE by initiating flow through the HEPA filter and charcoal adsorbers from the control room.

3. Verifying that the Control Room temperature is ≤ 90°F; once per 12 hours.

~~4. Verifying that the Fuel Pool Ventilation System is OPERABLE by initiating flow through the HEPA filter and charcoal adsorbers from the control room.~~

Basis

The containment is designed for an accident pressure of 55 psig.[1] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a temperature of about 104°F. With these initial conditions, following a LOCA, the temperature of the steam-air mixture at the peak accident pressure of 55 psig is 283°F.

Prior to initial operation, the containment was strength-tested at 63 psig and then leak rate tested. The design objective of this preoperational leak rate test was established as 0.1% by weight per 24 hours at 55 psig. This leakage rate is consistent with the construction of the containment,[2] which is equipped with independent leak-testable penetrations and contains channels over all unaccessible containment liner welds, which were independently leak-tested during construction.

Accident analyses have been performed on the basis of a leakage rate of 0.1% by weight per 24 hours. With this leakage rate and with a reactor power level of 2530 MWt, the potential public exposure would be below 10 CFR 100 guideline values in the event of the Maximum Hypothetical Accident.[3]

The performance of a periodic integrated leak rate test during plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, this periodic leak rate test is to be performed without preliminary leak detection surveys or leak repairs and containment isolation valves are to be closed in the normal manner.

This normal manner is a coincident two-of-four high radiation or two-of-four high containment pressure signals which will close all containment isolation valves not required for engineered safety features except the component cooling lines' valves which are closed by CHP only. The control system is designed on a two-channel (right and left) concept with redundancy and physical separation. Each channel is capable of initiating containment isolation.[4]

The Type A test requirements including pretest test methods, test pressure, acceptance criteria, and reporting requirements are in accordance with the Containment Leak Rate Testing Program.[5,6]

The frequency of the periodic integrated leak rate test is keyed to the refueling schedule for the reactor because these tests can best be performed during refueling shutdowns. The specified frequency is based on three major considerations. First is the low probability of leaks in the liner because of (a) the test of the leak tightness of the welds during erection; (b) conformance of the complete containment to a low leak rate at 55 psig during preoperational testing which in consistent with 0.1% leakage at design basis accident (DBA) conditions: and (c) absence of any significant stresses in the liner during reactor operation.

6.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage to the engineered safeguards rooms, from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident, to as low as practical. The systems include the Containment Spray system, ~~and~~ the Safety Injection system, the Shutdown Cooling System, and ~~including~~ the containment sump suction piping. This program shall include the following:

a. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

c. The portion of the shutdown cooling system that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 255 psig.

d. Piping from valves CV-3029 and CV-3030 to the discharge of the safety injection pumps and containment spray pumps shall be hydrostatically tested at no less than 100 psig.

e. The maximum allowable leakage from the recirculation heat removal systems' components (which include valve stems, flanges and pump seals) shall not exceed 0.2 gallon per minute under the normal hydrostatic head from the SIRW tank (approximately 44 psig).

6.5.3 Post Accident Sampling Program

This program provides controls which will ensure the capability to accurately determine the airborne iodine concentration in vital areas and which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. This program shall include the following:

a. Training of personnel,

b. Procedures for sampling and analysis, and

c. Provisions for maintenance of sampling and analytic equipment.

6.5.5 Containment Structural Integrity Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Containment Structural Integrity Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 3, ~~1989~~ July 1990.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Containment Structural Integrity Surveillance Program inspection frequencies.

6.5.6 Primary Coolant Pump Flywheel Surveillance Program

Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the upper flywheels each refueling.

6.5.7 Inservice Inspection and Testing Program

This program provides controls for inservice inspection and testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda (B&PV Code) as follows:

B&PV Code terminology for inservice testing activities	Required interval for performing inservice testing activities
Weekly	≤ 7 days
Monthly	≤ 31 days
Quarterly or every 3 months	≤ 92 days
Semiannually or every 6 months	≤ 184 days
Every 9 months	≤ 276 days
Yearly or annually	≤ 366 days
Biennially or every 2 years	≤ 731 days

b. The provisions of Surveillance Requirement 4.0.2 are applicable to the above required intervals for performing inservice testing activities;

c. The provisions of Surveillance Requirement 4.0.3 are applicable to inservice testing activities; and

d. Nothing in the B&PV Code shall be construed to supersede the requirements of any Technical Specification.

6.5.13 Reserved

6.5.14 Containment Leak Rate Testing Program

Programs shall be established to implement the leak rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The Type A test program shall meet the requirements of 10 CFR 50, Appendix J, Option B and shall be in accordance with the guidelines of Regulatory Guide 1.163, "Performance-Based Containment Leakage-Test Program, dated September 1995." The Type B and Type C test program shall meet the requirements of 10 CFR 50, Appendix J, Option A, as modified by the exemption from certain requirements of 10 CFR 50 Appendix J which was granted in an NRC letter to Consumers Power Company dated December 6, 1989.

~~The containment is designed for an accident pressure, P_a, of 55 psig; the maximum allowable containment leak rate, L_a, at P_a, shall be 0.1% of containment air weight per day.~~

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.64 psig (FSAR Table 14.18.1-4).

The maximum allowable containment leak rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

Leak rate acceptance criteria are:

a. Containment leak rate acceptance criteria is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leak rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

~~b. Air lock testing acceptance criteria are:~~
~~1) Overall air lock leak rate is $\leq 0.60\ L_a$ when tested at $\geq P_a$;~~
~~2) For each door, leak rate is $\leq 0.023\ L_a$ when pressurized to ≥ 10 psig.~~

b. Air lock leak rate acceptance criteria is $\leq 0.023\ L_a$ for each door, when pressurized to ≥ 10 psig.

The Surveillance interval extensions of LCO 4.0.2 are not applicable to the Containment Leak Rate Testing Program requirements.

The provisions of LCO 4.0.3 are applicable to the Containment Leak Rate Testing Program requirements.

6.5.15 Process Control Program

a. The Process Control Program shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 10 CFR 71, Federal and State regulations, and other requirements governing the disposal of the radioactive waste.

b. Changes to the Process Control Program:

1. Shall be documented and records of reviews performed shall be retained as required by the Quality Program, CPC-2A. This documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluation justifying the change(s) and

   b) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after approval by the plant superintendent.

6.6 REPORTING REQUIREMENTS

The following reports shall be submitted in accordance with 10 CFR 50.4.

6.6.1 Occupational Radiation Exposure Report

This report shall include a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignment to various duty functions may be estimates based on pocket dosimeter, electronic dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions. The report shall be submitted by April 30 of each year.

6.6.2 Radiological Environmental Operating Report

The Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM) and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

6.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM) and Process Control Program, and shall be in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

6.6.4 Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the NRC to arrive no later than the fifteenth of each month following the calendar month covered by the report.