PORTLAND GENERAL ELECTRIC COMPANY
EUGENE WATER & ELECTRIC BOARD
AND
PACIFIC POWER & LIGHT COMPANY

TROJAN NUCLEAR PLANT

Operating License NPF-1
Docket 50-344
License Change Application 99

This License Change Application requests modification of the Technical Specifications contained in Appendix A to Operating License NPF-1. In order to maintain compliance with NRC regulations, changes are proposed to: (a) revise the reactor vessel material irradiation surveillance schedule, (b) include additional Containment isolation valves, and (c) revise references to peak Containment pressures.

PORTLAND GENERAL ELECTRIC COMPANY

for

Bart D. Withers
Vice President
Nuclear

Subscribed and sworn to before me this 27th day of January 1984.

Notary Public of Oregon

My Commission Expires: August 9, 1987

LICENSE CHANGE APPLICATION 99

The following changes to Appendix A of Facility Operating License NPF-1 are requested (proposed replacement pages are provided as Attachment 1):

1. Page 3/4 4-27. The capsule withdrawal schedule in 10 CFR 50, Appendix H, was changed to require compliance with ASTM E 185-82 for capsules withdrawn after July 26, 1983. The withdrawal schedule in ASTM E 185-82 is based on effective full power years (EFPY) rather than calendar years as required by the current Technical Specification. Westinghouse Topical Report WCAP-9469, "Analysis of Capsule U From Portland General Electric Company Trojan Reactor Vessel Radiation Surveillance Program", provided a recommended removal schedule for the remaining capsules based on ASTM E 185-79 requirements. The withdrawal schedule requirements of ASTM E 185-82 did not change from the 1979 version of this code, and therefore the withdrawal schedule provided is in compliance with ASTM E 185-82.

2. Page 3/4 6-20. This change adds the isolation valves in the inner-door seal leak test lines of the personnel air locks to Table 3.6-1 of Appendix A to the Operating License. These valves are SV-6991 and SV-6992. Each valve is subjected to periodic Type C leakage rate testing in accordance with 10 CFR 50, Appendix J; therefore, these valves should not be annotated as being excluded from Type C leakage rate testing.

3. Page 3/4 6-24. In Table 3.6-1 of Technical Specification 3/4.6.3 under the Manual Valves section, four isolation valves (10610A, 10610B, 10611A, and 10611B) in the integrated leakage rate test lines have been added. This change also includes two isolation valves (LD-001 and LD-003) in the inner-door seal leak test lines of the personnel air locks. Each valve is subjected to periodic Type C leakage rate testing.

4. Page B 3/4 4-5. The heatup and cooldown curves shown in Figures 3.4-2 and 3.4-3 are applicable for the first 15 EFPY rather than 12 EFPY as stated in Paragraphs 3 and 4 on this page.

5. Page B 3/4 4-9. In accordance with the change to 10 CFR 50, Appendix H, ASTM E 185-82 is now the accepted version of this standard. The reference to ASTM E 185-73 as the version for removing and evaluating the capsule specimens is

therefore being changed to ASTM E 185-82. Additionally, in Paragraph 3, Table 4.4-3 is incorrectly referenced and is being changed to read Table 4.4-5.

6. Page B 3/4 6-2. This License Change Application revises Basis 3/4.6.1.4 (Internal Pressure), which references the peak Containment pressures evaluated for an initial positive Containment pressure of +1.6 psig, as a result of recent analyses for initial positive Containment pressure of +2.0 psig. The ambiguity in expressing peak pressures in psig will be clarified in terms of the assumptions used in recent analyses.

7. Page B 3/4 6-3. In Technical Specification Basis 3/4.6.1.6 (Containment Structural Integrity), the inappropriate and incorrect reference to the maximum peak pressure analyzed for a DBA LOCA has been deleted.

