Week of August 23, 2004—Tentative

There are no meetings scheduled for the week of August 23, 2004.

Week of August 30, 2004—Tentative

There are no meetings scheduled for the week of August 30, 2004.

Week of September 6, 2004—Tentative

Wednesday, September 8, 2004

9:30 a.m. Discussion of Office of Investigations (OI) Programs and Investigations (Closed—Ex.7).

2 p.m. Discussion of Intragovernmental Issues (Closed—Ex. 1 & 9).

*The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Dave Gamberoni, (301) 415–1651.

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The NRC Commission Meeting Schedule can be found on the Internet at: *http://www.nrc.gov/what-we-do/policy-making/schedule.html*

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The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (*e.g.* braille, large print), please notify the NRC's Disability Program Coordinator, August Spector, at 301–415–7080, TDD: 301–4152100, or by e-mail at *aks@nrc.gov*. Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

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This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov*.

Dated: July 29, 2004.

Dave Gamberoni,

Office of the Secretary.

[FR Doc. 04–17710 Filed 7–30–04; 9:55 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to section 189a.(2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. The Act requires the Commission publish notice of any amendments issued, or proposed to be issued and grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from July 9, 2004 through July 22, 2004. The last biweekly notice was published on July 20, 2004 (69 FR 43457).

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would 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 from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination. Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene.

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example in derating or shutdown of the facility. Should the Commission take action prior to the expiration of either the comment period or the notice period, it will publish in the **Federal Register** a notice of issuance. Should the Commission make a final No Significant Hazards Consideration Determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed within 60 days, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements: (1) The name, address and telephone number of the requestor or petitioner; (2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also set forth the specific contentions which the petitioner/requestor seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner/requestor intends to rely in proving the contention at the hearing. The petitioner/requestor must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner/requestor intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner/requestor to relief. A petitioner/requestor who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing.

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff; (3) E-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *HEARINGDOCKET@NRC.GOV*; or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov*. A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

Nontimely requests and/or petitions and contentions will not be entertained absent a determination by the Commission or the presiding officer of the Atomic Safety and Licensing Board that the petition, request and/or the contentions should be granted based on a balancing of the factors specified in 10 CFR 2.309(a)(1)(i)–(viii).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov*.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois, Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of amendment request: May 21, 2004.

Description of amendment request: The proposed amendment would revise Technical Specifications 5.6.6, ''Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR),'' by adding a reference to the use of previous Nuclear Regulatory Commission approved Code Cases N–640 and N–588 as acceptable methods for determining reactor pressure vessel (RPV) pressure temperature (P–T) limits.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated?

The use of Code Cases N–588 and N–640 has been approved for Braidwood and Byron Stations. The use of P–T limits based on these Code Cases will continue to ensure that the RPV integrity is maintained under all conditions.

Thus there is no increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not involve the use or installation of new equipment. No equipment will be operated in a new or different manner. No new or different system interactions are created and no new processes are introduced. The proposed change will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases.

Based on this evaluation, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed TS change does not involve a significant reduction in a margin of safety?

The P–T limits provide assurance that RPV integrity is maintained. The use of Code Cases N–588 and N–640 has been previously approved by the NRC for Braidwood and Byron Stations and will continue to ensure that RPV integrity is maintained.

Thus, there is no reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Edward J. Cullen, Deputy General Counsel, Exelon BSC-Legal, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1 (BVPS–1), Beaver County, Pennsylvania

Date of amendment request: June 28, 2004.

Description of amendment request: The proposed amendment would revise the BVPS–1 Technical Specification (TS) 4.4.5.4.a.8 to modify the definition of steam generator (SG) tube inspection to exclude the portion of the tube within the tube sheet below the W* distance. The W* distance is defined as the distance from the top of the tube sheet to the bottom of the W* length (7.0 in. on the hot leg side) including the distance from the top of the tube sheet to the bottom of the WEXTEX (Westinghouse explosive tube expansion) Transition (approximately 0.25 in.) plus uncertainties (0.12 in.). The proposed amendment would also revise the SG tube repair criteria of TS 4.4.5.4.a.6 to indicate that service-induced degradation within the W* distance or less than 8.0 in. below the top of the tube sheet shall be repaired upon detection. The proposed amendment would also add TS 4.4.5.2.e to require a 100% rotating pancake coil probe inspection of the hot leg tube sheet W* distance, add new W* terminology definitions in TS 4.4.5.4.a.11, and add a new reporting criteria for W* inspection information to TS 4.4.5.5.d.1 and TS 4.4.5.5.e. This proposed amendment would be effective for only one operating cycle, as the licensee plans to replace SGs during the 2006 refueling outage.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

No. The proposed change modifies the [BVPS–1] TSs to incorporate steam generator (SG) tube inspection scope based on WCAP–14797, Revision 2 [''Generic W* Tube Plugging Criteria for 51 Series Steam Generator Tubesheet Region WEXTEX Expansions,'' dated March 2003 (proprietary)]. Of the various accidents evaluated in the [BVPS–1] Updated Final Safety Analysis Report (UFSAR), the proposed changes only affect the steam generator tube rupture (SGTR) event evaluation and the postulated steam line break (SLB) accident evaluation. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a safe shutdown earthquake (SSE); however, the seismic analysis of Series 51 steam generators has shown that axial loading of the tubes is negligible during an SSE.

For the SGTR event, the required structural margins of the steam generator tubes will be maintained by the presence of the tubesheet. Tube rupture is precluded for cracks in the Westinghouse explosive tube expansion (WEXTEX) region due to the constraint provided by the tubesheet. Therefore, Regulatory Guide (RG) 1.121, ''Bases for Plugging Degraded PWR [pressurized-water reactor] Steam Generator Tubes,'' margins against burst are maintained for both normal and postulated accident conditions.

