Entergy

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Pilgrim Nuclear Power Station
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Michael A. Balduzzi
Site Vice President

December 14, 2004

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

SUBJECT: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No. 50-293
License No. DPR-35

Technical Specifications Amendment Request to Relocate Administrative Titles and Responsibilities and Other Administrative Changes

REFERENCE: NUREG-1433, Standard Technical Specifications for General Electric Plants, BWR/4, Revision 3

LETTER NUMBER: 2.04.083

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (Entergy) hereby proposes to amend its Facility Operating License, DPR-35. These changes are consistent with the content in Standard Technical Specifications (NUREG-1433, Revision 3) and changes previously approved by the NRC for other facilities. Entergy has reviewed the proposed amendment in accordance with 10 CFR 50.92 and concludes it does not involve a significant hazards consideration.

Commitments made by the licensee in this letter are listed in Attachment 2.

Entergy requests approval of the proposed amendment by December 30, 2005. Once approved, the amendment shall be implemented within 60 days.

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If you have any questions or require additional information, please contact Bryan Ford at (508) 830-8403.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 14th day of December 2004.

Sincerely,

Michael A. Balduzzi

Michael A. Balduzzi

ES/dm

Enclosure: Evaluation of the proposed change – 11 pages

Attachments:
1. Proposed Technical Specification and Bases Changes (mark-up) – 27 pages
2. List of Regulatory Commitments – 1 page

cc:

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ENCLOSURE

EVALUATION OF THE PROPOSED CHANGE

ENCLOSURE

Evaluation of the Proposed Change

Subject: Technical Specifications Amendment Request to Relocate Administrative Titles and Responsibilities and Other Administrative Changes

1. Description

Entergy Nuclear Operations, Inc. (Entergy) is requesting to amend Operating License DPR-35 for Pilgrim Nuclear Power Station (PNPS). The proposed changes would revise the Operating License, Technical Specifications (TS):

(1) To eliminate certain administrative requirements for Safety Limit violations that are adequately addressed in 10 CFR 50.36(c)(1)(i)(A), 10 CFR 50.72, 10 CFR 50.73, and by procedures. Elimination of duplicative regulatory reporting requirements will avoid future, and eliminate existing, inconsistent or conflicting regulatory requirements.

(2) To replace plant-specific titles with generic titles. Actual plant-specific titles that fulfill the generic titles will be relocated from the TS to the Final Safety Analysis Report (FSAR), which will facilitate future Pilgrim organization title changes.

(3) To remove the remaining responsibilities of the Operations Review Committee (ORC).

(4) To replace descriptive details specified in TS 3.13.A.1 associated with 10 CFR 50.55a(f), "Inservice testing requirements," with reference to the "Inservice Code Testing Program." Similar detail from TS 4.13.A.1 replaced with editorial re-wording to more closely match presentation in NUREG-1433 Specification 5.5.7, "Inservice Testing Program."

(5) To make administrative changes to TS 5.5.4, "Radioactive Effluent Controls Program," to more closely match presentation in NUREG-1433 Specification 5.5.4.

(6) To make editorial corrections and clarifications.

These proposed changes are considered administrative and will enhance consistency with the BWR/4 Standard Technical Specifications, NUREG-1433.

Entergy requests approval of the proposed amendment by December 30, 2005. Once approved, the amendment shall be implemented within 60 days.

2. Proposed Changes

2.1 Delete the following administrative reporting and restart authorization requirements that apply in the event of a Safety Limit violation, and editorial rewording to reflect these deletions:

1. TS Section 2.2.1, "Within one hour notify the NRC Operations Center in accordance with 10CFR50.72."

2. TS Section 2.2.3, "The Station Director and Senior Vice President - Nuclear and the Nuclear Safety Review and Audit Committee (NSRAC) shall be notified within 24 hours."

3. TS Section 2.2.4, "A Licensee Event Report shall be prepared pursuant to 10CFR50.73. The Licensee Event Report shall be submitted to the Commission, the Operations Review Committee (ORC), the NSRAC and the Station Director and Senior Vice President - Nuclear within 30 days of the violation."

4. TS Section 2.2.5, "Critical operation of the unit shall not be resumed until authorized by the Commission."

2.2 Replace the following plant-specific titles with generic titles as shown, and include TS 5.2.1 requirement to retain specific titles of those personnel fulfilling the responsibilities in the Final Safety Analysis Report (FSAR):

1. TS 5.1.1, "Station Director" is replaced with "plant manager" (in two locations).

2. TS 5.1.2, "Nuclear Operations Supervisor (NOS)" and "NOS" are replaced with "control room supervisor (CRS)" and "CRS" (three locations).

3. TS 5.2.1.a, last sentence, is revised to state: "These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Pilgrim Station Final Safety Analysis Report (FSAR)."

4. TS 5.2.1.b, "Station Director" is replaced with "plant manager."

5. TS 5.2.1.c, "The Vice President – Operations" is replaced with "A specified corporate officer."

6. TS 5.2.2.h, "Operations Department Manager" is replaced with "operations manager or assistant operations manager" and the specific position titles "Nuclear Watch Engineers," Nuclear Operations Supervisors," and "Nuclear Plant Operators" are removed.

7. TS 5.2.2.i, "The Shift Control Room Engineer (SCRE)" is replaced with "An individual" (similar change in three locations). "Nuclear Operations Supervisor (NOS)" is replaced with "unit operations shift crew." Other editorial changes are made for consistency.

8. TS 5.5.1, "the approval of the Chemistry and Radiological Department Managers" is replaced with "the approval of the plant manager."

9. TS 5.7.1, "Health Physics personnel" is replaced with "radiation protection personnel."

10. TS 5.7.1.c, "Radiation Protection Manager" is replaced with "radiation protection manager."

11. TS 5.7.2, "the Nuclear Watch Engineer on duty" is replaced with "an SRO on duty."

12. TS 5.7.2, "health physics supervision" is replaced with "radiation protection supervision."

2.3 Remove Operations Review Committee (ORC) responsibilities as indicated:

1. TS 3.7.A.2.b, Footnote *, remove "ORC approved" criterion for the stated administrative control.

2. TS 5.5.1, remove "review and acceptance by the Operations Review Committee and."

2.4 Modify the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) inservice testing requirements as indicated:

1. For the definition of REFUELING INTERVAL, replace "ASME Code, Section XI IWP and IWV" with "Inservice Code Testing Program."

2. For TS 3.13 and for 4.13 Applicability, delete "or equivalent". For TS 3.13 Objective, delete "(safety related) or equivalent (important to safety)". For TS 3.13.A.1 and for 4.13 Objective, replace "safety and safety related" with "ASME Code Class 1, 2, and 3."

3. For TS 3.13.A.1 and for 4.13.A.5, insert "Inservice Code Testing Program" in place of "ASME Boiler and Pressure Vessel Code." Also for TS 3.13.A.1 delete the follow on descriptive detail: "Section XI "Rules for Inservice Testing of Nuclear Power Plant Components" Subsections IWP and IWV as required by 10CFR50.55a(f), except where specific relief has been granted by the NRC pursuant to 10CFR50.55a(f)(6)(i)."

