approximately 21 miles southeast of Hartford, Connecticut, on the east bank of the Connecticut River. Notice of these oral limited appearance sessions will be published in the **Federal Register** and/or made available to the public at the NRC Public Document Room.

Documents related to this proceeding are available electronically through the Agencywide Documents access and Management System (ADAMS), with access to the public through the NRC's Internet Web site (Public Electronic Reading Room Link, *http://www.nrc.gov/NRC/ADAMS/index.html).* The NRC Public Documents Room (PDR) and many public libraries have terminals for public access to the Internet. Documents that may relate to this proceeding that are dated earlier than December 1, 1999, are available in microfiche form (with print form available on one-day recall) for public inspection at the PDR, Room 0–1 F21, NRC One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852–2738.

Dated: July 12, 2001.

For the Atomic Safety and Licensing Board.

Ann Marshall Young,

Chair, Administrative Judge.

[FR Doc. 01–17952 Filed 7–17–01; 8:45 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–354]

PSEG Nuclear, LLC, Hope Creek Generating Station; Exemption

1.0 Background

The PSEG Nuclear LLC (PSEG or the licensee) is the holder of Facility Operating License No. NPF–57 which authorizes operation of the Hope Creek Generating Station (HCGS). The license provides, among other things, that the facility is subject to all rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (NRC, the Commission) now or hereafter in effect.

The facility consists of a boiling water reactor located in Salem County in New Jersey.

2.0 Request/Action

Title 10 of the Code of Federal Regulations, part 50, appendix G, requires that pressure-temperature (P–T) limits be established for reactor pressure vessels (RPVs) during normal operating and hydrostatic or leak rate testing conditions. Specifically, 10 CFR part 50, appendix G, states that ''[t]he appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions.'' In addition, Appendix G of 10 CFR part 50 specifies that the requirements for these limits ''must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code].''

By letter dated December 1, 2000, as supplemented by letters dated February 12, May 7, and May 14, 2001, PSEG submitted a license amendment request to increase the HCGS core thermal power level by 1.4 percent. The amendment request included proposed P–T limit curves for the HCGS RPV. As part of the same submittal, PSEG requested an exemption from specific requirements of 10 CFR 50.60(a) and Appendix G. The proposed exemption would allow the use of ASME Code Cases N–588, ''Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1,'' and N–640, ''Alternative Reference Fracture Toughness for Development of P–T Limit Curves for ASME Section XI, Division 1,'' as alternative methods for complying with the fracture toughness requirements in 10 CFR Part 50, Appendix G. The proposed amendment relies, in part, on the requested exemption since the proposed P–T limit curves for the HCGS RPV were developed based on the use of Code Cases N–588 and N–640. Pursuant to 10 CFR 50.60(b), proposed alternatives to the requirements in Appendices G and H of 10 CFR Part 50 may be used by licensees when the Commission grants an exemption under 10 CFR 50.12.

3.0 Discussion

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR part 50, when (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present. The licensee's application states that the proposed exemption meets the special circumstances provisions in 10 CFR 50.12(a)(2)(ii), which states that ''[a]pplication of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.''

As previously discussed, the licensee has requested an exemption to use ASME Code Cases N–588 and N–640 as alternative methods for complying with the fracture toughness requirements in 10 CFR Part 50, Appendix G. The underlying purpose of 10 CFR part 50, appendix G, is to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. This is accomplished through these regulations that, in part, specify fracture toughness requirements for ferritic materials of the reactor coolant pressure boundary. The staff's review related to each of the Code Cases is discussed below.

Code Case N–588

Code Case N–588 amends the provisions of the 1989 Edition of ASME Section XI, Appendix G, by permitting the postulation of a circumferentially oriented reference flaw as the limiting flaw in an RPV circumferential weld for the purpose of establishing RPV P–T limits. The 1989 Edition of ASME Section XI, Appendix G, would require that such a reference flaw be postulated as an axially oriented flaw in the circumferential weld.

The licensee addressed the technical justification for this exemption by citing industry experience and aspects of RPV fabrication which support the postulation of circumferentially oriented flaws for these welds. The reference flaw is a postulated flaw that accounts for the possibility of a prior existing defect that may have gone undetected during the fabrication process. Postulating the ASME Section XI, Appendix G reference flaw in a circumferential weld is physically unrealistic and overly conservative, because the length of the flaw is 1.5 times the vessel wall thickness, which is much longer than the width of the circumferential weld. Industry experience with the repair of weld indications found during preservice inspection, inservice nondestructive examinations, and data taken from destructive examination of actual vessel welds confirms that any remaining defects are small, laminar in nature, and do not cross transverse to the weld bead. Therefore, any postulated defects introduced during the fabrication process, and not detected during subsequent nondestructive examinations, would only be expected to be oriented in the direction of weld fabrication. ASME Code Case N–588 also provides appropriate procedures for determining the stress intensity factors for use in developing RPV P–T limits in accordance with ASME Code, Section XI, Appendix G, procedures. The procedures allowed by ASME Code Case

N–588 are conservative and provide a margin of safety in the development of RPV P–T operating and pressure test limits that will prevent nonductile fracture of the vessel.