The W* length supplies the necessary resistive force to preclude pullout loads under both normal operating and accident conditions. The contact pressure results from the WEXTEX expansion process, thermal expansion mismatch between the tube and tubesheet and from the differential pressure between the primary and secondary side. The proposed changes do not affect the other systems, structures, components or operational features. Therefore, the proposed change results in no significant increase in the probability of the occurrence of an SGTR or SLB accident.

The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event. Primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed change since the tubesheet enhances the tube integrity in the region of the WEXTEX expansion by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and collapse is strengthened by the tubesheet in that region. At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) below the W* length is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region.

SLB leakage is limited by leakage flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of crack face opening compared to free span indications. The total leakage, that is, the combined leakage for all such tubes meet[s] the industry performance criterion, plus the combined leakage developed by any other alternate repair criteria, will be maintained below the maximum allowable SLB leak rate limit, such that off-site doses are maintained less than 10 CFR 100 guideline values and the limits evaluated in the [BVPS–1] UFSAR.

Therefore, based on the above evaluation, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed changes do not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity is expected to be maintained for all plant conditions upon implementation of the W* methodology.

The proposed changes do not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. The proposed changes maintain the required structural margins of the steam generator tubes for both normal and accident conditions. NRC [Nuclear Regulatory Commission] Regulatory Guide (RG) 1.121 is used as the basis in the development of the W* methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC staff for meeting General Design Criteria 14, 15, 31, and 32 by reducing the probability and consequences of an SGTR. RG 1.121 concludes that by determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service or repaired, the probability and consequences of a[n] SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society for Mechanical Engineers (ASME) [Boiler and Pressure Vessel] Code.

For primarily axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. WCAP–14797, Revision 2, defines a length, W*, of degradation free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces (with applicable safety factors applied). Application of the W* criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the W* criteria.

Plugging of steam generator tubes reduces the reactor coolant flow margin for core cooling. Implementation of W* methodology at [BVPS–1] will result in maintaining the margin of flow that may have otherwise been reduced by tube plugging.

Based on the above, it is concluded that the proposed changes do not result in a significant reduction [in a margin of safety] as defined in the [UFSAR] or [B]ases of the plant [TSs].

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for Licensee: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Richard J. Laufer.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit 3 Nuclear Generating Plant, Citrus County, Florida

Date of amendment request: July 8, 2004.

Description of amendment request: The amendment request proposes to delete one-time use footnotes that have expired or have already been used from the Crystal River Unit 3 (CR–3) Improved Technical Specifications (ITS). Specifically, obsolete notes will be removed from ITS 3.8.1, ''AC Sources—Operating (Emergency Diesel Generator),'' ITS 3.7.9, ''Nuclear Services Seawater System,'' and ITS 3.7.18, ''Control Complex Cooling System.'' This change is administrative in nature and does not alter any operating license requirements.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below and states that the amendment request:

1. Does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Each footnote was added to ITS through the license amendment process. The activities supported by the footnotes were performed and, therefore, the footnotes have no further utility. Deleting the footnotes is administrative in nature and does not affect plant conditions that could impact accident probability or consequences. Therefore, granting this LAR [license amendment request] does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does not create the possibility of a new or different type of accident from any accident previously evaluated.

The proposed license amendment deletes footnotes that were used on a one-time basis for several specifications. The proposed LAR will not result in changes to the design, physical configuration of the plant or the assumptions made in the safety analysis. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does not involve a significant reduction in the margin of safety.

The deletion of the footnotes from the ITS does not affect properties of plant components or their operation. Therefore, granting this LAR does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Steven R. Carr, Associate General Counsel—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Acting Section Chief: Michael L. Marshall, Jr.

Indiana Michigan Power Company, Docket Nos. 50–315, Donald C. Cook Nuclear Plant, Unit 1, Berrien County, Michigan

Date of amendment request: June 25, 2004.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) and the bases to reduce the temperature at which shutdown and control rod drop tests are performed from greater than or equal to 541 degrees Fahrenheit to greater than or equal to 500 degrees Fahrenheit. Additionally, the proposed amendment would make format changes to improve the TS appearance.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

Response: No.

The probability of occurrence of an accident previously evaluated is not altered by the proposed amendment. The proposed change does not impact the integrity of the reactor coolant system pressure boundary and, therefore, does not increase the potential for the occurrence of a loss-of-coolant accident. The change does not make any physical changes to the facility design, material or construction standards, and the proposed change is not an initiator or contributor to any currently evaluated accident. The format changes are intended to improve appearance, and do not alter any requirements. Thus, neither the probability nor the consequences of a previously analyzed accident are significantly increased.

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

Response: No.

The rod drop test is routinely performed during each refueling outage. Decreasing the test temperature will not create the possibility of a new or different accident. The proposed test conditions remain bounded by the analysis of record since the rod drop time assumed in the accident analysis will not be changed. The format changes are intended to improve appearance, and do not alter any requirements. Since no new failure modes are associated with the proposed changes, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The Technical Specification change does not involve a significant reduction in margin because the acceptance criterion for the rod drop time will not change. The proposed change will reduce the minimum rod drop test temperature from greater than or equal to 541 degrees Fahrenheit to greater than or equal to 500 degrees Fahrenheit. This will slightly increase the measured test rod drop time. The measured test rod drop time, however, will be within the current Technical Specification limit of 2.4 seconds. The format changes are intended to improve appearance, and do not alter any requirements. Therefore, the margin of safety is not impacted by the proposed amendment.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: L. Raghavan.