4. For TS 4.13.A.1 and 4.13.A.2 combine as TS 4.13.A.1. Delete the text of TS 4.13.A.1 and replace the 4.13.A.2 introduction "Test Frequencies for Code" with "The ASME OM Code" (Note that the "Code Terminology" and associated "Frequencies" Table remains unchanged).

5. Similarly, Bases detail is corrected and draft changes are provided for information.

2.5 Make the following administrative changes to TS 5.5.4, "Radioactive Effluent Controls Program":

1. TS 5.5.4.b, replace "10 CFR 20, Appendix B, Table 2, Column 2" with "ten times the concentration values in Appendix B, Table 2, Column 2, to 10 CFR 20.1001 – 20.2402."

2. TS 5.5.4.e, delete "and projected dose" from the current sentence and add second sentence "Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days."

3. TS 5.5.4.g, reword "effluents to areas beyond" adding clarifying phrases to read "effluents from the site boundary to areas at or beyond."

4. TS 5.5.4.j, add "beyond the site boundary" after "member of the public."

2.6 Make the following editorial changes, corrections, or clarifications:

1. Remove the ** note from TS 3.7.A.5.

2. TS 3.5.A.5 misspelled word "rector" is corrected to "reactor."

3. TS 3.8.2, on page 3/4.8-2, delete one of two periods in the APPLICABILITY statement. Additionally, the misspelled word "CHANNEI" in TS 4.8.2.3 is corrected to "CHANNEL."

4. TS 4.9.A.c, on page 3/4.9-2, has a typo in the last sentence reference 4.9.A.1.b.1, which is corrected to 4.9.A.1.b.2.

5. TS 3.9.B.2 last sentence on page 3/4.9-4, "and the NRC is notified within one (1) hour as required by 10 CFR 50.72" is deleted.

6. TS 4.9.A.4.b, on page 3/4.9-4, correct the abbreviation for the unit hertz to "Hz" (without subscripting the "z").

7. TS 3.11.C.2, on page 3/4/11-3, correct the typographical reference to Table "3.3-1" from "3.3.1."

3. Background

These proposed changes are consistent with the latest revision of the BWR/4 Standard Technical Specifications, NUREG-1433 (Revision 3, dated 3/31/2004). The proposed changes are also consistent with specific changes that have been made to NUREG-1433 since its initial issuance as Revision 0, dated 9/28/92.

1. Removal of the administrative reporting and restart authorization requirements, that apply in the event of a Safety Limit violation, have been specifically addressed by the Technical Specification Task Force (TSTF) in change TSTF-5, which was NRC approved on June 11, 1996. The basis for this change was that the requirements are addressed in 10 CFR 50.36(c)(1)(i)(A), which requires notification and reporting in accordance with 10 CFR 50.72 and 50.73, and Commission approval for resuming operation.

2. Use of generic titles and removal of plant-specific titles has been specifically addressed in change TSTF-65, which was NRC approved on December 2, 1997. The acceptance of this change was based on the commitment to relocate and control plant specific titles in the FSAR. This change did not eliminate any qualifications, responsibilities or requirements for these positions.

3. Removing responsibility details of the ORC was generally endorsed by the NRC in a letter from William T. Russell (NRC) dated October 25, 1993. The acceptance of this change was based on concluding that specific requirements were not necessary to be included in TS to meet 10 CFR 50.36(c)(5), which states: "Administrative controls are the provisions related to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." ORC responsibilities are maintained within plant procedures, consistent with the Quality Assurance Program commitments, adequately assures safe operation.

4. Section 50.55a of 10 CFR requires that IST of certain ASME Code Class 1, 2, and 3 pumps and valves be performed in accordance with the ASME OM Code and applicable addenda, except where alternatives have been authorized or relief has been requested by the licensee and granted by the Commission pursuant to paragraphs (a)(3)(i), (a)(3)(ii),or (f)(6)(i) of 10 CFR 50.55a. In accordance with 10 CFR 50.55a(f)(4)(ii), licensees are required to comply with the requirements of the latest edition and addenda of the ASME Code incorporated by reference in the regulations 12 months prior to the start of the subsequent 120-month IST program intervals. Accordingly, licensees whose subsequent 120-month (10-year) IST program interval began after November 22, 2000, are required to comply with the 1995 Edition with the 1996 Addenda of the ASME OM Code. Similarly, licensees whose 120-month (10-year) IST program interval began after October 28, 2003, are required to comply with the 1998 Edition through 2000 Addenda of the ASME OM Code. In accordance with 10 CFR 50.55a(f)(4)(iv), licensees may use portions of subsequent editions and addenda provided that all related requirements of the respective edition and addenda are met.

   In proposing alternatives or requesting relief, the licensee must demonstrate that: (1) the proposed alternatives provide an acceptable level of quality and safety; (2) compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety; or (3) conformance is impractical for the facility. Section 50.55a of 10 CFR authorizes the Commission to approve alternatives and to grant relief from ASME Code requirements upon making necessary findings. NRC guidance contained in Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," provides acceptable alternatives to ASME Code requirements. Further guidance is given in GL 89-04, Supplement 1, and NUREG-1482, "Guidance for Inservice Testing at Nuclear Power Plants."

Amendment 149 to PNPS TS, dated September 28, 1993, created Specification 3/4.13, "Inservice Code Testing." This specification was based on similarity to the previous BWR/5 Standard TS (NUREG-0123) Section 4.0.5.b. Since then, various administrative and editorial changes have been made to the Standard TS for the Inservice Testing Program that reflect changes to the regulations and guidance found acceptable to the Commission. Based on minimizing duplication with the Regulations, the current BWR/4 Standard TS, NUREG-1433, Revision 3, Specification 5.5.7 has eliminated explicit reference to ASME subsections and explicit reference to regulations governing relief from the Code.

5. The administrative clarifications proposed for TS 5.5.4, "Radioactive Effluent Controls Program," were specifically addressed as part of change TSTF-285, which was NRC approved on June 29, 1999.

Each of these changes have been incorporated into the most recent issued revision of the Standard TS NUREG-1433, Revision 3, and have been NRC reviewed and approved on other dockets as acceptable administrative changes with no adverse impact on the health and safety of the public.

4. Technical Analysis

The proposed changes (1) to remove the administrative reporting and restart authorization requirements that apply in the event of a Safety Limit violation; (2) to replace plant-specific titles with generic titles; (3) to remove responsibilities of the ORC; (4) to delete regulatory detail for the Inservice Code Testing Program; and (5) other administrative corrections and clarifications; are administrative with no technical change in requirements. As such, no specific regulatory requirements or guidance applies. Additionally, the changes are consistent with the latest revision of the BWR/4 Standard Technical Specifications, NUREG-1433 (Revision 3, dated 3/31/2004).

