PSEG Nuclear LLC
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PSEG
Nuclear LLC

DEC 23 2002

LR-N02-0406
LCR H02-015

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

Gentlemen:

REQUEST FOR CHANGE TO REACTOR MATERIAL SURVEILLANCE PROGRAM
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

PSEG Nuclear LLC (PSEG) hereby requests a change to the Hope Creek Generating Station (HCGS) reactor vessel material surveillance program required by 10CFR50, Appendix H. This change will incorporate the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) into the HCGS licensing basis. Consistent with the process established between the NRC and the BWRVIP, this change is being processed as a license amendment to facilitate NRC review and approval.

The License Change Request proposes to change the Hope Creek Updated Final Safety Analysis Report (HCUFSAR), Section 5.3, "Reactor Vessel", and Appendix 5A, "Compliance with 10CFR50, Appendix G and H."

PSEG has evaluated the proposed changes in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c) and has determined that this request involves no significant hazards considerations. An evaluation of the requested changes is provided in Attachment 1 to this letter. The marked up HCUFSAR pages affected by the proposed changes are provided in Attachment 2. In accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

PSEG plans to implement the proposed changes in the Fall of 2004 to support deletion of work from Refueling Outage 13 (RFO13). Approval of this proposed change is being requested by April 2004 with the changes effective 30 days after approval.

A008

If you have any questions or require additional information, please contact Mr. Michael Mosier at (856) 339-5434.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 12/23/2002

Sincerely,

D.F. Garchow

D. F. Garchow
Vice President – Projects and Licensing

Attachments (2)

C: Mr. H. Miller, Administrator – Region I
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Mr. George Wunder, Project Manager – Hope Creek
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USNRC Senior Resident Inspector – Hope Creek (X24)

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HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

EVALUATION OF CHANGES TO THE HOPE CREEK UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR)

EVALUATION OF CHANGES TO THE HOPE CREEK UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR)

REQUEST FOR CHANGE TO THE HOPE CREEK UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR)

1. DESCRIPTION

PSEG Nuclear, LLC (PSEG) proposes to revise the licensing basis for Hope Creek Generating Station (HCGS) by replacing the current plant-specific reactor pressure vessel (RPV) material surveillance program with the Boiling Water Reactor (BWR) Integrated Surveillance Program (ISP), which was approved by the NRC in its Safety Evaluation (SE) dated February 1, 2002 (Reference 1).

2. PROPOSED CHANGES

Hope Creek Updated Final Safety Analysis Report (HCUFSAR), Section 5.3.1.6.1, "Compliance with Reactor Vessel Material Surveillance Program Requirements", and Appendix 5A, Section 5A.4, "Reactor Pressure Vessel Surveillance Specimens" will be revised to include the BWR ISP and as such, the surveillance capsule withdrawal schedule will be in accordance with the ISP. Section 5.3.4, "References" will include references to BWR ISP program submittals. The proposed revision to the Hope Creek Updated Final Safety Analysis Report (HCUFSAR) reflecting this change is provided in Attachment 2.

3. BACKGROUND

The BWR ISP was developed in response to an issue raised by the NRC staff regarding the potential lack of adequate unirradiated baseline Charpy V-notch (CVN) data for one or more materials in plant-specific RPV surveillance programs at several BWRs. The lack of baseline properties would inhibit a licensee's ability to effectively monitor changes in the fracture toughness properties of RPV materials in accordance with Appendix H to 10 CFR 50. The BWR ISP, as approved by the NRC, resolves this issue. Implementation of the ISP will provide additional benefits. When the original surveillance materials were selected for plant-specific surveillance programs, the state of knowledge concerning RPV material response to irradiation and post-irradiation fracture toughness was not the same as it is today. As a result, many facilities did not include what would be identified today as the plant's limiting RPV materials in their surveillance programs. Hence, this effort to identify and evaluate materials from other BWRs, which may better represent a facility's limiting materials, should improve the overall evaluation of BWR RPV embrittlement. Second, the inclusion of data from the testing of BWR Owners' Group (BWROG) Supplemental Surveillance Program (SSP) capsules will improve overall quality of the data being used to evaluate BWR RPV embrittlement. Finally, implementation of the ISP is also expected to reduce the cost of surveillance testing and analysis since surveillance materials that are of little or no value (either because they lack adequate unirradiated baseline CVN data or because they are not the best representative materials) will no longer be tested.

4. TECHNICAL ANALYSIS

Reference 1 concludes that the proposed ISP, if implemented in accordance with the conditions in the SE, has been determined to be an acceptable alternative to all existing BWR plant-specific RPV surveillance programs for the purpose of maintaining compliance with the requirements of Appendix H to 10 CFR Part 50 through the end of current facility 40 year operating licenses.