Nebraska Public Power District, Docket No. 50–298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: July 15, 2004.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) Section 3.8.1, AC Sources—Operating, Condition B, to extend the allowed outage time for one Diesel Generator (DG) inoperable from 7 days to 14 days and TS Section 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air, Limiting Condition for Operation, to allow the use of temporary fuel oil storage tanks to supply the required fuel oil storage inventory.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The Standby AC Power System (Diesel Generators) provides onsite electrical power to vital systems should offsite electrical power be interrupted. It is not an initiator to any accident previously evaluated. Therefore, the extended period of operation with one diesel generator inoperable and the seven day required fuel oil supply being provided in part by temporary storage tanks will not increase the probability of an accident previously evaluated.

The Standby AC Power System acts to mitigate the consequences of design basis accidents that assume a loss of offsite power. For that purpose, redundant diesel generators are provided to protect against a single failure. During the Technical Specification seven day allowed outage time, an operating unit is allowed by the Technical Specifications to remove one diesel generator from service, thereby losing this single failure protection. During the requested fourteen day allowed outage time for fuel oil storage tank cleaning and coating maintenance activities, the inoperable diesel generator will be maintained available to start and load, with a minimum of five (5) hours of fuel available in the day tank. Manual actions contained in approved procedures to provide fuel from temporary storage tanks to either the operable diesel generator or the inoperable but available diesel generator will be implemented. A risk evaluation determined that the probability of failure to implement the contingency actions is sufficiently low that it does not adversely impact the availability of the Standby AC Power System.

The vulnerability to external events, seismic, high winds and fire, was also evaluated and judged to be not significant due to the low probability of these events during the period of time this proposed amendment will be in effect, and the defense in depth strategies being put in place during the tank maintenance activities.

In the event that fuel stored in the temporary tanks is not available to support full load operation of the diesel generator beyond four (4) days, replenishment of fuel oil from offsite can be accomplished in approximately 24 hours through the use of existing purchase orders for fuel oil and diesel fuel analysis. Therefore, during the period of the extended allowed outage time and the use of temporary fuel oil storage tanks, there is no significant increase in the consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

Operation with one diesel generator inoperable but available for an extended period or with part of the required diesel fuel stored in temporary tanks does not involve any new mode of plant operation or different function for plant equipment. Operation in this configuration does introduce proceduralized manual actions to supply fuel to either diesel generator from the permanent storage tank or the temporary tank. These actions can be accomplished within the five hours of full load diesel operation from fuel stored in the day tank. A risk evaluation determined that the probability of failure to implement the contingency actions is sufficiently low that it does not adversely impact the availability of the Standby AC Power System. There are no new accident precursors generated due to this temporary extension of allowed outage time or the use of a temporary fuel oil storage system.

3. Do the proposed changes involve a significant reduction in the margin of safety?

Response: No.

A single failure of the operable fuel oil transfer pump could prevent DG operation beyond five hours. Proceduralized manual actions to supply fuel to either diesel generator from the permanent storage tank or the temporary tank will be implemented to mitigate this single failure vulnerability. These actions can be accomplished within the five (5) hours of full load diesel operation from fuel stored in the day tank. A risk evaluation determined that the probability of failure to implement the contingency actions is sufficiently low that it does not adversely impact the availability of the Standby AC Power System. Therefore, during the extended allowed outage time and the use of a temporary fuel oil storage system, the Standby AC Power System maintains the ability to provide a source of on-site AC power adequate for maintaining the safe shutdown of the reactor following abnormal operational transients and postulated accidents.

IEEE [Institute of Electrical and Electronics Engineers] Design Standard 308–1970, ''IEEE Criteria for Class 1E Electric Systems for Nuclear Power Generating Station,'' Section 5.2.4, ''Standby Power Supply,'' Paragraph 6), ''Energy Storage,'' contains the requirement for stored energy capacity to be the longer of (a) seven days or (b) time required to replenish the energy from sources away from the generating unit's site following the limiting design basis event. Cooper Nuclear Station's Updated Safety Analysis Report documents that the Standby AC Power System conforms to the applicable sections of IEEE 308–1970.

The Diesel Generator Diesel Oil Storage and Transfer System will be configured to ensure a minimum fuel oil inventory to support greater than four (4) days of full load diesel generator operation is maintained in the operable permanent storage tank. Existing crosstie capabilities in the fuel storage and transfer system piping, in conjunction with proceduralized manual actions, ensure the four day fuel supply is available to either diesel generator. The remaining three (3) day fuel supply will be stored in temporary non-Class I tanks and would potentially be vulnerable to external events. The vulnerability to external events, seismic, high winds and fire, was evaluated and judged to be not significant due to the low probability of these events during the period of time this proposed amendment will be in effect, and the defense in depth strategies being put in place during the tank maintenance activities.

In the event that fuel stored in the temporary tanks is not available to support full load operation of the diesel generator beyond four (4) days, replenishment of fuel oil from offsite can be accomplished in approximately 24 hours through the use of existing purchase orders for fuel oil and diesel fuel analysis.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: Robert A. Gramm.

Pacific Gas and Electric Company, Docket No. 50–133, Humboldt Bay Power Plant, Unit 3, Humboldt County, California

Date of amendment request: June 8, 2004.

Description of amendment request: The Humboldt Bay Power Plant, Unit 3, is a decommissioning nuclear power plant that was permanently shutdown in July 1976. The plant is currently in a safe storage (SAFSTOR) condition to ensure that necessary plant systems will be operated and maintained as needed to preserve safe conditions within the facility to prevent deterioration until active decommissioning can commence. All spent fuel is stored in the spent fuel pool. Pacific Gas and Electric Company (PG&E) has proposed a license amendment to clarify the technical specifications applicability to current plant conditions and practices. Specifically, the requested changes clarify that:

(1) Fuel fragments within the spent fuel pool totaling less than one fuel assembly and damaged fuel assembly UD–6N do not have to be stored in containers made of neutron absorbing material. Furthermore, that one additional assembly can be removed from a neutron absorbing container to perform fuel handling activities.

(2) The control station for Humboldt Bay Units 1 and 2 is considered to be anywhere on the +27 foot operating deck.