4.1 TS Section 2.2 provides notification, reporting, and restart requirements to be met in the event of a Safety Limit violation. TS Sections 2.2.1, 2.2.4, and 2.2.5, which are proposed for deletion, are addressed by the requirements of 10 CFR 50.36(c)(1)(i)(A). Furthermore, TS Section 2.2.1 is addressed by 10 CFR 50.72 and TS 2.2.4 is addressed by 10 CFR 50.73; however, the TS 2.2.4 30-day requirement to submit the Licensee Event Report (LER) is no longer consistent with the latest provisions of 10 CFR 50.73, which allow 60-day reporting. This change will correct that inconsistency.

Also proposed for deletion is TS 2.2.3, which directs notification of the Station Director, Vice President – Nuclear, and the Nuclear Safety Review and Audit Committee within 24 hours. Assurance of these administrative notifications is adequately controlled by plant procedures.

TS 2.2.2.A and 2.2.2.B will be renumbered to 2.2.1 and 2.2.2 because of the deletion of the above TS sections.

Removal of duplicative reporting requirements from the Technical Specifications results in simplification of the Technical Specifications and Bases and less administrative burden to track duplicative reporting requirements. Adequate administrative controls exist in administrative programs at Pilgrim for the identification and necessary reporting of safety limit violations in accordance with 10 CFR 50.36, 10 CFR 50.72 and 10 CFR 50.73.

In summary, the necessary notification, reporting, and restart requirements to be met in the event of a Safety Limit violation are adequately addressed by existing regulations and plant procedures. As such, these changes are administrative with no technical change in requirements.

4.2 Replacing plant-specific titles with generic titles, and including a TS commitment (in TS 5.2.1) to retain specific titles in the Final Safety Analysis Report (FSAR) of those personnel fulfilling the responsibilities does not eliminate any qualifications, responsibilities or requirements for these positions. Members of the plant staff assigned to these positions shall continue to meet or exceed the minimum qualifications required by TS 5.3, "Unit Staff Qualifications."

Any change of the relocated specifications in the FSAR will be strictly controlled in accordance with the provisions of 10 CFR 50.59, "Changes, tests, and experiments" to determine if the proposed changes will require prior NRC review and approval. Additionally, reporting of any changes to the NRC will be made in accordance with 10 CFR 50.71(e), "Maintenance of records, making of reports."

Additional administrative plant staff position clarifications outlined in Section 2, Proposed Changes, are also consistent with NUREG-1433, and are discussed below.

In TS 5.2.2.h, the requirement for an Operations Department management position to hold a senior reactor operator license is clarified to include the flexibility of the "operations manager or assistant operations manager." Since both positions are responsible for directing the licensed activities of licensed operators, there is no adverse impact to safe plant operations due to this change.

In TS 5.2.2.h, the discussion of the specific position titles of "Nuclear Watch Engineers" and "Nuclear Operations Supervisors" holding a senior reactor operator (SRO) license, and the "Nuclear Plant Operators" holding a reactor operator (RO) license is also eliminated. The generic requirements for SRO and RO on-shift positions are adequately addressed in TSs 5.2.2.b, 5.2.2.c, and 5.2.2.e, as well as 10 CFR 50.54(k), 50.54(l), and 50.54(m). Elimination of these plant-specific titles from this Section is consistent with the intent of replacing plant-specific titles with generic titles.

In TS 5.5.1, the required management level for approval of the changes to the ODCM is made more restrictive by replacing the "Chemistry and Radiological Department Managers" with "the plant manager." This change is made for consistency with NUREG-1433, replaces plant-specific titles with generic titles, and does not preclude the continued approvals of the Chemistry and Radiological Department Managers. As such, there is no adverse impact to safe plant operations due to this change.

In TS 5.7.1 and 5.7.2 reference to "health physics" personnel / supervision is replaced with "radiation protection" to more appropriately reflect the departmental responsibilities. The title case presentation of the "Radiation Protection Manager" in 5.7.1.c is made a generic (i.e., lower case) title "radiation protection manager" consistent with other changes to generic titles.

In TS 5.7.2, "the Nuclear Watch Engineer on duty" is replaced with "an SRO on duty." The NUREG-1433 presentation of "shift supervisor" suggests the equivalent Pilgrim position of NOS. The proposed change allows maintaining the current requirement for Nuclear Watch Engineer (i.e., shift manager) to retain this responsibility, but also allows for future procedure revision to assign this responsibility to the NOS if desired. Since there is no actual change to existing requirements, and the possible allowed future change is consistent with the standard TS, there is no adverse impact to safe plant operations due to this change.

In summary, the necessary qualifications, responsibilities or requirements for these positions are adequately addressed by existing regulations and regulatory controls imposed for future changes to the FSAR. As such, these administrative changes do not adversely impact the public health and safety.

4.3 The Operations Review Committee (ORC) responsibilities were relocated from the Pilgrim TS in Amendment 177 on July 31, 1998. However, two references to ORC review and approval responsibilities were overlooked for concurrent relocation.

TS 3.7.A.2.b, Footnote *, references ORC approval of the administrative controls used to intermittently open primary containment isolation valves closed to satisfy TS required actions. The corresponding allowance in NUREG-1433, TS 3.6.1.3, Actions Note 1, does not include any reference to approval authority for the administrative control. Also, TS 5.5.1.b specifies requirements for implementing licensee-initiated changes to the Offsite Dose Calculation Manual (ODCM), which include "review and acceptance by the Operations Review Committee." The corresponding requirement in NUREG-1433, Specification 5.5.1.b, does not specify the details of programmatic review(s) – only the final approval required by the plant manager.

In summary, the necessary ORC responsibilities are adequately addressed in licensee controlled documents, without explicit TS requirements, as previously approved by the NRC. As such, these changes are administrative with no technical change in requirements.

4.4 The proposed changes to the REFUELING INTERVAL definition and to TS 3/4.13.A reflect administrative changes only. References to ASME Section XI are revised to reflect current regulations and ASME OM Code and the PNPS specific program name, "Inservice Code Testing Program." The administrative reference to "safety and safety related" and "or equivalent" is corrected to match the regulation, which specifically addresses Code Class 1, 2, and 3 pumps and valves. Since the regulations of 10 CFR 50.55a already adequately enforce the requirements, eliminating detailed reference to the regulation, and explicit reference to regulations governing relief from the Code, the proposed Specification retains only the specific performance frequency definitions for Code terminology. These changes result in a Specification essentially equivalent to the BWR/4 Standard TS (NUREG-1433) Specification 5.5.7, "Inservice Testing Program." This change does not impact the April 30, 2004 NRC review of the PNPS 4th 10-Year Inservice Code Testing Program, which remains the basis for the current program implementation.

