License No. NPF–57, issued to the licensee for operation of the Hope Creek Generating Station located in Lower Alloways Creek Township, Salem County, New Jersey. Notice of Consideration of Issuance of this amendment was not published in the Federal Register.

The purpose of the licensee's amendment request was to revise the Hope Creek Generating Station (HCGS) Updated Final Safety Analysis Report, Section 9.2.5, regarding the Station Service Water System and Ultimate Heat Sink.

The NRC staff has concluded that the licensee's request cannot be granted. The licensee was notified of the Commission's denial of the proposed change by a letter dated December 24, 1996.

By January 30, 1997, the licensee may demand a hearing with respect to the denial described above. Any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date.

A copy of any petitions should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to M. J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502, attorney for the licensee.

For further details with respect to this action, see (1) the application for amendment dated August 30, 1996, and (2) the Commission's letter to the licensee dated December 24, 1996.

These documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070.

Dated at Rockville, Maryland, this 24th day of December 1996.

For the Nuclear Regulatory Commission.

John F. Stolz,

Director, Project Directorate I–2, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 96–33253 Filed 12–30–96; 8:45 am]

BILLING CODE 7590–01–P

Advisory Committee on Nuclear Waste Seeking Qualified Candidates

AGENCY: Nuclear Regulatory Commission.

ACTION: Request for résumés.

SUMMARY: The Nuclear Regulatory Commission (NRC) is seeking qualified candidates for possible appointment to its Advisory Committee on Nuclear Waste (ACNW). One opening is expected on the committee in mid-1997.

ADDRESSES: Submit résumés to: Ms. Jude Himmelberg, Office of Personnel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

FOR FURTHER INFORMATION, CALL: 1–800–952–9678. Please refer to Announcement Number 97–1002.

SUPPLEMENTARY INFORMATION: The ACNW is a part-time advisory group established by the NRC in 1988 to provide independent technical review and advice on the disposal of nuclear waste, including all aspects of nuclear waste disposal facilities, as directed by the Commission. This advice covers activities related to licensing, operation, and closure of high-level and low-level radioactive waste disposal facilities and associated rulemakings, regulatory guides and NRC staff technical positions. The ACNW also reviews performance assessment evaluations of waste disposal facilities.

The committee interacts with representatives of the NRC, the Advisory Committee on Reactor Safeguards, the Department of Energy, other Federal, State, and local agencies, Indian Nations, and private organizations as appropriate.

A wide variety of engineering and scientific skills are needed to conduct the broad-based reviews required in the committee's work. Engineers and scientists are needed with work experience in the high-level and low-level radioactive waste disposal programs coupled with broad experience in a pertinent technical field, such as nuclear engineering and technology, nuclear fuel cycle analysis, geoscience, chemistry, and materials science.

Applicants should have a minimum of 20 years' work experience in related fields, or fields that can be applied directly to the work of the committee, including graduate level education. In addition to the length of the work experience, applicants should have achieved a level of distinction in their discipline and must be able to devote approximately 50-100 days per year to committee business. Most meetings are held in Rockville, Maryland. Some additional travel is required to other sites.

NRC regulations and policies restrict the participation of members in areas where these members have conflicts of interest. The degree to which an individual's participation in ACNW activities will be restricted is considered in the selection progress. Each qualified candidate's financial interests must be reconciled with applicable Federal and NRC rules and regulations prior to final appointment. This might require divestiture of securities issued by nuclear industry entities, or discontinuance of industry-funded research contracts or grants.

A résumé describing the educational and professional background of the candidate, including special accomplishments, professional references, and current address and telephone number should be provided. All qualified candidates will receive careful consideration. Appointment will be made without regard to race, color, religion, national origin, sex, age, or disabilities. Candidates must be citizens of the United States. Applications will be accepted until February 20, 1997.

Dated: December 24, 1996.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 96–33246 Filed 12–30–96; 8:45 am]

BILLING CODE 7590–01–P

Proposed Generic Communication; Degradation of Steam Generator Internals

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of opportunity for public comment.

SUMMARY: The Nuclear Regulatory Commission (NRC) is proposing to issue a generic letter concerning the degradation of steam generator internals at foreign pressurized-water reactor facilities. The purpose of the proposed generic letter is to (1) re-alert addressees to the previously communicated findings of damage to steam generator internals, namely, tube support plates and tube bundle wrappers, at foreign PWR facilities; (2) emphasize to addressees the importance of performing comprehensive examinations of steam generator internals to ensure steam generator tube structural integrity is maintained in accordance with the requirements of Appendix B to 10 CFR Part 50; and (3) request all addressees to submit information that will enable the NRC staff to verify whether or not the condition of addressees' steam generator

internals comply and conform with the current licensing basis for their respective facilities. The NRC is seeking comment from interested parties regarding both the technical and regulatory aspects of the proposed generic letter presented under the **SUPPLEMENTARY INFORMATION** heading.