(3) References to certain technical specification section designators that contain typographical errors have been corrected.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

No. The proposed changes provide either clarification to reflect plant conditions or correct typographical errors. Existing accident analysis assumptions bound the proposed addition of not storing fuel fragments, which may be considered as less than or equal to a fuel assembly, in a container made with neutron absorbing material. The proposed changes involve no changes to plant systems or accident analysis, and as such, do not affect initiators of analyzed events or assumed mitigation of accidents. Therefore, the proposed changes do not increase the probability or consequences of an accident previously evaluated.

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

No. The proposed changes provide either clarification to reflect plant conditions or correct typographical errors. Existing accident analysis assumptions bound the proposed addition of not storing fuel fragments, which may be considered as less than or equal to a fuel assembly, in a container made with neutron absorbing material. The proposed changes do not involve a physical alteration to the plant, add any new equipment, or require existing equipment to be operated in a manner different from the present design. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident evaluated.

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

No. The proposed changes provide either clarification to reflect existing plant conditions or correct typographical errors. Existing accident analysis assumptions bound the proposed addition of not storing fuel fragments, which may be considered as less than or equal to a fuel assembly, in a container made with neutron absorbing material. They have no effect on plant equipment, operating practices or safety analysis assumptions. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Richard F. Locke, Esquire, Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Claudia Craig.

Pacific Gas and Electric Company, Docket No. 50–133, Humboldt Bay Power Plant, Unit 3, Humboldt County, California

Date of amendment request: June 23, 2004.

Description of amendment request: The Humboldt Bay Power Plant, Unit 3, is a decommissioning nuclear power plant that was permanently shutdown in July 1976. The plant is currently in a safe storage (SAFSTOR) condition to ensure that necessary plant systems will be operated and maintained as needed to preserve safe conditions within the facility to prevent deterioration until active decommissioning can commence. All spent fuel is stored in the spent fuel pool. Currently, the facility operating license only allows maintaining the facility in SAFESTOR. At the time the license condition for SAFSTOR was specified, Pacific Gas and Electric Company (PG&E), the licensee, had intended to maintain SAFSTOR until the Department of Energy (DOE) established a permanent repository for spent fuel. The licensee has recently reassessed its near-term options for the facility and in December of 2003 applied for a license to store its spent fuel in an onsite dry cask independent spent fuel storage installation (ISFSI). Moving the spent fuel to an ISFSI would permit the licensee to begin significant decommissioning activities. Consequently, PG&E has submitted a license amendment request to permit the licensee to proceed with decontamination and decommissioning activities in accordance with applicable NRC requirements and the regulations for decommissioning reactors in 10 CFR 50.82.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

No. The proposed change eliminates the restriction to remain in SAFSTOR status, and allows PG&E to take actions necessary to decommission and decontaminate the facility in accordance with NRC regulations. The proposed change involves no changes to plant systems or accident analysis, and as such, do not affect initiators of analyzed events or assumed mitigation of accidents. Therefore, the proposed changes do not increase the probability or consequences of an accident previously evaluated.

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

No. The proposed change eliminates the restriction to remain in SAFSTOR status, and allows PG&E to take actions necessary to decommission and decontaminate the facility in accordance with NRC regulations. The proposed change does not involve a physical alteration to the plant, add any new equipment, or require existing equipment to be operated in a manner different from the present design. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident evaluated.

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

No. The proposed change eliminates the restriction to remain in SAFSTOR status, and allows PG&E to take actions necessary to decommission and decontaminate the facility in accordance with NRC regulations. The proposed change has no effect on plant equipment, operating practices or safety analysis assumptions. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Richard F. Locke, Esquire, Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Claudia Craig.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: March 31, 2004.

Description of amendment request: The proposed change will allow operation in regions of the power/flow map currently restricted by the requirements of interim corrective actions (ICAs) and certain limiting conditions for operations (LCOs) of Technical Specification 3.4.1. The oscillation power range monitor (OPRM) will allow operations in the regions restricted by the administrative controls mentioned above by using inputs from the local power range monitoring (LPRM) system to monitor core conditions and generate a reactor protection system (RPS) trip when required to prevent a violation of the minimum critical power ratio (MCPR) safety limit.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the three standards of 10 CFR 50.92(c). The NRC staff's analysis is presented below:

1. Does the Proposed Change Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated?

The proposed change would allow operation in regions of the power/flow map currently restricted by administrative controls. The purpose of the administrative controls were to ensure adequate capability to detect and suppress conditions consistent with the onset of a thermal-hydraulic (T–H) event which is postulated to cause a violation of the MCPR safety limit. The mitigation of a T–H instability event will be ensured by the RPS trip signal generated by the OPRM prior to challenging the MCPR safety limit. Since automatic protective functions of the OPRM will be replacing administrative controls which require operator action, the probability or consequence of a T–H instability event is not significant. Therefore, the proposed change does not result in a significant increase in the probability or consequence of an accident previously evaluated.

2. Does the Proposed Change Create the Possibility of a New or Different Kind of Accident From any Accident Previously Evaluated?

The proposed change would allow operation in regions of the power/flow map currently restricted by administrative controls. The OPRM system uses inputs from the LPRMs to monitor core conditions and generate a RPS trip when required. Quality requirements for software design, testing, implementation and module self-testing of the OPRM system provide assurance that no new equipment malfunctions due to software errors are created. The design of the OPRM system also ensures that neither operation nor malfunction of the OPRM system will adversely impact the operation of other systems, and no accident or equipment malfunction of these other systems could cause the OPRM system to malfunction or cause a different kind of accident. Therefore, operation with the OPRM system does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the Proposed Change Involve a Significant Reduction in a Margin of Safety?