4.5 The following administrative changes involve no technical change and serve to enhance the consistency of the PNPS TS with the NUREG-1433 Standard TS for consistent use and application for the PNPS operating staff and NRC regulator:

1. TS 5.5.4.b, replace "10 CFR 20, Appendix B, Table 2, Column 2" with "ten times the concentration values in Appendix B, Table 2, Column 2, to 10 CFR 20.1001 – 20.2402." These values provide reasonable assurance that the levels of radioactive materials in bodies of water in unrestricted areas will result in exposures within (1) the Section II.A design objectives of appendix I to 10 CFR Part 50 and (2) restrictions authorized by 10 CFR 20.1301(e).

The existing PNPS TS 5.5.4.b, references the old Part 20.1 – 20.602, Appendix B, Table II (typographically presented as "Table 2"), as allowed by 10 CFR 20.1008. Current requirements for the content of TS concerning radioactive effluents are contained in 10 CFR 50.36a. 10 CFR 50.36a requires licensees to maintain control over radioactive material in gaseous and liquid effluents to unrestricted areas, produced during normal reactor operations, including expected occurrences, to levels that are as low as reasonably achievable (ALARA). For power reactors, Appendix I to 10 CFR Part 50 contains the numerical guidance to meet the ALARA requirement. The dose values specified in Appendix I of 10 CFR Part 50 are small percentages of the implicit limits in the old 10 CFR 20.106 and the explicit limits in 10 CFR 20.1301. As secondary controls, the instantaneous concentration release rates required by this TS were chosen by the NRC to help maintain annual average releases of radioactive material in gaseous and liquid effluents to within the dose values specified in Appendix I of 10 CFR Part 50. For the purposes of STS 5.5.4.b, 10 CFR Part 20 is used as a source of reference values only. These TS requirements allow operational flexibility, compatible with considerations of health and safety, which may temporarily result in release rates which, if continued for the calendar quarter, would result in radiation doses higher than specified in Appendix I of 10 CFR Part 50. However, these releases are within the implicit limits in the old 10 CFR Part 20.106 and the explicit limits in 10 CFR Part 20.1302, which references 10 CFR Part 20, Appendix B, concentrations. These referenced concentrations in the old 10 CFR Part 20 are specific values, which relate to an annual dose of 500 mrem. The liquid effluent radioactive effluent concentration limits given in Appendix B, Table 2, Column 2 to 10 CFR 20.1001 - 20.2402 are based on an annual dose of 50 mrem total effective dose equivalent. Since an instantaneous release concentration corresponding to a dose rate of 500 mrem/year has been acceptable as a TS limit for liquid effluents, which applies at all times to assure that the values in Appendix I of 10 CFR Part 50 are not likely to be exceeded, it is not necessary to reduce this limit by a factor of 10.

The use of effluent concentration values that are 10 times those listed in Appendix B, Table 2, Column 2 to 10 CFR 20.1001 - 20.2402 will not have a negative impact on the ability to continue to operate within the design objectives in Appendix I to 10 CFR 50. Thus, the change to STS 5.5.4.b maintains the same overall level of liquid effluent control while retaining the operational flexibility that exists with TS under the previous 10 CFR Part 20. This limitation (i.e., less than 10 times the concentration values...) provides reasonable assurance that the levels of radioactive materials in bodies of water in Unrestricted Areas will result in exposures within (1) the Section II.A design objectives of Appendix I to 10 CFR 50 and (2) restrictions authorized by 10 CFR 20.1301(e).

2. TS 5.5.4.e, delete "and projected dose" from the current sentence and add second sentence "Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days." This change is an administrative clarification approved by the NRC in TSTF-308, and presented in NUREG-1433, Revision 3. This avoids possible misinterpretation that projecting doses for the current calendar quarter, as well as for the current calendar year, are required every 31 days. This clarification does not reflect any change in requirements or procedures.

3. TS 5.5.4.g, reword "effluents to areas beyond" adding clarifying phrases to read "effluents from the site boundary to areas at or beyond." This change is an administrative clarification approved by the NRC in TSTF-258, and presented in NUREG-1433, Revision 3. This clarification does not reflect any change in requirements or procedures.

4. TS 5.5.4.j, add "beyond the site boundary" after "member of the public." This change is an administrative clarification approved by the NRC in TSTF-258, and presented in NUREG-1433, Revision 3. This clarification does not reflect any change in requirements or procedures.

4.6 The following editorial changes, corrections, or clarifications involve no technical change and serve to clarify the use and application of TS for the operating staff:

1. Remove the ** note from TS 3.7.A.5 since it was only applicable through 1998.

2. TS 3.5.A.5 misspelled word "rector" is corrected to "reactor." During the processing of License Amendment 200, dated April 22, 2003, "reactor" was misspelled in TS 3.5.A.5.

3. TS 3.8.2, on page 3/4.8-2, has two periods in the APPLICABILITY statement. The extra period at the end of the sentence is removed and the misspelled word "CHANNEl" in TS 4.8.2.3 is corrected to "CHANNEL." These typographical errors were inadvertently introduced during License Amendment 177, dated July 31, 1998.

4. TS 4.9.A.c, on page 3/4.9-2, has a typo in the last sentence reference to "4.9.A.1.b.1," which is corrected to "4.9.A.1.b.2." This typographical error makes incorrect reference to the Specification, which was inadvertently introduced during the License Amendment 179, dated December 18, 1998.

5. TS 3.9.B.2 last sentence on page 3/4.9-4, "and the NRC is notified within one (1) hour as required by 10 CFR 50.72" is deleted. This is adequately required by the 10 CFR 50.72 and applicable plant procedure implementation of the regulation.

6. TS 4.9.A.4.b, on page 3/4.9-4 correct the abbreviation for the unit hertz to "Hz" (without subscripting the "z"). This was a typographical error only.

7. TS 3.11.C.2, on page 3/4/11-3, reference to "Table 3.3.1" is revised to correctly reference "Table 3.3-1." This was a typographical error only.

Elimination of duplicative regulatory reporting requirements will avoid future, and eliminate existing, inconsistent or conflicting regulatory requirements. The proposed use of generic personnel titles will allow Pilgrim the flexibility to revise position titles while still meeting the appropriate personnel qualifications required by TS 5.3, "Unit Staff Qualifications." Additionally, the use of generic personnel titles will reduce and/or eliminate the need for future license amendments related to revised position titles. Administrative corrections and enhancements serve to clarify the use and application of TS for the operating staff.

These proposed changes are considered administrative with no adverse impact on the public health and safety.

5. <u>Regulatory Safety Analysis</u>

5.1 <u>No Significant Hazards Consideration</u>

Entergy Nuclear Operations, Inc. (Entergy) is proposing to modify the Pilgrim Technical Specifications (TS): (1) to remove the administrative reporting and restart authorization requirements that apply in the event of a Safety Limit violation; (2) to replace plant-specific titles with generic titles; (3) to remove the remaining responsibilities of the Operations Review Committee (ORC); (4) to delete regulatory detail for the Inservice Code Testing Program; and (5) to make other administrative corrections or clarifications.