Reference 1 requires that each licensee (1) provide information regarding what specific neutron fluence methodology will be implemented as part of participation in the ISP and (2) address the neutron fluence methodology compatibility issue as it applies to the comparison of neutron fluences calculated for its RPV versus the neutron fluences calculated for surveillance capsules in the ISP which are designated to represent its RPV. In a letter dated March 29, 2002 PSEG (Reference 2) committed to updated fluence calculations in accordance with Regulatory Guide 1.190 by May 1, 2005. All future fluence calculations for purposes of establishing values for RPV and ISP surveillance capsules will be compatible with NRC approved fluence methodologies.

5. REGULATORY SAFETY ANALYSIS

5.1. No Significant Hazards Consideration

PSEG Nuclear LLC (PSEG) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10CFR50.92, "Issuance of amendment," as discussed below:

5.1.1. Does the change involve a significant increase in the probability or consequences of an accident previously analyzed?

Response: No.

The proposed change implements an integrated surveillance program that has been evaluated by the NRC staff as meeting the requirements of paragraph III.C of Appendix H to 10 CFR 50. Consequently, the proposed change does not significantly increase the probability of any accident previously evaluated. The proposed change provides the same assurance of RPV integrity. As a result, the consequences of any accident previously evaluated are not significantly increased.

Therefore, this proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously analyzed.

5.1.2. Does the change create the possibility of a new or different kind of accident from any accident previously analyzed?

Response: No

The proposed change revises the HCGS licensing basis to reflect participation in the ISP. The proposed change does not involve a modification of the design of plant structures, systems or components (SSC). Also, the proposed change will not degrade the reliability of SSCs important to safety since protective features will not be deleted or modified. The proposed change will not impact the manner in which the plant is normally operated. The proposed change maintains an equivalent level of RPV material surveillance and does not introduce any new accident initiators. Therefore, this proposed amendment does not create the possibility of a new or different kind of accident from any previously analyzed.

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

Response: No

The proposed change has been evaluated as providing an acceptable alternative to the plant-specific RPV material surveillance program that meets the requirements of the regulations for RPV material surveillance. Therefore, these changes do not involve a significant reduction in margin of safety.

Based on the above, PSEG concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10CFR50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2. Applicable Regulatory Requirements/Criteria

PSEG proposes to revise the licensing basis for HCGS by replacing the plant-specific RPV material surveillance program with the BWR ISP. This change is acceptable because the BWR ISP has been approved by the NRC staff as meeting the requirements of paragraph III.C of Appendix H to 10 CFR 50 for an integrated surveillance program. In accordance with the criteria set forth in 10 CFR 50.92, PSEG has evaluated the proposed UFSAR changes and determined it does not represent a significant hazards consideration.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6. ENVIRONMENTAL CONSIDERATION

PSEG has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, 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 need be prepared in connection with the proposed amendment.

7. REFERENCES

7.1. EPRI 1003346, BWRVIP-86-A, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan, October 2002.

7.2. Letter LR-N02-0039, Request for Change to Technical Specifications Extended Use Pressure-Temperature Curves, dated March 29, 2002.

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REVISIONS TO THE UPDATED FINAL SAFETY ANALYSIS REPORT

HOPE CREEK UPDATED FINAL SAFETY ANALYSIS PAGES WITH PROPOSED CHANGES

The following Updated Final Safety Analysis Report pages are affected by this change request:

Insert #1 for page 5.3-11:

The program for implementation of the scheduling and testing of the surveillance specimens is governed and controlled by BWR Vessel Internals Project (BWRVIP) Integrated Surveillance Program (ISP). The ISP is defined in BWRVIP-86-A, Updated BWR Vessel an Integrated Surveillance Program (ISP) Implementation Plan (reference 5.3-12). The NRC has issued a safety evaluation for the BWRVIP ISP and is included in Appendix B of BWRVIP-86-A.

The withdrawal schedule will be in accordance with the BWRVIP ISP and is:

1. The first set at the 30° azimuth was withdrawn during the 5th refueling outage at 6.01 EFPY.

2. The second set at the 120° azimuth will be withdrawn when the accumulated neutron fluence of the capsule corresponds to the projected EOL ¼ T reactor vessel fluence. This is projected to be withdrawn in 2014 at approximately 22 EFPY.

3. The third set is considered a standby capsule. The ISP considers this set a license renewal candidate.

Insert #2 for page 5.3-34:

5.3-12 EPRI 1003346, BWRVIP-86-A, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan, October 2002.

Insert #3 for page 5A-9:

as specified in section 5.3.1.6.1

Insert #4 for page 5A-10:

criteria found in the BWR Vessel Internals Project (BWRVIP) Integrated Surveillance Program (ISP), reference 5.3-12.

Replace with Insert 1

The withdrawal schedule of the three sets of specimens in the reactor is planned as follows:

1. The first set at the 30° azimuth was withdrawn during the 5th refueling outage.

2. The second set will be withdrawn when its exposure corresponds to fifteen effective full power years or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel inner wall location, whichever comes first. This withdrawal will be scheduled for the nearest vessel refueling date based on the above criteria.

3. The third set will be held to the EOL (not less than once, nor more than twice the peak EOL vessel fluence at the vessel inside surface). This capsule may be held without testing following withdrawal.