The proposed change would allow operation in regions of the power/flow map currently restricted by administrative controls. The margin of safety for the unmitigated T–H instability event will not be significantly reduced due to the capability of the OPRM to automatically detect and suppress conditions which might result in an MCPR safety limit violation. The automatic functions of the OPRM will be replacing administrative controls which rely on operator action to prevent an unmitigated T–H instability event. The OPRM will maintain the margin of safety while significantly reducing the burden on the control room operators. Therefore, operation with the OPRM system does not involve a significant reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Date of amendment requests: June 29, 2004.

Description of amendment requests: The proposed amendments would revise the Technical Specifications (TS) to implement the following miscellaneous changes: (1) Revise the reporting period of TS 2.2.5 from 30 days to 60 days for the safety limit violations Licensee Event Report, (2) revise the frequency of Surveillance Requirement (SR) 3.4.3.1.2 of TS 3.4.3.1, ''Pressurizer Heatup and Cooldown Limits,'' to reflect pressurizer spray cyclic limits being governed by the temperature differentials between the spray nozzle and the spray line, (3) revise TS 5.5.2.11.f.1 of TS 5.5.2.11, ''Steam Generator (SG) Tube Surveillance Program,'' to correct typographical errors, (4) remove TS 5.5.2.14, ''Configuration Risk Management Program (CRMP),'' in accordance with **Federal Register** Notice Vol. 64, No. 137 (July 19, 1999), and (5) revise TS 5.7.1.5, ''Core Operating Limits Report (COLR),'' to delete revision numbers and dates from the referenced documents in this section consistent with the NRC-approved industry Technical Specifications Task Force (TSTF) Standard Technical Specifications Traveler number TSTF–

363, ''Revise Topical Report References in ITS (Improved Technical Specifications) 5.6.5 COLR,'' and incorporate editorial corrections.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Southern California Edison (SCE) proposes to modify the San Onofre Units 2 and 3 Technical Specifications (TS) to accomplish several improvements by providing consistency with current Code of Federal Regulations (CFR) Licensee Event Report (LER) reporting requirements, clarifying a pressurizer heatup/cooldown Surveillance Requirement, TS editorial corrections, removing TS redundancy to the Maintenance Rule in accordance with **Federal Register** Notice Vol. 64, No. 137 (July 19, 1999), and eliminating need for TS amendment requests for cited Core Operating Limits Report (COLR) reference revisions consistent with the NRC approved Industry Technical Specifications Task Force (TSTF) Standard Technical Specifications Traveler number TSTF–363, ''Revise Topical Report References in ITS (Improved Technical Specifications) 5.6.5 COLR.'' These proposed changes do not involve any change in the design or operation of the plant. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Modifying the Technical Specifications to provide consistency with current CFR LER reporting requirements, clarify a pressurizer heatup/cooldown Surveillance Requirement, incorporate editorial corrections, remove TS redundancy to the Maintenance Rule in accordance with **Federal Register** Notice Vol. 64, No. 137 (July 19, 1999), and to eliminate need for TS amendment requests for cited COLR reference revisions consistent with the NRC approved Industry Technical Specifications Task Force (TSTF) Standard Technical Specifications Traveler number TSTF–363, ''Revise Topical Report References in ITS (Improved Technical Specifications) 5.6.5 COLR'' does not involve any change in the design or operation of the plant. Therefore, a possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. The proposed change does not involve a significant reduction in a margin of safety.

Evaluation of these proposed modifications to the Technical Specifications to provide consistency with current CFR LER reporting requirements, clarify a pressurizer heatup/cooldown Surveillance Requirement, incorporate editorial corrections, remove TS redundancy to the Maintenance Rule in accordance with **Federal Register** Notice Vol. 64, No. 137 (July 19, 1999), and to eliminate need for TS amendment requests for cited COLR reference revisions consistent with the NRC approved Industry Technical Specifications Task Force (TSTF) Standard Technical Specifications Traveler number TSTF–363, ''Revise Topical Report References in ITS (Improved Technical Specifications) 5.6.5 COLR'' does not involve any change in the design or operation of the plant and therefore does not create any reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Date of amendment requests: June 30, 2004.

Description of amendment requests: The proposed amendments would revise Technical Specification (TS) 5.5.2.15, ''Containment Leakage Rate Testing Program.'' Specifically, the licensee proposes a one-time extension of the ten-year period of the performance-based leakage rate testing program for Type A tests as prescribed by Nuclear Energy Institute 94–01, Revision 0, ''Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J.'' The ten-year interval between integrated leakage rate tests is to be extended to 15 years from the previous integrated leakage rate tests. Under the current TS requirements, which include an allowance of a 15-month extension, the next Type A test would be performed during the Cycle 14 refueling outages currently planned for November 2005 (Unit 2) and June 2006 (Unit 3). The requested change reflects a one-time deferral of the next Type A containment integrated leak rate test to no later than March 30, 2010 (Unit 2) and September 9, 2010 (Unit 3). This proposed change is based on and has been evaluated using the ''risk informed'' guidance in Regulatory Guide 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-informed Decisions on Plant-Specific Changes to the Licensing Basis.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed revision to Technical Specifications adds a one time extension to the current interval for Type A testing (10 CFR 50, Appendix J, Option B, Integrated Leak Rate Testing). The current test interval of 10 years, based on past performance, would be extended on a one time basis to 15 years from the last Type A test. The proposed extension to Type A testing does not involve a significant increase in the consequences of an accident since research documented in NUREG–1493, ''Performance-Based Containment System Leakage Testing Requirements,'' September 1995, has found that, generically, very few potential containment leakage paths are not identified by Type B and C tests. The NUREG concluded that reducing the Type A testing frequency to one per twenty years was found to lead to an imperceptible increase in risk. A high degree of assurance is provided through testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. The last Type A tests show leakage to be below acceptance criteria, indicating a leak tight containment. Inspections required by the American Society of Mechanical Engineers (ASME) Code Section XI (Subsections IWE and IWL) and maintenance rule monitoring (10 CFR 50.65, ''Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants) are performed in order to identify indications of containment degradation that could affect that leak tightness. Type B and C testing required by Technical Specifications will identify any containment opening such as valves that would otherwise be detected by the Type A tests. These factors show that a Type A test extension will not represent a significant increase in the consequences of an accident.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed revision to Technical Specifications adds a one time extension to the current interval for Type A testing (10 CFR 50, Appendix J, Option B, Integrated Leak Rate Testing). The current test interval of 10 years, based on past performance, would be extended on a one time basis to 15 years from the last Type A test. The proposed extension to Type A testing cannot create the possibility of a new or different type of accident since there are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure mode creating an accident or affecting the mitigation of an accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The proposed revision to Technical Specifications adds a one time extension to the current interval for Type A testing (10 CFR 50, Appendix J, Option B, Integrated Leak Rate Testing). The current test interval of 10 years, based on past performance, would be extended on a one time basis to 15 years from the last Type A test. The proposed extension to Type A testing will not significantly reduce the margin of safety. The NUREG 1493, ''Performance-Based Containment System Leakage Testing Requirements,'' September 1995, generic study of the effects of extending containment leakage testing found that a 20 year extension in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG 1493 found that, generically, the design containment leakage rate contributes about 0.1 percent to the individual risk and that the decrease in Type A testing frequency would have a minimal affect on this risk since 95% of the potential leakage paths are detected by Type C testing. Regular inspections required by the American Society of Mechanical Engineers (ASME) Code Section XI (Subsections IWE and IWL) and maintenance rule monitoring (10 CFR 50.65, ''Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants) will further reduce the risk of a containment leakage path going undetected.