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No. The proposed change is administrative in nature and does not involve the modification of any plant equipment or affect basic plant operation. There is no impact to any accident previously evaluated.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No. The proposed change does not involve any physical alteration of plant equipment and does not change the method by which any safety-related system performs its function. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. The methods governing plant operation and testing remain consistent with current safety analysis assumptions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No. The proposed change represents the relocation of specific Technical Specification requirements, based on regulatory guidance and previously approved changes for other stations or deletion of detail redundant to regulations or no longer applicable (i.e., expired one-time exceptions). The proposed change is administrative in nature, does not negate or revise any existing requirement, and does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by requirements that are retained, but relocated from the Technical Specifications. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 <u>Environmental Consideration</u>

A review has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

6. Precedents

The NRC has approved similar changes (e.g., changes adopting TSTF-5, TSTF-65, TSTF-258, and TSTF-308) in a number of amendments. Examples include Waterford Steam Electric Station, Unit 3, amendment No. 188 dated April 3, 2003, Cooper Nuclear Station amendment No. 200 dated July 15, 2003, Callaway Plant, Unit 1, amendment No. 155 dated June 3, 2003, and Calvert Cliffs Nuclear Power Plant, amendment No. 259, dated July 15, 2003.

7. References

1. NUREG-1433, Rev. 3, "Standard Technical Specifications, General Electric Plants, BWR/4."

2. 10 CFR 50.36, "Technical specifications."

3. Pilgrim Nuclear Power Station amendment No. 177, dated July 31, 1998.

4. Waterford Steam Electric Station, Unit 3, amendment No. 188 dated April 3, 2003.

5. Cooper Nuclear Station amendment No. 200 dated July 15, 2003.

6. Callaway Plant, Unit 1, amendment No. 155 dated June 3, 2003.

7. Calvert Cliffs Nuclear Power Plant, amendment No. 259, dated July 15, 2003.

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION AND BASES

CHANGES (MARK-UP)

Inservice Code Testing Program

REFUELING INTERVAL

REFUELING INTERVAL applies only to ~~ASME Code, Section XI IWP and IWV~~ surveillance tests. For the purpose of designating frequency of these code tests, a REFUELING INTERVAL shall mean at least once every 24 months.

REFUELING OUTAGE

REFUELING OUTAGE is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant after that refueling. For the purpose of designating frequency of testing and surveillance, a REFUELING OUTAGE shall mean a regularly scheduled outage; however, where such outages occur within 11 months of completion of the previous REFUELING OUTAGE, the required surveillance testing need not be performed until the next regularly scheduled outage.

SAFETY LIMIT

The SAFETY LIMITS are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences, but it indicates an operational deficiency subject to regulatory review.

SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY means that the reactor building is intact and the following conditions are met:

e

1. At least one door in each access opening is closed.
2. The standby gas treatment system is operable.
3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

SIMULATED AUTOMATIC ACTUATION

SIMULATED AUTOMATIC ACTUATION means applying a simulated signal to the sensor to actuate the circuit in question.

SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of: (a) a test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals; (b) the testing of one system, subsystem, train or other designated components at the beginning of each subinterval.

SURVEILLANCE FREQUENCY

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

The SURVEILLANCE FREQUENCY 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 schedule and

2.0 SAFETY LIMITS

2.1 Safety Limits

2.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% of rated core flow:

THERMAL POWER shall be ≤ 25% of RATED THERMAL POWER.

2.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% of rated core flow:

MINIMUM CRITICAL POWER RATIO shall be ≥ 1.06.

2.1.3 Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the normal active fuel zone.

2.1.4 Reactor steam dome pressure shall be ≤ 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 Safety Limit Violation

With any Safety Limit not met the following actions shall be met:

~~2.2.1~~ ~~Within one hour notify the NRC Operations Center in accordance with 10CFR50.72.~~

~~2.2.2~~ Within two hours

2.2.1 ~~A.~~ Restore compliance with all Safety Limits, and

2.2.2 ~~B.~~ Insert all insertable control rods.

~~2.2.3~~ ~~The Station Director and Senior Vice President - Nuclear and the Nuclear Safety Review and Audit Committee (NSRAC) shall be notified within 24 hours.~~

~~2.2.4~~ ~~A Licensee Event Report shall be prepared pursuant to 10CFR50.73. The Licensee Event Report shall be submitted to the Commission, the Operations Review Committee (ORC), the NSRAC and the Station Director and Senior Vice President - Nuclear within 30 days of the violation.~~

~~2.2.5~~ ~~Critical operation of the unit shall not be resumed until authorized by the Commission.~~

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

A. Core Spray and LPCI Systems (Cont)

4. During Run, Startup, and Hot Shutdown Modes with the LPCI system inoperable, restore the LPCI system to Operable status within 7 days and maintain both core spray systems and the diesel generators Operable. Otherwise, be in at least Cold Shutdown within 24 hours.

5. Two low pressure injection/spray subsystems shall be Operable during Cold Shutdown and Refuel Modes unless the ~~rector~~ reactor head is removed, the spent fuel pool gates are removed, and water level is at greater than or equal to elevation 114 foot, except as specified in 3.5.A.6.

6. During Cold Shutdown and Refuel Modes unless the reactor head is removed, the spent fuel pool gates are removed, and water level is at greater than or equal to elevation 114 foot:

   a. With one of the required low pressure injection/spray subsystems inoperable, restore the inoperable required low pressure injection/spray subsystem to Operable status within 4 hours. Otherwise, take immediate action to suspend activities with potential for draining the reactor vessel.

   b. With both of the required low pressure injection/spray subsystems inoperable, take immediate action to suspend activities with potential for draining the reactor vessel and restore 1 low pressure injection/spray subsystem to Operable status within 4 hours. Otherwise, take immediate action to restore secondary containment and one standby gas treatment system to Operable status and to restore isolation capability in each required secondary containment penetration flow path not isolated.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

A. Core Spray and LPCI Systems (Cont)

1. c. Motor Operated Valve Operability — As Specified in 3.13

   d. Core Spray Header Δp Instrumentation

Check	Once/day
Calibrate	Once/3 months
Test Step	Once/3 months

2. This section intentionally left blank

3. LPCI system testing shall be as follows:

a. Simulated Automatic Actuation Test	Once/ Operating Cycle
b. Pump Operability.	When tested as specified in 3.13, verify that each LPCI pump delivers 4800 GPM at a head across the pump of at least 380 ft.
c. Motor Operated Valve Operability	As Specified in 3.13

3.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

5. All containment isolation check valves are operable or at least one containment isolation valve in each line having an inoperable valve is secured in the isolated position**

Primary Containment Isolation Valves

2. b. In the event any automatic Primary Containment Isolation Valve becomes inoperable, at least one containment isolation valve in each line having an inoperable valve shall be deactivated in the isolated condition. (This requirement may be satisfied by deactivating the inoperable valve in the isolated condition. Deactivation means to electrically or pneumatically disarm, or otherwise secure the valve.)*

* Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under ~~ORC-approved~~ administrative controls.