REASON FOR CHANGE

On Friday, May 27, 1983, the NRC published in the Federal Register a Final Rule amending the fracture toughness requirements for light water nuclear power reactors and the requirements for reactor vessel material surveillance programs. These rule changes affected 10 CFR 50, Sections 50.12, 50.55(a), 50.60, and Appendices G and H. As a result of the change to 10 CFR 50, Appendix H, Technical Specification Table 4.4-5 must be revised to reflect the schedule changes for specimen capsule removal. Additionally, the Bases for Technical Specification 4.4.9.1 are being revised in accordance with the changes made to Appendices G and H and to correct typographical errors.

Piping used to test the inner-door seals in the 45-ft and 93-ft elevation personnel air locks of Containment can, as a result of a seismic event and failure of the outer seal on the inner-door, provide a direct path from Containment to the outside atmosphere. Therefore, the valves on these lines should be considered Containment isolation valves and should be leakage rate tested in accordance with 10 CFR 50, Appendix J. When the Technical Specifications were originally written, these valves were not installed.

The isolation valves in the integrated leakage rate test lines were considered not to be Containment isolation valves when the Technical Specifications were originally written. However, since 1980 they have been subjected to Type C local leakage rate tests, and they are listed in Updated FSAR Table 6.2-1, "Containment Isolation Barriers".

A computer modeling error, as described in LER 82-07, invalidated the DBA Containment analyses described in Technical Specification Basis 3/4.6.1.4. Appropriate changes are proposed to reflect recent anslyses which provide new bases for the limits of Technical Specification 3/4.6.1.4.

The current wording is ambiguous regarding the Containment design basis and reference pressures for the DBA analyses. The proposed changes clarify Basis 3/4.6.1.4.

Technical Specification Basis 3/4.6.1.6 (Containment Structural Integrity) includes an inappropriate and incorrect reference to the maximum peak pressure analyzed for a DBA LOCA. Structural integrity is required not only for a DBA LOCA but, in addition, extreme environments such as seismic activity and floods. The value of 59.3 psig is superseded by recent internal pressures allowed under Technical Specification Basis 3/4.6.1.4.

SAFETY/ENVIRONMENTAL EVALUATION

Summary of Changes

Pursuant to amendments to 10 CFR 50, Appendices G and H, Technical Specification Table 4.4-5 is being changed to reflect the new schedule requirements for removal and analysis of reactor vessel material irradiation surveillance specimens. Additionally, the Bases for Technical Specification 4.4.9.1 are being changed to reflect the reference in 10 CFR 50, Appendix H, to ASTM E 185-82, and to correct several typographical errors.

Installation of personnel air lock leak detection systems at the 45-ft and 93-ft elevations of Containment resulted in the installation of two Containment penetrations. Valves on each of these penetrations are considered Containment isolation valves. When updating the FSAR, the integrated leakage rate test penetration isolation valves were properly categorized as Containment isolation valves.

Containment penetrations must be capable of being made relatively leaktight to minimize the spread of radioactivity following a Design Basis Accident. Designation and leakage rate testing of Containment penetration isolation valves are included in the Technical Specifications to ensure this objective is met. Designating the valves discussed in this LCA as Containment isolation valves requiring Type C leakage rate testing is consistent with the Plant safety design.

The current Technical Specification Basis 3/4.6.1.4 reports peak pressures of 57.8 and 59.8 psig corresponding to initial Containment pressures of 0.0 and +1.6 psig, which were found to be invalid due to the use of an incorrect computer model. This revision incorporates results from new analyses assuming an initial positive Containment pressure of +2.0 psig. The basis for expressing results in psig is clarified.

In Technical Specification Basis 3/4.6.1.6, the sentence, "Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 59.3 psig in the event of a LOCA", will be deleted.