The proposed generic letter was endorsed by the Committee to Review Generic Requirements (CRGR) on December 17, 1996. The relevant information that was sent to the CRGR will be placed in the NRC Public Document Room. The NRC will consider comments received from interested parties in the final evaluation of the proposed generic letter. The NRC's final evaluation will include a review of the technical position and, as appropriate, an analysis of the value/impact on licensees. Should this generic letter be issued by the NRC, it will become available for public inspection in the NRC Public Document Room.

DATES: Comment period expires January 30, 1997. Comments submitted after this date will be considered if it is practical to do so, but assurance of consideration cannot be given except for comments received on or before this date.

ADDRESSES: Submit written comments to Chief, Rules Review and Directives Branch, U.S. Nuclear Regulatory Commission, Mail Stop T–6D–69, Washington, DC 20555–0001. Written comments may also be delivered to 11545 Rockville Pike, Rockville, Maryland, from 7:30 am to 4:15 pm, Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, 2120 L Street, N.W. (Lower Level), Washington, D.C.

FOR FURTHER INFORMATION CONTACT: Stephanie M. Coffin, (301) 415–2778.

SUPPLEMENTARY INFORMATION:

NRC Generic Letter 96–XX: Degradation of Steam Generator Internals

Addressees

All holders of operating licenses for pressurized water reactors (PWRs), except those licenses that have been amended to possession-only status.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to (1) re-alert addressees to the previously communicated findings of damage to steam generator internals, namely, tube support plates and tube bundle wrappers, at foreign PWR facilities; (2) emphasize to addressees the importance of performing comprehensive examinations of steam generator internals to ensure steam generator tube structural integrity is maintained in accordance with the requirements of Appendix B to 10 CFR Part 50; and (3) request all addressees to submit information that will enable the NRC staff to verify whether or not the condition of addressees' steam generator internals comply and conform with the current licensing basis for their respective facilities.

Background

The NRC issued Information Notice (IN) 96–09 and IN 96–09, Supplement 1 to alert addressees to findings of damage to steam generator internals at foreign PWR facilities.

Description of Circumstances

Foreign authorities have reported various steam generator tube support plate damage mechanisms. The affected steam generators are similar, but not identical, to Westinghouse Model 51 steam generators. As previously documented in IN 96–09 and IN 96–09, Supplement 1, one damage mechanism involved the wastage of the uppermost support plate caused by the misapplication of a chemical cleaning process. A second damage mechanism involved broken tube support plate ligaments at the uppermost, and sometimes at the next lower, tube support plates. The support plate ligaments broke near a radial seismic restraint and near an antirotation key; the damage apparently dates back to initial startup of the affected plants. According to foreign authorities, the ligaments may have broken because of excessive stress during the final thermal treatment of the monobloc steam generators, which in turn was caused by inadequate clearance for differential thermal expansion between the support plates, wrapper, and seismic restraints.

As previously documented in IN 96–09, Supplement 1, a third damage mechanism involved wastage not associated with chemical cleaning and affected tube support plates at various elevations. This damage mechanism is active (progressive) and apparently involves a corrosion or erosion-corrosion mechanism of undetermined origin.

The staffs of potentially affected foreign reactors are currently inspecting steam generators for evidence of the various damage mechanisms, both visually and with eddy current testing. Tubes without adequate lateral support are being plugged.

In 96–09, Supplement 1, also documented that cooling transients involving the injection of large quantities of auxiliary feedwater may have been a key factor in the steam generator wrapper drop phenomenon observed at a foreign PWR facility. These cooling transients are believed to have been particularly severe for two units as a result of the use of a special operating procedure to accelerate the transition from hot to cold shutdown. The weight of the wrapper assembly and support plates is borne by six tenons mounted on the steam generator shell. The wrapper is nominally free to expand axially relative to the shell. However, it is postulated that an interference fit developed between the wrapper and the seismic restraints (mounted to the shell) as a result of differential thermal expansion associated with the cooling transients at the seventh support plate elevation. This interference fit prevented axial expansion of the wrapper, which led to excessive vertical bearing loads at the tenon supports, thus causing localized wrapper failure at this location and downward displacement of the wrapper (20 millimeters, maximum). Poor quality wrapper support welds may also have contributed to this failure. Repairs have been implemented at the affected foreign PWR facility. Wrapper dropping is being monitored in all steam generators of similar design. The monitoring is through online instrumentation and through visual inspections during outages. In addition to the wrapper dropping problem, cracking of the wrapper above the original upper support was discovered at the same foreign unit. The cause of the cracking is not yet known.