~~** Check valve 30-CK-432 will be considered operable until reverse flow testing is performed no later than the 1998 maintenance outage.~~

4.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

4. Combined main steam lines: 46 scfh @ 23 psig.

where P_a = 45 psig
L_a = 1.0% by weight of the contained air @ 45 psig for 24 hrs.

Primary Containment Isolation Valves

2. b. 1. The primary containment isolation valves surveillance shall be performed as follows:

a. At least once per operating cycle the operable primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

b. Test primary containment isolation valves:

1. Verify power operated primary containment isolation valve operability as specified in 3.13.

2. Verify main steam isolation valve operability as specified in 3.13.

LIMITING CONDITIONS FOR OPERATION

3.8 PLANT SYSTEMS (CONT)

2. Mechanical Vacuum Pump Isolation Instrumentation

LCO 3.8.2

Four channels of the Main Steam Line Radiation Monitoring System Radiation - High function for the mechanical vacuum pump shall be OPERABLE

APPLICABILITY:

Whenever any main steam isolation valve is open with steam flowing.

ACTIONS:

---NOTE---
Separate Condition Entry is allowed for each channel.

A. One or more required channels inoperable.

1. Restore channel to OPERABLE status within 24 hours.

OR

2. ---NOTE---
Not Applicable if inoperable channel is the result of an inoperable isolation valve.

Place channel or associated trip system in trip within 24 hours.

B. Required Action and associated Completion Time of Condition A not met.

OR

Mechanical vacuum pump isolation capability not maintained.

1. Isolate mechanical vacuum within 12 hours.

OR

2. Isolate Main Steam Lines within 12 hours.

OR

3. Be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.8 PLANT SYSTEMS (CONT)

2. Mechanical Vacuum Pump Isolation Instrumentation

---NOTE---
When a channel is placed in an inoperable status solely for the performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains mechanical vacuum pump isolation capacity.

---

1. Perform a CHANNEL CHECK every 12 hours.

2. Calibrate the trip units every 92 days.

3. Perform a CHANNEL CALIBRATION every 24 months. The allowable trip value shall be ≤ 5.5 x normal background.

4. Perform a LOGIC SYSTEM FUNCTIONAL TEST including isolation valve actuation every 24 months.

LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

A. Auxiliary Electrical Equipment (Cont)

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

A. Auxiliary Electrical Equipment Surveillance (Cont)

1. Verifying de-energization of the emergency buses and load shedding from the emergency buses.

2. Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency buses with permanently connected loads, energizes the auto-connected emergency loads through the load sequence, and operates for ≥ 5 minutes while its generator is loaded with the emergency loads.

   During performance of this surveillance verify that HPCI and RCIC inverters do not trip.

   The results shall be logged.

c. Once per operating cycle with the diesel loaded per 4.9.A.1.b verify that on diesel generator trip, secondary (offsite) AC power is automatically connected within 11.8 to 13.2 seconds to the emergency service buses and emergency loads are energized through the load sequencer in the same manner as described in 4.9.A.1.b.~~1~~. 2

   The results shall be logged.

3.9 AUXILIARY ELECTRICAL SYSTEM (Cont)

B. Operation with Inoperable Equipment

Whenever the reactor is in Run Mode or Startup Mode with the reactor not in a Cold Condition, the availability of electric power shall be as specified in 3.9.B.1, 3.9.B.2, 3.9.B.3, 3.9.B.4, and 3.9.B.5.

1. From and after the date that incoming power is not available from the startup or shutdown transformer, continued reactor operation is permissible under this condition for:

   a. 3 days with the startup transformer inoperable

   or

   b. 7 days with the shutdown transformer inoperable

   During this period, both diesel generators and associated emergency buses must remain operable.

2. From and after the date that incoming power is not available from both startup and shutdown transformers, continued operation is permissible, provided both diesel generators and associated emergency buses remain operable, all core and containment cooling systems are operable, and reactor power level is reduced to 25% of design ~~and the NRC is notified within one (1) hour as required by 10CFR50.72~~.

3. From and after the date that one of the diesel generators or associated emergency bus is made or found to be inoperable for any reason, continued reactor operation is permissible in accordance with Specifications 3.4.B.1, 3.5.F.1, 3.7.B.1.c, 3.7.B.1.e, 3.7.B.2.c, and 3.7.B.2.e if Specification 3.9.A.1 and 3.9.A.2.a are satisfied.

4.9 AUXILIARY ELECTRICAL SYSTEM (Cont)

Auxiliary Electrical Equipment Surveillance (Cont)

3 Emergency 4160V Buses A5-A6 Degraded Voltage Annunciation System.

   a. Once each operating cycle, calibrate the alarm sensor.

   b. Once each 31 days perform a channel functional test on the alarm system.

   c. In the event the alarm system is determined inoperable under 3.b above, commence logging safety related bus voltage every 30 minutes until such time as the alarm is restored to operable status.

4. RPS Electrical Protection Assemblies

   a. Each pair of redundant RPS EPAs shall be determined to be operable at least once per 6 months by performance of an instrument functional test.

   b. Once per 18 months each pair of redundant RPS EPAs shall be determined to be operable by performance of an instrument calibration and by verifying tripping of the circuit breakers upon the simulated conditions for automatic actuation of the protective relays within the following limits:

Overvoltage	$\leq$ 132 volts
Undervoltage	$\geq$ 108 volts
Underfrequency	$\geq$ 57 Hz

LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLY (Cont)

C. Minimum Critical Power Ration MCPR (Cont'd)

2. The operating limit MCPR values as a function of the τ are given in Table 3.3.1 of the Core Operating Limits Report where τ is given by specification 4.11.C.2. 3.3-1

SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLY (Cont)

C. Minimum Critical Power Ration MCPR (Cont'd)

b. The average scram time to dropout of Notch 34 is determined as follows:

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

Where: an n - number of surveillance tests performed to date in the cycle.

N_i - number of active control rods measured in the i^{th} surveillance test.

τ_i - average scram time to dropout of Notch 34 of all rods measured in the i^{th} surveillance test.

c. The adjusted analysis mean scram time (τ_B) is calculated as follows:

$$\tau_B = \mu + 1.65 \left[\frac{N_1}{\sum_{i=1}^{n} N_i} \right]^{1/2} \sigma$$

Where:

μ - mean of the distribution for average scram insertion time to dropout of Notch 34, 0.937 sec.

N_1 - total number of active control rods at BOC during the first surveillance test.

σ - standard deviation of the distribution for average scram insertion time to the dropout of Notch 34, 0.021 seconds.

LIMITING CONDITIONS FOR OPERATION

3.13 INSERVICE CODE TESTING

Applicability:

Applies to ASME Code Class 1, 2 and 3 ~~or equivalent~~ pumps and valves.

Objective:

To assure the operational readiness of ASME Code Class 1, 2, and 3 ~~(Safety Related) or equivalent (important to safety)~~ pumps and valves.