Effect on Technical Specifications

Table 4.4-5 in Technical Specification 4.4.9.1 is being changed to reflect the new capsule withdrawal schedule required by 10 CFR 50, Appendix H. This schedule corresponds with the schedule recommended for the Trojan Nuclear Plant in WCAP-9469, which was prepared in accordance with ASTM E 185-79. The schedule requirements in ASTM E 185-82, which is the referenced version of the standard in 10 CFR 50, Appendix H, are the same as those in ASTM E 185-79; therefore, the proposed schedule is in compliance with 10 CFR 50, Appendix H. This change is administrative in nature, and does not result in an unreviewed safety question.

Containment isolation valves are addressed in Trojan Nuclear Plant Technical Specifications, Appendix A to License NPF-1, Section 3/4.6.3, Page 3/4 6-15. This section of the Technical Specifications details required operating times, if applicable, and testing for all Containment isolation valves. Table 3.6-1 lists all such valves, their function, and required isolation time.

The Technical Specifications should be revised to add isolation valves SV-6991, SV-6992, 10610A, 10610B, 10611A, 10611B, LD-001, and LD-003. A 3-sec closure time for valves SV-6991/6992 is a reasonable maximum closure time before suspecting a problem or malfunction of the valves.

Containment isolation valves are addressed in Technical Specification Section 3/4.6.1.2. This section details operation and surveillance requirements for Containment isolation valves, as well as leakage rate testing of the valves. The change described above does not affect, nor require revision to, Section 3/4.6.1.2.

Containment air locks are addressed in Technical Specification Section 3/4.6.1.3. This section details operation and surveillance requirements of the air locks, as well as leakage rate testing of the air locks. The change described above does not affect, nor require revision to, Section 3/4.6.1.3.

These changes have no effect on the Technical Specifications for Containment internal pressure:

3.6.1.4 Primary Containment internal pressure shall be maintained between -1.1 and +1.6 psig.

3.6.1.6 The structural integrity of the Containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

Specifications 3.6.1.4 and 3.6.1.6 remain valid. The changes reflect the results of recent valid Containment DBA LOCA analyses and the deletion of an inappropriate statement. In both cases, the meaning and intent of the Bases are not altered.

Effect on Bases for Technical Specifications

The changes to the Technical Specification Bases are administrative in nature, and therefore do not decrease the margin of safety as defined therein.

Applicable sections for the Technical Specification Bases 3/4.6.1.2, 3/4.6.1.3, and 3/4.6.3 are not affected by this change.

The Bases for Technical Specifications 3.6.1.4 and 3.6.1.6 are changed as described above.

The affected Bases are:

B 3/4.6.1.4 Internal Pressure

This Basis is modified to include recent valid Containment peak pressure analyses. The maximum peak pressure resulting from a DBA LOCA, with an initial pressure of +2.0 psi above ambient barometric pressure, is 59.9 psig. The Containment design pressure is not exceeded, and the meaning and intent of this Basis remains unchanged.

B 3/4.6.1.6 Containment Structural Integrity

The statement, "Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 59.3 psig in the event of a LOCA", has been deleted. This statement in the Basis was misleading because structural integrity is required to withstand a 60-psig design pressure due to a LOCA, as well as other design loads. The meaning and intent of this Basis has not changed.

Effect on FSAR

Section 5.3.1.6 of the Updated FSAR will be revised in the 1984 Amendment to reference ASTM E 185-82, rather than ASTM E 185-79. This change is administrative in nature, and does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated FSAR, nor does it create the possibility for an accident or malfunction of a different type than any evaluated previously in the Updated FSAR.

Updated FSAR Section 6.2.4 details Containment isolation components which confine to the Containment any radioactivity that may be released following a Design Basis Accident. Section 6.2.4.1 states that Type I fluid lines, where possible, will have one isolation valve located inside Containment and one valve outside. General Design Criterion 56 has similar requirements. The 45-ft and 93-ft elevation inner-door leak test lines have both isolation valves outside Containment. Hence, Section 6.2.4.3, Design Evaluation, should note three vice two exceptions to General Design Criteria 54 through 57. A description of the exception should follow along with reasons for its acceptability. Also, cited in this section are FSAR instrument diagrams, Figures 6.2-1 through 6.2-49, detailing the configuration of Containment penetrations and their isolation valves. The 45-ft and 93-ft elevation air lock inner-door leak test lines are not included in these instrument diagrams, but should be. The isolation valves on these lines should be added to FSAR Table 6.2-1, "Containment Isolation Barriers". The manual valves in the 45-ft elevation test line should be marked "locked closed" in the figure and table.