Therefore the proposed change does not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

Southern Nuclear Operating Company, Inc. Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendment request: June 28, 2004.

Description of amendment request: The proposed amendments would revise existing Technical Specifications (TSs) 3.4.13, ''RCS [Reactor Coolant System] Operational Leakage,'' TS 5.59, ''Steam Generator [SG] Tube Surveillance Program,'' and TS 5.610, ''Steam Generator Tube Inspector Report.'' It would also add a new TS 3.4.17, ''Steam Generator Tube Integrity.'' These changes would facilitate the implementation of industry initiative NEI [Nuclear Energy Institute] 97–06, ''Steam Generator Program Guidelines,'' which would allow for a comprehensive, performance-based approach to managing SG performance at Farley Nuclear Plant, Units 1 and 2.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

Response: No.

The proposed change requires a Steam Generator Program that includes performance criteria that will provide reasonable assurance that the steam generator (SG) tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE.

The structural integrity performance criterion is:

''All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.''

The accident induced leakage performance criterion is:

''The primary to secondary accident induced leakage rate for all design basis accidents, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. For FNP Units 1 and 2, leakage is not to exceed 1 gpm [gallons per minute] total for all three SGs. Exceptions to the 1 gpm limit can be applied if approved by the NRC in conjunction with approved alternate repair criteria.''

The operational LEAKAGE performance criterion is:

The RCS operational primary to secondary LEAKAGE through any one SG shall be limited to 150 gpd [gallons per day].

A steam generator tube rupture (SGTR) event is one of the design basis accidents analyzed as part of the plant licensing basis. In the analysis of a SGTR event, a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in the licensing basis plus the LEAKAGE rate associated with a double-ended rupture of a single tube is assumed.

For other design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (*i.e.*, they are assumed not to rupture). For FNP Units 1 and 2, these analyses assume that primary to secondary LEAKAGE for all SGs is 1 gpm. The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

The SG performance criteria proposed in this change to the TS identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the Steam Generator Program required by the proposed change to the TS. The program, defined by NEI 97–06, Steam Generator Program Guidelines, includes a framework that incorporates a balance of prevention, inspection, evaluation, plugging, and leakage monitoring.

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the TS for operational leakage and for DOSE EQUIVALENT I–131 in primary coolant to ensure the plant is operated within its analyzed condition. The analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gpm with no more than 500 gpd in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I–131 are at the technical specification values before the accident.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current TS and enhances the requirements for SG inspections. The proposed change does not adversely impact any other previously evaluated design basis accident and is an improvement over the current TS.

Therefore, the proposed change does not affect the consequences of a SGTR accident and the probability of such an accident is reduced. In addition, the proposed changes do not affect the consequences of a MSLB, rod ejection, or a reactor coolant pump locked rotor event.

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

Response: No.

The proposed performance based requirements are an improvement over the requirements imposed by the current TS.

Implementation of the proposed Steam Generator Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the Steam Generator Program will be an enhancement of SG tube performance. Primary to secondary LEAKAGE that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

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

Response: No.

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the Steam Generator Program to manage SG tube inspection, assessment and plugging. The requirements established by the Steam Generator Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TS.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed change to the TS.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Section Chief: Stephanie M. Coffin, Acting.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of amendment request: April 26, 2004.

Description of amendment request: The proposed amendments would revise the Technical Specification Section 5.5.12, ''Primary Containment Leakage Rate Testing Program'' to reflect a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT). This change would extend the 10 year interval between ILRTs to 15 years from the previous ILRT.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed Technical Specification change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed revision to Technical Specification 5.5.12 (''Primary Containment Leakage Rate Testing Program'') involves a one-time extension to the current interval for Type A containment testing. The current test interval of ten (10) years would be extended on a one-time basis to no longer than fifteen (15) years from the last Type A test. The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner which the plant is operated or controlled. The reactor containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such the reactor containment itself and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident. Therefore, the proposed Technical Specification change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG–1493, that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. HNP [Hatch Nuclear Plant] Unit 2 ILRT test history supports this conclusion. NUREG–1493 concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk. The integrity of the reactor containment is subject to two types of failure mechanisms which can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as design change control and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the reactor containment itself combined with the containment inspections performed in accordance with ASME [American Society of Mechanical Engineers] Section XI, the Maintenance Rule and the containment coatings program serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Therefore, the proposed Technical Specification change does not involve a significant increase in the consequences of an accident previously evaluated.