Specification:

A. Inservice Code Testing of Pumps and Valves

1. Based on the Facility Commercial Operation Date, Inservice Code Testing of ~~Safety and safety-related~~ ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ~~ASME Boiler and Pressure Vessel Code, Section XI "Rules for Inservice Testing of Nuclear Power Plant Components" Subsections IWP and IWV as required by 10CFR50.55a(f), except where specific relief has been granted by the NRC pursuant to 10CFR50.55a(f)(6)(i).~~ Inservice Code Testing Program

SURVEILLANCE REQUIREMENTS

4.13 INSERVICE CODE TESTING

Applicability:

Applies to the periodic testing requirements of ASME Code Class 1, 2 and 3 ~~or equivalent~~ pumps and valves.

Objective:

To assess the operational readiness of ~~safety and safety-related~~ ASME Code Class 1, 2, and 3 pumps and valves by performance of inservice tests.

Specification:

A. Inservice Code Testing of Pump and Valves

~~1. Inservice Code Testing activities shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50.55a(f), with the exemptions and alternate testing that have been approved by the NRC pursuant to 10CFR50.55a(f)(6)(i). These exemptions and alternate testing are included in the PNPS Inservice Testing Program.~~

~~2.~~ ~~Test Frequencies for Code~~ The ASME OM Code Terminology ~~when performing~~ for Inservice Test activities is as follows:

Code Terminology	Frequencies
Weekly	7 Days
Monthly	31 Days
Quarterly or 3 Mths	92 Days
Semiannually/ 6 Mths	184 Days
9 Months	276 Days
Yearly/Annually	366 Days
Biannual/2 Yrs	732 Days

LIMITING CONDITIONS FOR OPERATION

3.13 INSERVICE CODE TESTING

SURVEILLANCE REQUIREMENTS

4.13 INSERVICE CODE TESTING

3. The provisions in Definitions (1.0) for REFUELING INTERVAL, SURVEILLANCE FREQUENCY, and SURVEILLANCE INTERVAL are applicable to Code testing and to the above frequencies for performing Code testing activities.

4. Performance of Code testing shall be in addition to other specified Surveillance Requirements.

5. Nothing in the ~~ASME Boiler and Pressure Vessel Code~~ shall supersede the requirements of Technical Specifications.

Inservice Testing Program Code

BASES:

3.13 and 4.13 Inservice Code Testing

Inservice Code Testing Program.

The Limiting Conditions for Operation establishes the requirement that inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ~~periodically updated edition of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10CFR50, Section 50.55a(f).~~

The detailed procedures for testing of pumps and valves are documented in the PNPS Inservice Code Testing Program.

This specification includes a clarification of the frequencies for performing the testing activities required by ~~Section XI of~~ the ASME OM ~~Boiler and Pressure Vessel~~ Code and applicable Addenda. This clarification is provided to ensure consistency in Surveillance Frequencies throughout the Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice testing activities.

Under the terms of this Specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME OM ~~Boiler and Pressure Vessel~~ Code and applicable Addenda. ~~For example:~~

- ~~Technical Specifications require components to be declared operable prior to entry into an operational mode. The ASME B&PV Code provision which allows pumps and valves to be tested up to one week after return to normal operation is superseded (and not allowed) by the more restrictive requirements of Technical Specifications.~~
- ~~The allowance for a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable is superseded (and not allowed) by the more restrictive Technical Specification definition of operability which does not allow a grace period.~~

BASES:

3/4.4 STANDBY LIQUID CONTROL SYSTEM

Surveillance Requirements

3.4.C

If both SLC subsystems are inoperable for reasons other than condition 3.4.A, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed completion time of 8 hours is considered acceptable given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor.

3.4.D

If any action and associated completion time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to Hot Shutdown within 12 hours. The allowed completion time of 12 hours is reasonable, based on operating experience, to reach Hot Shutdown from full power conditions in an orderly manner and without challenging plant systems.

4.4.1

Demonstrating that each SLC System pump develops a flow rate of 39 gpm at a minimum system head of 1275 psig ensures that pump performance is acceptable during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal Code performance. Testing the pumps and valves in accordance with the Inservice Testing Program [ASME B&PV Code Section XI (Articles IWP and IWV, except where specific relief is granted)] adequately assesses component operational readiness.

4.4.2

This Surveillance ensures that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 48 months at alternating 24 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance were performed at power. Various components of the system are individually tested periodically, thus making more frequent testing of the entire system unnecessary.

4.4.3

This Surveillance verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

BASES

ACTIONS

3.5.A.6.b

During Cold Shutdown and Refuel Modes unless the reactor head is removed, the spent fuel pool gates are removed, and water level is at greater than or equal to elevation 114 foot with both of the required ECCS injection/spray subsystems inoperable, all coolant inventory makeup capability may be unavailable. Therefore, actions must immediately be initiated to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended. One ECCS injection/spray subsystem must also be restored to Operable status within 4 hours.

If at least one low pressure ECCS injection/spray subsystem is not restored to Operable status within the 4 hour Completion Time, additional actions are required to minimize any potential fission product release to the environment. This includes ensuring secondary containment is Operable; one standby gas treatment subsystem is Operable; and secondary containment isolation capability (i.e., one isolation valve and associated instrumentation are Operable or other acceptable administrative controls to assure isolation capability) in each associated penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases. Operability may be verified by an administrative check, or by examining logs or other information, to determine whether the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the Operability of the components. If, however, any required component is inoperable, then it must be restored to Operability status. In this case, Surveillance may need to be performed to restore the component to Operable status. Actions must continue until all required components are Operable.

The 4 hour completion time to restore at least one low pressure injection/spray subsystem to Operable status ensures that prompt action will be taken to provide the required cooling capacity or to initiate actions to place the plant in a condition that minimizes any potential fission product release to the environment.

SURVEILLANCE REQUIREMENTS

the Inservice Code Testing (IST) Program

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. To increase the availability of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated valves are tested in accordance with ASME B&PV Code, Section XI (IWP and IWV, except where specific relief is granted) to assure their operability. The frequency and methods of testing are described in the PNPS IST

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

program. The PNPS IST Program is used to assess the operational readiness of pumps and valves that are ~~safety related or important to safety.~~ ASME Code Class 1, 2 and 3 When components are tested and found inoperable the impact on system operability is determined, and corrective action or Limiting Conditions of Operation are initiated. A simulated automatic actuation test once each cycle combined with code inservice testing of the pumps and valves is deemed to be adequate testing of these systems.

The surveillance requirements provide adequate assurance that the core and containment cooling systems will be operable when required.

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

BASES

SURVEILLANCE REQUIREMENTS

SR 4.5.B.1.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR suppression pool cooling mode flow path provides assurance that the proper flow path exists for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR suppression pool cooling mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the subsystem is a manually initiated system. This frequency has been shown to be acceptable based on operating experience.