A review of the Updated FSAR shows that this change does not involve an unreviewed safety question. This is concluded from the following:

1. An increase in the occurrence probability of an accident or malfunction of safety-related equipment previously analyzed in the FSAR is not probable.

2. There is no increase in the consequences of an accident or malfunction of safety-related equipment previously analyzed in the FSAR. This change ensures proper consideration and testing of the valves listed as Containment isolation valves. The reliability of the Containment structure as a barrier to the release of radioactivity is improved, not degraded.

3. There is no possibility of a different type of accident or malfunction of safety-related equipment from that previously analyzed in Chapter 15 of the Updated FSAR.

4. There is no reduction in the margin of safety, as defined in the Bases of Appendix A Technical Specification.

5. Additional valves will be leakage rate tested; however, no change to the test methodology or procedures contained in the FSAR is required.

Section 6.2.1.3.4 of the Updated FSAR describes the estimated peak Containment pressures resulting from a DBA LOCA with several assumed initial Containment pressures. Included in this section is a description of the Containment analysis containing the computer error.

The change to Technical Specification Basis 3/4.6.1.4 will not directly affect the Updated FSAR. However, a change to the FSAR has been initiated to reflect the peak pressures obtained in the recent analyses.

An unreviewed safety question does not exist for the following reasons:

1. An increase in the occurrence probability of an accident or malfunction of safety-related equipment previously analyzed in the FSAR will not occur, since the change to Basis 3/4.6.1.4 affects the results of a postulated accident and not any factors contributing to the initiation of an accident. The change to Basis 3/4.6.1.6 deletes a misleading statement, and the original intent and meaning remains with no effect on the probability of an accident or malfunction of safety-related equipment.

2. An increase in the consequences of an accident or malfunction of safety-related equipment previously analyzed in the FSAR will not result. The latest analytical results indicate the Containment design pressure is not exceeded during a postulated DBA LOCA. Therefore, the consequences of a DBA LOCA have not changed. Deletion of the statement in Basis 3/4.6.1.6 is not applicable to this concern.

3. The possibility of a different type of accident or malfunction of safety-related equipment previously analyzed in the FSAR is not anticipated for the same reasons as in Item I above.

4. A reduction in the margin of safety, as defined in the Bases of the Appendix A Technical Specifications, will not result. The latest analyses predict lower maximum peak Containment pressures. The margin below the design Containment pressure is correspondingly larger.

   Peak Containment pressure (in gauge, psig) resulting from a DBA LOCA does not remain constant with changing barometric pressure. The difference between Containment and barometric pressure varies depending on the value of the latter. This sensitivity was examined for the range of barometric pressures experienced in Portland over the past several years.

   The results of this examination indicate a variation of less than 0.1 psi in peak Containment pressure (above ambient barometric pressure), about the nominal case of 14.7 psia barometric pressure. Therefore, even with an increase of 0.1 psi, the analytical results will not exceed the Containment design pressure.

   The statement deletion in Basis 3/4.6.1.6 does not impact the safety margin.

5. Tests and procedures contained in the FSAR are not affected.

Environmental Effect

Adverse environmental effects are not anticipated. The change to the Bases reaffirms the previous conclusions that the Containment design pressure will not be exceeded.

Effect on Other Licensing Documents, Commitments, or Criteria

1. Valves SV-6991, SV-6992, LD-001, and LD-003 will be added to Table 4.1-1 of PGE-1022, "Inservice Testing Program for Pumps and Valves".

