Portland General Electric Company

Bart D. Withers Vice President

October 31, 1986

Trojan Nuclear Plant
Docket 50-344
License NPF-1

Director of Nuclear Reactor Regulation
ATTN: Mr. Steven A. Varga
Director, PWR-A
Project Directorate No. 3
U.S. Nuclear Regulatory Commission
Washington DC 20555

Dear Sir:

License Change Application 147

Attached are three signed originals and 40 conformed copies of License Change Application (LCA) 147 requesting amendment of Operating License NPF-1. This LCA addresses the reactor vessel material irradiation surveillance schedule and new 10 CFR 50, Appendix G, pressure-temperature limits. An LCA fee of $150 is attached in accordance with 10 CFR 170.

Also attached is one signed copy of a Certificate of Service for LCA 147 to the chief executive of the county in which the facility is located and the Director of the State of Oregon, Department of Energy.

Sincerely,

CP Yundt for

Bart D. Withers
Vice President
Nuclear

c: Mr. Lynn Frank, Director
State of Oregon
Department of Energy

Mr. Michael J. Sykes
Chairman of County Commissioners

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PDR ADOCK 05000344
P PDR

121 S.W. Salmon Street, Portland, Oregon 97204

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)
)
PORTLAND GENERAL ELECTRIC COMPANY,) Docket 50-344
THE CITY OF EUGENE, OREGON, AND) Operating License NPF-1
PACIFIC POWER & LIGHT COMPANY)
)
(TROJAN NUCLEAR PLANT))

CERTIFICATE OF SERVICE

I hereby certify that copies of License Change Application 147 to the Operating License for Trojan Nuclear Plant, dated October 31, 1986, have been served on the following by hand delivery or by deposit in the United States mail, first class, this 31st day of October 1986:

Mr. Lynn Frank, Director
State of Oregon
Department of Energy
625 Marion St NE
Salem OR 97310

Mr. Michael J. Sykes
Chairman of County Commissioners
Columbia County Courthouse
St. Helena OR 97051

Bill Kershul
B. L. Kershul, Acting Manager
Nuclear Regulation Branch
Nuclear Safety & Regulation

Subscribed and sworn to before me this 31st day of October 1986.

Carole A. Hodgdon
Notary Public of Oregon

My Commission Expires: August 9, 1987

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

Operating License NPF-1
Docket 50-344
License Change Application 147

This License Change Application requests modifications of the Technical Specification contained in Appendix A to Operating License NPF-1. In order to maintain compliance with NRC regulations, changes are proposed to revise the reactor vessel material irradiation surveillance schedule and address new 10 CFR 50, Appendix G, pressure-temperature limits.

PORTLAND GENERAL ELECTRIC COMPANY

By C P Yundt for
Bart D. Withers
Vice President
Nuclear

Subscribed and sworn to before me this 31st day of October 1986.

Carole A. Hodgdon
Notary Public of Oregon

My Commission Expires: August 9, 1987

LICENSE CHANGE APPLICATION 147

The proposed replacement pages to Appendix A of Facility Operating License NPF-1 are provided as Attachment 1. A description of the changes to the existing Technical Specifications are as follow:

Pages 3/4 4-25 and 3/4 4-26. Changes to Section IV.A.2 of 10 CFR 50, Appendix G, require that when the core is not critical, and pressure exceeds 20 percent of the preservice system hydrostatic test pressure (620 psi), that the temperature of the closure flange regions must exceed the reference temperature of the material in those regions by at least 120°F for normal operation. The heatup and cooldown rate pressure-temperature limit curves of Figures 3.4-2 and 3.4-3 have been revised to reflect this requirement as well as the results of the analysis of radiation specimen Capsule X. For the purpose of the change, the reference temperature for the head flange was determined to be 20°F from Technical Specification Table B 3/4 4-1. Possible instrument errors of 10°F and 60 psig have been included in the curves.

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 is required by the current Technical Specification. WCAP-10861, "Analysis of Capsule X From Portland General Electric Company Trojan Reactor Vessel Radiation Surveillance Program", provided a recommended removal schedule for the remaining Capsules, which is in compliance with ASTM E 185-82.

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 10 EFPY rather than 12 EFPY as stated in Paragraphs 3 and 4 on Page B 3/4 4-5. A paragraph was added to discuss the bases for the "knee" in Figures 3.4-2 and 3.4-3.

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. In the final paragraph on this page the applicability of the heatup and cooldown curves is being changed from the first 15 EFPY to the first 10 EFPY to reflect the Capsule X analysis.

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 and the analysis of radiation specimen Capsule X, Technical Specification Table 4.4-5 should be revised to reflect the schedule changes for future specimen Capsule removal. 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. Finally, Figures 3.4-2 and 3.4-3 and the Technical Specification Bases for these figures are being revised to reflect new 10 CFR 50, Appendix G, pressure-temperature limits and the results from the analysis of radiation specimen Capsule X.

SIGNIFICANT HAZARDS DETERMINATION

In accordance with the requirements of 10 CFR 50.92, this application is judged to involve no significant hazards based upon the following information:

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

   This change would have no effect upon the probability or consequences of an accident previously evaluated. Operating limits are being adjusted to incorporate empirical data obtained through analysis of irradiation specimen Capsule X. The Capsule X data demonstrates the reactor vessel is less sensitive to radiation affects than predicted by Regulatory Guide 1.99, Revision 1. Changing these operating limits does not affect the probability or consequences of previously evaluated accidents. The Plant will operate within these new limits ensuring that the probability and consequences of previously evaluated accidents are unchanged.

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

   The possibility of a new kind of accident is not created since the operating limits are merely being updated according to changes in federal regulations (10 CFR 50, Appendices G and H) and the Capsule X analysis report. No physical changes are being made to the Plant or operating requirements. The revised heatup and cooldown curves and schedule for radiation specimen removal reflect empirical data obtained through analysis of radiation surveillance specimen Capsule X.

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

   The equipment of concern is the reactor pressure vessel. The margin of safety for the reactor pressure vessel will not be affected. The predicted affects of irradiation on the vessel have been determined to be more conservative than the actual effects as determined by the analysis of surveillance Capsule X.

The change in the operating limits is based upon actual data and ensures that original assumptions, adequate conservatisms, and the margin of safety of the reactor pressure vessel remain as initially designed.

In the April 6, 1983 Federal Register, the NRC published a list of examples of amendments that are not likely to involve significant hazards considerations. Example Number 7 of that list applies to these proposed changes and states that a change similar to the following would not likely involve a significant hazards consideration:

"A change to make a license conform to changes in the regulations where the license change results in very minor changes to facility operations clearly in keeping with the regulations."