SR 4.5.B.1.2

and the system will pass the required flow rate

Verifying that each RHR pump ~~develops a flow rate ≥ 5100 gpm (Ref. 1)~~ while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that pump performance has not degraded~~during the cycle~~. ~~Flow is a normal test of centrifugal pump performance required by ASME Code, Section XI (Ref. 2). This test confirms one point on the pump design curve, and the results are indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance.~~ The frequency of this SR is in accordance with the Inservice Testing Program, Specification 3/4.13.

REFERENCES

1. FSAR, Section 14.5.

2. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~ Title 10 Code of Federal Regulations Part 50.55a

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS
BASES

REFERENCES

1. FSAR, Section 4.8
2. FSAR, Section 14.5.
~~3. ASME, Boiler and Pressure Vessel Code, Section XI.~~
~~Title 10 Code of Federal Regulations Part 50.55a~~

3. ASME OM Code

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS
BASES

SURVEILLANCES

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. To increase the availability of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The ASME Code Class pumps and motor operated valves are tested in accordance with the Inservice Code Testing (IST) Program ~~ASME B&PV Code, Section XI (IWP and IWV, except where specific relief is granted)~~ to assure their operability. The frequency and methods of testing are described in the PNPS IST program. The PNPS IST Program is used to assess the operational readiness of ASME Code Class pumps and valves ~~that are safety related or important to safety.~~ When components are tested and found inoperable the impact on system operability is determined, and corrective action or Limiting Conditions of Operation are initiated. A simulated automatic actuation test once each cycle combined with code inservice testing of the pumps and valves is deemed to be adequate testing of these systems.

The surveillance requirements provide adequate assurance that the core and containment cooling systems will be operable when required.

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

3/4.5.D. BASES Reactor Core Isolation Cooling (RCIC) System

BACKGROUND

The RCIC is designed to provide makeup to the nuclear system as part of the planned operation for periods when the normal heat sink is unavailable. The Station Nuclear Safety Operational Analysis, FSAR Appendix G, shows that RCIC also serves as redundant makeup system on total loss of all offsite power in the event that HPCI is unavailable. In all other postulated accidents and transients, the ADS provides redundancy for the HPCI.

SPECIFICATION

The requirement that RCIC be operable when reactor coolant temperature is greater than 365°F is included in Specification 3.5.D.1 to clarify that RCIC need not be operable during certain testing (e.g., reactor vessel hydro testing at high reactor pressure and low reactor coolant temperature). 365°F is approximately equal to the saturation steam temperature at 150 psig.

ACTION

Based on this and judgments on the reliability of the HPCI system, an allowable repair time of 14 days is specified.

SURVEILLANCES

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. To increase the availability of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The ASME Code Class pumps and motor operated valves are tested in accordance with the Inservice Code Testing (IST) Program ~~ASME B&PV Code, Section XI (IWP and IWV, except where specific relief is granted)~~ to assure their operability. The frequency and methods of testing are described in the PNPS IST program. The PNPS IST Program is used to assess the operational readiness of ASME Code Class pumps and valves~~, that are safety related or important to safety.~~ When components are tested and found inoperable the impact on system operability is determined, and corrective action or Limiting Conditions of Operation are initiated. A simulated automatic actuation test once each cycle combined with code inservice testing of the pumps and valves is deemed to be adequate testing of these systems.

The surveillance requirements provide adequate assurance that the core and containment cooling systems will be operable when required.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The ~~Station Director~~ plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

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

5.1.2 The ~~Nuclear Operations~~ Control room Supervisor (~~NOS~~ CRS) shall be responsible for the control room command function. During any absence of the ~~NOS~~ from the control room while the unit is in an operational mode other than Cold Shutdown or Refueling, 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 ~~NOS~~ from the control room while the unit is in Cold Shutdown or Refueling, an individual with an active SRO license or Reactor Operator (RO) license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

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

a. INSERT ~~Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions,~~ or in equivalent forms of documentation. These requirements shall be documented in the Pilgrim Station Final Safety Analysis Report (FSAR);

b. The ~~Station Director~~ plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

c. A specified corporate officer ~~The Vice President - Operations~~ for Pilgrim 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; and

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. A non-licensed operator shall be on site when fuel is in the reactor and an additional non-licensed operator shall be assigned when the reactor is in an operational mode other than Cold Shutdown or Refueling.

(continued)

INSERT page 5.0-2

..., including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, ...

5.2 Organization

5.2.2 Unit Staff (continued)

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 in an operational mode other than Cold Shutdown or Refueling, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.

c. At least two licensed ROs shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips.

d. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.i 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 minimum requirements.

e. Higher grade licensed operators may take the place of lower grade licensed or unlicensed personnel.

f. An individual qualified in radiation protection procedures 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.

g. The amount of overtime worked by unit staff members performing safety-related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

or assistant operations manager

h. The operations ~~Department~~ Manager, ~~Nuclear Watch Engineers, and Nuclear Operations Supervisors~~ shall hold a Senior Reactor Operator License. ~~The Nuclear Plant Operators shall hold a Reactor Operator License.~~

An individual

unit operations shift crew

i. ~~The Shift Control Room Engineer (SCRE)~~ shall provide advisory technical support to the ~~Nuclear Operations Supervisor(NOS)~~ in the areas of engineering and accident assessment. ~~In addition, the SCRE~~ shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. ~~A Shift Control Room Engineer~~ with a Senior Reactor Operator license may simultaneously serve ~~as SCRE and SRO.~~

This individual

a required SRO position

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented and maintained.

5.5.1 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 the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release, reports required by Specification 5.6.2 and Specification 5.6.3.

Licensee initiated changes to the ODCM:

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

1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s); and

2. a determination that the change(s) maintain the levels 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;

b. Shall become effective after ~~review and acceptance by the Operations Review Committee and~~ the approval of the ~~Chemistry and Radiological Department~~ plant Manager~~s~~; and

c. 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 in 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 (i.e., month and year) the change was implemented.

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

ten times the concentration values in

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~~; to 10CFR20.1001 - 20.2402

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

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

INSERT

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

from the site boundary

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas at or beyond the site boundary conforming to the following:

1. For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

2. For Iodine-131, Iodine-133, Tritium, and all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

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

(continued)

INSERT page 5.0-8

.... . Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

5.5 Program and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

i. 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 > 8 days in gaseous effluents released to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR Section C.3.4.1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 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 require either of the following:

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

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

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

d. Proposed changes that meet the criteria of Specification 5.5.6b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 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~~ radiation protection personnel) 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 Protection Manager in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, 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 Nuclear Watch Engineer~~ an SRO on duty or ~~health physics~~ radiation protection 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

(Continued)

ATTACHMENT 2

LIST OF REGULATORY COMMITMENTS

List of Regulatory Commitments

The following table identifies those actions committed to by Pilgrim in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

REGULATORY COMMITMENT	DUE DATE
Relocate to FSAR plant-specific title and indicate relationship to generic titles used in the Specifications.	Within 60 days of license amendment approval.