A discussion of the extent of compliance to 10CFR50, Appendix H is provided in Appendix 5A.

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Sections 4.1.4.5 and 4.3.2.8.

5.3.1.6.3 Predicted Irradiation Effects on Vessel Beltline Materials

Estimated maximum changes to initial RT_{NDT} and upper shelf requirements. Welds on the RPV were examined in accordance with methods prescribed and meet the acceptance requirements specified by ASME B&PV Code, Section III.

5.3.4 References

5.3-1 General Electric Company, "An Analytical Study on Brittle Fracture of GE-BWR Vessel Subject to the Design Basis Accident," NEDO 10029, July 1969.

5.3-2 Watanabe, H., "Boiling Water Reactor Feedwater Nozzle/Sparger Final Report," NEDE 21821-2 (Proprietary Version) and NEDO 21821-2 (Non-Proprietary Version), General Electric Company, August 1979.

5.3-3 Cooke, F., "Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors," NEDO 21778-A, General Electric Company, December 1978.

5.3-4 General Electric Company, "BWR Radiation Effects Design Curve," NEDO 20651, Figure 4-1, March 1975.

5.3-5 General Electric Company, "RPV Surveillance Materials Testing and Fracture Toughness Analysis," GE-NE-A164-1294, R1, DRF 137-0010-7, December 1997.

5.3-6 General Electric Company, "10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 through BWR/6 Vessels," NEDO-32205-A, Rev. 1, February 1994.

5.3-7 General Electric Company, "Basis for GE RTNDT Estimation Method," NEDC-32399-P, September 1994.

5.3-8 Structural Integrity Associates, Inc., ''Revised Pressure-Temperature Curves for hope Creek,'' SIR-00-136, Rev. 0, November 3, 2000.

5.3-9 Welding Research Council, PVRC Ad Hoc Group on Toughness Requirements, ''PVRC Recommendations on Toughness Requirements for Ferritic Materials,'' WRC Bulletin 175, August 1972.

5.3-10 ASME Boiler and Pressure Vessel Code, Code Case N-588, ''Alternative Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels,'' Section XI, Division 1, Approved December 12, 1997.

5.3-11 ASME Boiler and Pressure Vessel Code, Code Case N-640, ''Alternative Reference Fracture Toughness for Development of P-T Limit Curves,'' Section XI, Division 1, Approved February 26, 1999.

Add insert 2

shows weld metal surveillance specimens to have been fabricated away from the root of the weld. Therefore, it is assumed that weld metal surveillance specimens represent only heat/lot D53040/1125-02205 material.

The number of surveillance specimen capsules and the number of specimens are in compliance with ASTM E185-73. The capsule holders inside the vessel are located at 30°, 120°, and 300° azimuths. The capsule located at the 30° azimuth was removed during the fifth refueling outage. Capsule contents, including number and orientation of specimens, are given in Table 5A-18.

Insert 3

The withdrawal schedule for the surveillance program capsules meet the requirements of ASTM Standard E 185-82. The lead factors for the HCGS surveillance capsules are 1.01 at the inside surface of the vessel and 1.46 at one-quarter of the way through the vessel wall measured from the inside surface. These lead factors were calculated assuming that the vessel is symmetrical. This assumption was made because the vessel qualification program did not provide for measurements of vessel radii to identify any angular locations where the inside diameter of the vessel is larger than nominal. Hence, it is possible that a surveillance capsule could be located at an extended radius position. This would provide surveillance sample test results lower than calculated and nonconservative values for the peak fluence when it is estimated from the capsule data using the aforementioned lead factors.

The orientations of the surveillance specimens are acceptable since the data indicate that radiation embrittlement is independent of specimen orientation. Longitudinally oriented CVN specimens from the heat affected zone (HAZ) simulate the conditions of longitudinal production weld joints.

The End-of-Life (EOL) calculated peak fluence at the inside diameter of the vessel is 7.56×10^{17} n/cm^2 (E > 1.0 Mev) and at one quarter of the vessel thickness is 5.24×10^{17} n/cm^2 (E > 1.0 Mev). The

Replace with Insert 4

withdrawal of the capsules will be according to the ~~following criteria:~~

~~1. The first set was withdrawn during the fifth refueling outage.~~

~~2. The second set will be withdrawn when its exposure corresponds to fifteen effective full power years or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel inner wall location, whichever comes first. This withdrawal will be scheduled for the nearest vessel refueling date based on the above criteria.~~

~~3. The third set will be held to the EOL (not less than once, nor more than twice the peak EOL vessel fluence at the vessel inside surface). This capsule may be held without testing following withdrawal.~~

The construction tolerances on the reactor vessel required that the minimum (nominal) radius of the vessel be maintained. The applicable version of the ASME B&PV Code did allow for areas of the vessel to have larger radii. The measurement acceptance techniques for the vessel were either the use of a template to test the minimum diameter or a series of measurements to determine the diameter at various points. The measurement technique did not require the identification of the locations where the vessel diameter is longer than nominal. Hence the lead factors were calculated for the nominal dimension.