2. The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed revision to the Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The reactor containment and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. The proposed Technical Specification change does not involve a physical change to the plant or the manner in which the plant is operated or controlled. Therefore, the proposed Technical Specification change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed TS change does not involve a significant reduction in a margin of safety.

The proposed revision to Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The specific requirements and conditions of the Primary Containment

Leakage Rate Testing Program, as defined in Technical Specifications, exist to ensure that the degree of reactor containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leakage rate limit specified by Technical Specifications is maintained. The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications.

HNP Unit 2 and industry experience strongly supports the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and the Coatings Program serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Additionally, the on-line containment monitoring capability that is inherent to inerted BWR containments allows for the detection of gross containment leakage that may develop during power operation. The combination of these factors ensures that the margin of safety that is inherent in plant safety analysis is maintained. Therefore, the proposed Technical Specification change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephanie M. Coffin, Acting.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of amendment request: June 22, 2004.

Description of amendment request: The proposed amendments would revise the Technical Specification (TS), Appendix A in order to change the frequency of the logic system functional test, for the 4 kV emergency busses' loss of power instrumentation, from once every 18 months to once every 24 months.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

This is a proposed change to the surveillance requirement (SR) for the logic system functional test (LSFT) of the loss of power (LOP) instrumentation for Plant Hatch Units 1 and 2 (SR 3.3.8.1.4). The LOP instrumentation functions to monitor the voltage on the 4 kV emergency busses and, if necessary, to disconnect these busses from the offsite power source and re-connect them to on-site power. This would, of course, be necessary if a bus experienced a loss of, or a degraded, voltage. This ensures an adequate response to a loss of coolant accident (LOCA) if that accident were to occur simultaneously with a loss of off-site power (LOSP). The probability of occurrence of a previously evaluated event, such as a LOCA/LOSP, will not increase since the LOP instrumentation is not being physically altered as a result of this change in such a manner which may increase the likelihood of failure. In fact, it is not being physically altered at all as a result of this submittal.

Additionally, no other safety related equipment or components designed to prevent the occurrence of a previously evaluated event are being physically altered or otherwise affected as a result of this TS change request.

The consequences of a previously evaluated event will not increase as a result of revising the surveillance frequency for the LOP instrumentation. Review of surveillance histories demonstrates adequate performance for the LOP relays in ultimately connecting the emergency power sources to the distribution bus, justifying the revision in the surveillance frequency. Therefore, the LOP instrumentation can be reasonably expected to perform its function in a LOCA/LOSP event, even with the revised frequency for the LSFT.

For the above reasons, the change in the LSFT frequency does not involve a significant increase in the probability or consequences of a previously evaluated event.

2. The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

The LOP instrumentation is not being physically altered. Furthermore, its operation and maintenance will remain within the design bases. The only proposed change is the frequency of the logic system functional test. Since no new modes of operation are being introduced, a new or different kind of accident from any previously evaluated is not created.

3. The proposed change does not involve a significant reduction in the margin of safety.

The function of the LOP instrumentation is to ensure that the emergency power distribution busses receive adequate power from either the off-site or on-site sources. The LOP relays will initiate a transfer of the emergency 4 kV busses to the on-site diesel generators on a loss of coolant accident with a concurrent loss of off-site power. The diesel logic will then sequence the cooling water pumps and other safety related equipment onto their respective emergency bus. This sequencing of loads is tested by a different surveillance requirement which is not affected by this TS change request and has already been revised to a frequency of once per 24 months. This proposed TS revision only changes the frequency of performance of the LSFT for the LOP instrumentation. A review of surveillance histories shows that these relays perform adequately in the re-connection of the emergency busses to the on-site power source. Some problems have been noted in the history review with the loss of off-site power annunciation. However, the annunciator does not affect the safety function of providing power to the distribution bus.

For the above reasons, the margin of safety is not reduced by this proposed Technical Specifications change.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephanie M. Coffin, Acting.

Tennessee Valley Authority, Docket Nos. 50–260 and 50–296, Browns Ferry Nuclear Plant (BFN), Units 2 and 3, Limestone County, Alabama

Date of amendment request: July 8, 2004 (TS–448)

Description of amendment request: The proposed amendment requests the modification of Technical Specification Section 5.5.12 ''Primary Containment Leakage Rate Testing Program'' to allow a one-time 5-year extension to the 10-year frequency of the performance-based leakage rate testing program for Type A tests. The proposed changes are submitted on a risk-informed basis as described in Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.* The risk-informed analysis supporting the proposed changes indicates that the increase in risk from extending the integrated leak rate test interval from 10 to 15 years is insignificant.

Basis for proposed no significant hazards consideration determination:

As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, ''Issuance of Amendment,'' as discussed below:

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

No. The proposed revision to TS adds a one-time extension to the current interval for Type A testing. The current test interval of 10 years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test. The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since the containment Type A testing extension is not a modification and the test extension is not of a type that could lead to equipment failure or accident initiation.

The proposed extension to Type A testing does not involve a significant increase in the consequences of an accident since research documented in NUREG–1493 has found that, generically, very few potential containment leakage paths are not identified by Type B and C tests. The NUREG concluded that reducing the Type A (ILRT) testing frequency to once per 20 years was found to lead to an imperceptible increase in risk. These generic conclusions were confirmed by a plant specific risk assessment.

Testing and the containment inspection programs in place at BFN provide a high degree of assurance that the containment will not degrade in a manner detectable only by Type A testing. The last four Type A tests show leakage to be below acceptance criteria, indicating a very leak tight containment. Type B and C testing required by TS will identify any containment opening such as valves that would otherwise be detected by the Type A tests. Inspections, including those required by the American Society of Mechanical Engineers code are also performed in order to identify indications of containment degradation that could affect that leak tightness.