The staff concurs with the licensee's conclusion that the postulation of an axially oriented flaw on a circumferential RPV weld is a level of conservatism that is not required to establish P–T limits to protect the reactor coolant system pressure boundary from failure during hydrostatic testing, heatup, and cooldown. Based on the manufacturing processes used to fabricate RPVs for U.S. facilities, it is reasonable to conclude that, if a significant defect were to exist in a circumferential weld, it would lie in the plane of the welding direction. The use of stress magnification factors which account for this difference in flaw orientation (i.e., account for a factor of approximately two in the difference in the applied pressure stress between the axial and circumferential directions) is acceptable.

The staff also notes that, Code Case N–588, Section 2214.3, includes changes to the methodology for determining the thermal stress intensity, KIT, which was incorporated into Section XI of the ASME Code after the 1989 Edition. The staff has reviewed the basis for these changes in the KIT methodology in detail. The staff accepts that the modifications made to the KIT methodology in Section 2214.3 of Code Case N–588 result in a determination of KIT that is consistent with the methodology found in the 1989 Edition of ASME Code Section XI, Appendix G, and that the use of equivalent KIT values for axial and circumferential flaws is acceptable.

Application of ASME Code Case N–588 when determining P–T operating limit curves per ASME Code, Section XI, appendix G, provides appropriate procedures for determining limiting maximum postulated defects and considering those defects in developing the P–T limits. This application of the code case maintains that margin of safety originally contemplated when ASME Code Section XI, appendix G was developed.

Based on the above considerations, the staff concludes that use of Code Case N–588 for development of the HCGS RPV P–T limit curves will meet the underlying purpose of Appendix G of 10 CFR part 50 with respect to protecting the integrity of the reactor coolant pressure boundary. In this case, since strict compliance with the requirements of 10 CFR 50.60(a) and 10 CFR part 50, appendix G, is not necessary to serve the overall intent of the regulations, the staff also concludes that application of Code Case N–588 for the HCGS meets the special circumstances provisions in 10 CFR 50.12(a)(2)(ii), for granting exemptions to the regulations.

Code Case N–640

Code Case N–640 amends the provisions of ASME Section XI, Appendix G, by permitting the use of the K_{Ic} equation as found in Appendix A in ASME Section XI, in lieu of the K_{Ia} equation as found in Appendix G in ASME Section XI. Use of the K_{Ic} equation in determining the lower bound fracture toughness in the development of the P–T operating limits curve is more technically correct than the use of the K_{Ia} equation since the rate of loading during a heatup or cooldown is slow and is more representative of a static condition than a dynamic condition. The staff has required use of the initial conservatism of the K_{Ia} equation since 1974 when the equation was codified. This initial conservatism was necessary due to the limited knowledge of RPV materials. Since 1974, additional knowledge has been gained about RPV materials, which demonstrates that the lower bound on fracture toughness provided by the K_{Ia} equation is well beyond the margin of safety required to protect the public health and safety from potential RPV failure. In addition, P–T curves based on the K_{Ic} equation will enhance overall plant safety by opening the P–T operating window with the greatest safety benefit in the region of low temperature operations.

Based on the above considerations, the staff concludes that use of Code Case N–640 for development of the HCGS RPV P–T limit curves will meet the underlying purpose of appendix G of 10 CFR part 50 with respect to protecting the integrity of the reactor coolant pressure boundary. In this case, since strict compliance with the requirements of 10 CFR 50.60(a) and 10 CFR part 50, appendix G, is not necessary to serve the overall intent of the regulations, the staff also concludes that application of Code Case N–640 for the HCGS meets the special circumstances provisions in 10 CFR 50.12(a)(2)(ii), for granting exemptions to the regulations.

4.0 Conclusion

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12(a), the exemption is authorized by law, will not endanger life or property or common defense and security, and is, otherwise, in the public interest. Also, special circumstances are present. Therefore, the Commission hereby grants PSEG Nuclear LLC an exemption from the requirements of 10 CFR 50.60(a) and 10 CFR part 50, appendix G, for HCGS.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not have a significant effect on the quality of the human environment (66 FR 33717).

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 12th day of July 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 01–17954 Filed 7–17–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Public Meeting on Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility

AGENCY: Nuclear Regulatory Commission (NRC).

ACTION: Notice of Meeting.

SUMMARY: NRC will host a public meeting in Rockville, Maryland. The meeting will provide an opportunity for discussion on the revised draft Chapter 3 entitled, ''Integrated Safety Analysis'' of NUREG–1520, Standard Review Plan (SRP) for the Review of a License Application for a Fuel Cycle Facility. The March 30, 2001, draft Chapter 3 can be found in both a ''clean'' and marked-up version in the NRC Public Electronic Reading Room under ''Recently Released Documents, April 3, 2001''. It can also be found on the Internet at the following website: *http://techconf.llnl.gov/cgi-bin/library?source=*&library=Part_70_lib.*

The web site can also be reached by the following method:

1. Go the main NRC web site at: *http://www.nrc.gov.*
2. Scroll down to the bottom of that page and click on the word ''Rulemaking.''
3. Scroll down on the Rulemaking page until the words ''Technical Conference'' appear. Click on those words.
4. On the page titled ''Welcome to the NRC Technical Conference Forum,'' click on the link ''Conference'' or ''Technical Conferences''.
5. Scroll down to the topic ''Draft Standard Review Plan and Guidance on Amendment to 10 CFR part 70.''
6. Select ''Document Library''.