2. ANS 56.8-1981:

   ANS 56.8-1981 directs that Containment isolation valves that are closed or that close automatically upon receipt of an isolation signal in response to controls intended to affect Containment isolation should be subjected to Type C valve leakage rate tests. The valves discussed in this LCA fall into this category. This license change ensures compliance with ANS 56.8-1981.

3. 10 CFR 50, Appendix J:

   10 CFR 50, Appendix J, directs that Containment isolation valves that provide a direct connection between the inside and outside atmospheres of the primary reactor Containment under normal operation be subjected to Type C valve leakage rate tests. The valves discussed in this LCA fall into this category. This license change ensures compliance with Appendix J of 10 CFR 50.

4. 10 CFR 50, Appendix A, General Design Criteria 54 and 56:

   Criterion 54, "Systems Penetrating Containment", states that piping systems penetrating Containment shall be provided with leak detection, isolation, and Containment capabilities. Additionally, determining if valve leakage is within acceptable limits is required.

   Criterion 56, "Primary Containment Isolation", requires that each line that connects directly to Containment atmosphere and penetrates the primary reactor Containment be provided with isolation valves. In addition, it requires one locked-closed isolation valve inside and one locked-closed isolation valve outside Containment for manual isolation valves, unless it can be demonstrated that the Containment isolation provisions are acceptable on some other defined basis. The 45-ft elevation leak test line isolation valves are both outside Containment. Due to their close proximity to the Containment barrier and that FSAR Section 6.2.4.1 allows exceptions to the criteria mentioned above, the present condition appears acceptable. The valves meet all other requirements placed on Containment isolation valves with regard to Seismic Category, class, etc.

5. Standard Review Plan (SRP):

   Sections 6.2.4, "Containment Isolation Systems", and 6.2.6, "Containment Leakage Testing", of the SRP discuss identification and testing of Containment isolation valves consistent with the requirements of 1, 2, and 3 above.

6. Regulatory Guide 1.141, "Containment Isolation Provisions for Fluid Systems", April 1978:

   Leak rate testing of the valves listed in this LCA is consistent with the requirements of Section C.1 of Regulatory Guide 1.141. The Company's In-House Position is one of partial compliance, allowing for a few minor exceptions. The fact that the 45-ft elevation leak test lines are both outside Containment would be an additional exception. This subject was previously addressed in (3) above. The Regulatory Guide endorses ANSI N271/ANS 56.2-1976, which requires identification and leakage rate testing of Containment isolation valves. Its criteria for locating Containment isolation valves are identical to General Design Criterion 56.

SIGNIFICANT HAZARDS DETERMINATION

It has been determined that this License Change Application involves no significant hazards considerations. Evaluation of the proposed amendment verifies that it does not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident for any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

This LCA was initiated in order to maintain compliance with NRC regulations. All changes are administrative in nature. The addition of the Containment isolation valves, the revised DBA Containment analyses and the changes made to the reactor vessel material surveillance schedule do not create any new problems or accidents or involve a significant increase in the probability of any previously evaluated accident or involve any reduction in a margin of safety. The schedule for irradiated specimen withdrawal has been moved up such that any potentially damaging radiation effects to the reactor vessel will be discovered at an earlier date than was previously scheduled, thus increasing the margin of safety. It is concluded that this proposed amendment does not involve any significant hazards considerations.

SCHEDULE CONSIDERATIONS

With the schedule change proposed in the attached Table 4.4-5, it will be necessary to remove Capsule X during the 1984 refueling outage. Therefore, it is requested that this change be processed prior to April 1, 1984.

BASIS FOR DETERMINATION OF AMENDMENT CLASS

This License Change Application has been determined to result in a Class II amendment in accordance with the criteria of 10 CFR 170.22. This proposed change is administrative in nature and does not involve a matter of safety or environmental significance. The fee for a Class II amendment is $1,200.

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