Discussion

The reported foreign experience highlights the potential for degradation mechanisms that may lead to tube support plate and tube bundle wrapper damage. The steam generator tube support plates support the tubes against lateral displacement and vibration and minimize bending moments in the tubes in the event of an accident. Support plate damage can impair their ability to perform this function and, thus, could potentially lead to the impairment of tube integrity. Vibration-induced fatigue could present a potential problem if tube support plates lose integrity, particularly in areas of high secondary side crossflows. As previously noted in IN 96–09, tube support plate signal anomalies found during eddy current testing of the steam generator tubes may be indicative of support plate damage or ligament cracking. Certain visual and video camera inspections on the secondary side of a steam generator may also provide useful information concerning the degradation of steam generator internals. The NRC staff will continue to monitor information on tube support plate and tube bundle wrapper damage as it becomes available from foreign authorities.

This letter also alerts addressees to the importance of performing comprehensive examinations of steam generator internals to ensure steam generator tube structural integrity is maintained in accordance with the requirements of Appendix B to 10 CFR Part 50. Criterion XI of Appendix B, ''Test Control,'' requires, in part, that a test program be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in the applicable design documents. The applicable steam generator tube design documents include General Design Criteria (GDCs) 14, 15, 30, 31, and 32 of 10 CFR Part 50, Appendix A and Section III of the ASME Boiler and Pressure Vessel code. Criterion XVI of Appendix B, ''Corrective Action,'' requires in part that measures be established to assure that conditions adverse to quality are promptly identified and corrected.

Requested Information

Within 60 days of the date of this generic letter, each addressee is requested to provide a written report that includes the following information for its facility:

(1) Discussion of the program in place, if any, to detect degradation of steam generator internals and a description of the inspection plans, including the inspection scope, frequency, methods, equipment and criteria, and plans for corrective action in the event degradation is found.

The discussion should include the following information:

(a) Whether past inspection records at the facility have been reviewed for indications of tube support plate signal anomalies from eddy current testing of the steam generator tubes that may be indicative of support plate damage or ligament cracking. If the addressee has performed such a review, include a discussion of the findings.

(b) Whether visual or video camera inspections on the secondary side of the steam generators have been performed at the facility to provide information on the condition of steam generator internals (e.g., support plates, tube bundle wrappers, or other components). If the addressee has performed such inspections, include a discussion of the findings.

(c) Whether degradation of steam generator internals has been detected at the facility, and how the degradation was assessed and dispositioned.

(2) If the addressee currently has no program in place to detect degradation of steam generator internals, the written response should include a discussion of the plans for establishing such a program, or a justification as to why no such program is needed.

Addressees are encouraged to work closely with industry groups on the coordination of inspections, evaluations, and repair options for all types of steam generator degradation that may be found.

The NRC is aware that the industry has developed generic industry guidance on performing steam generator inspections, and that this guidance is continually being updated. If an addressee intends to follow the guidance developed by the industry for this issue, reference to the relevant generic guidance documents is acceptable, and encouraged, as part of the response, as long as the referenced documents have been officially submitted to the NRC. However, additional plant-specific information will be needed.

Required Response

Within 30 days of the date of this generic letter, each addressee is required to submit a written response indicating: (1) Whether or not the requested information will be submitted and (2) whether or not the requested information will be submitted within the requested time period. Addressees who choose not to submit the requested information, or are unable to satisfy the requested completion date, must describe in their response any alternative course of action that is proposed to be taken, including the basis for the acceptability of the proposed alternative course of action.

NRC staff will review the responses to this generic letter and if concerns are identified, affected addressees will be notified.

Address the required written responses to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555-0001, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f).

Backfit Discussion

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), this generic letter transmits an information request for the purpose of verifying compliance with applicable existing regulatory requirements. Specifically, the requested information will enable the NRC staff to determine whether or not the condition of the addressees' steam generator internals comply and conform with the current licensing basis for their respective facilities. In particular, it would help ascertain whether or not the regulatory requirements pursuant to Appendix B to 10 CFR Part 50 are met, namely, (1) Criterion XI, ''Test Control,'' concerning the establishment of effective test programs for systems, structures and components, and (2) Criterion XVI, ''Corrective Action,'' which requires that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Additionally, no backfit is either intended or approved in the context of issuance of this generic letter. Therefore, the staff has not performed a backfit analysis.

Dated at Rockville, Maryland, this 23rd day of December 1996.

For the Nuclear Regulatory Commission.

David B. Matthews,

Acting Director, Division of Reactor Program Management, Office of Nuclear Reactor Regulation.

[FR Doc. 96–33250 Filed 12–30–96; 8:45 am]

BILLING CODE 7590–01–P

Proposed Generic Communication; Steam Generator Tube Inspection Techniques (M96401)

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of opportunity for public comment.

SUMMARY: The Nuclear Regulatory Commission (NRC) is proposing to issue a generic letter concerning steam generator tube inspection practices at pressurized-water reactor facilities. The purpose of the proposed generic letter is to (1) emphasize to addressees the importance of performing steam generator tube inservice inspections using qualified techniques in accordance with the requirements of Appendix B to 10 CFR Part 50, and (2) request certain information from addressees to verify whether or not steam generator tube inservice inspection practices comply and conform with the current licensing basis for their respective facilities. The NRC is seeking comment from interested