Therefore, the proposed TS change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

No. The change does not create the possibility of a new or different kind of accident from any accident previously analyzed. The proposed revision to TS adds a one-time extension to the current interval for Type A testing. The current test interval of 10 years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test. The proposed extension to Type A testing cannot create the possibility of a new or different type of accident since there are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure mode creating an accident or affecting the mitigation of an accident.

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

No. BFN Units 2 and 3 are General Electric BWR/4 plants with Mark I primary containments. The Mark I primary containment consists of a drywell, which encloses the reactor vessel; reactor coolant recirculation system and branch lines of the Reactor Coolant System; a toroidal-shaped pressure suppression chamber containing a large volume of water; and a vent system connecting the drywell to the water space of the suppression chamber. The primary containment is penetrated by personnel access hatches, piping, and electrical penetrations.

The integrity of the primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests and the overall leak-tight integrity of the primary containment is verified by a Type A integrated leak rate test as required by 10 CFR 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.'' These tests are performed to verify the essentially leak-tight characteristics of the primary containment at the design basis accident pressure. The proposed change for a one-time extension of the Type A tests does not affect the method for Type A, B, or C testing, or the test acceptance criteria. In addition, based on previous Type A testing results, TVA does not expect additional degradation during the extended period between Type A tests, which would result in a significant reduction in a margin of safety.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Acting Section Chief: Michael L. Marshall, Jr.

Tennessee Valley Authority, Docket No. 50–390, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: July 8, 2004.

Description of amendment request: The proposed amendment will revise the Technical Specification (TS) to remove the term ''inter-rack'' and associated wording from Surveillance Requirements 3.8.4.6 and 3.8.4.10 for the 125 Volt (V) Direct Current (DC) Electrical Power Subsystems of the Emergency Diesel Generators (DGs).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

No. The proposed TS change eliminates an inaccurate term and associated wording, but the actual TS amendment does not result in any change to the actual surveillance field test for the associated batteries. The proposed wording will only clarify the surveillances. Prior field tests were adequate to verify proper battery connection integrity since it tested the inside (inter-tier) jumper cable connections as if they were interchangeable with inter-rack. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed TS change does not alter the configuration of the plant's 125 V DC Electrical Power Subsystems of the Emergency DGs. The change does not directly affect plant operation. The change will not result in the installation of any new equipment or system or the modification of any existing equipment or systems. No new operations procedures, conditions, or modes will be created by this proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. The battery connection continuity check for the 125 V DC Electrical Power Subsystems of the Emergency DGs will continue to be monitored by the same process as previously performed. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Acting Section Chief: Michael L. Marshall, Jr.

Notice of Issuance of Amendments to Facility Operating Licenses

During the period since publication of the last biweekly notice, the

Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

AmerGen Energy Company, LLC, Docket No. 50–219, Oyster Creek Nuclear Generating Station (OCNGS), Ocean County, New Jersey, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1 (TMI–1), Dauphin County, Pennsylvania

Date of application for amendments: March 8, 2004.

Brief description of amendment: The amendments deleted the License Condition entitled ''Long Range Planning Program'' from the OCNGS and TMI–1 operating licenses. In addition, for TMI–1, the amendment relocated a requirement (regarding surveillance of the depth of water in the spent fuel pool) from the Long Range Planning Program to the Technical Specifications.

Date of Issuance: July 13, 2004.

Effective date: These license amendments are effective as of their date of issuance, and shall be implemented within 30 days of issuance.

Amendment Nos.: 244 and 250

Facility Operating License Nos. DPR–16 and DPR–50: Amendments revised the Operating Licenses and Technical Specifications.

*Date of initial notice in **Federal Register***: April 13, 2004 (69 FR 19563 and 19564). The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated July 13, 2004.

No significant hazards consideration comments received: No.

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1 (TMI–1), Dauphin County, Pennsylvania

Date of application for amendment: August 6, 2003, as supplemented February 13 and June 16, 2004.

Brief description of amendment: The amendment revised the reactor building tendon surveillance criteria to incorporate a reference to Title 10 of the Code of Federal Regulations (10 CFR), Section 50.55a. The amendment also includes an administrative change to provide consistency between Technical Specification Definition 1.22 (MEMBERS OF THE PUBLIC) and the definition contained in 10 CFR 20.1003, and a change to correct a typographical error in a reference title.

Date of issuance: July 13, 2004.

Effective date: As of the date of issuance and shall be implemented within 30 days.

Amendment No.: 251.

Facility Operating License No. DPR–50. Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register***: December 9, 2003 (68 FR 68655) and March 16, 2004 (69 FR 12363). The February 13, 2004, supplemental letter provided clarifying information and expanded the scope of the application as originally noticed. Therefore, the original proposed no significant hazards consideration determination was changed and republished. The June 16, 2004, supplement provided clarifying information, did not expand the scope of the application and did not change the NRC staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 13, 2004.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: December 24, 2003.

Brief description of amendment: The amendment deleted requirements from the Technical Specifications (TSs) 3.7.A.7.c and 4.7.A.7.c associated with hydrogen analyzers. The associated TS Bases are also deleted.

Date of issuance: July 22, 2004.

Effective date: As of the date of issuance, and shall be implemented within 60 days.

Amendment No.: 206.

Facility Operating License No. DPR–35: The amendment revised the TSs.

Date of initial notice in ***Federal Register***: April 13, 2004 (69 FR 19568).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 22, 2004.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: August 14, 2003, as supplemented by letters dated January 22, and May 6, 2004.

Description of amendment request: This license amendment modifies Technical Specification (TS) Table 3.3.6.1–1, ''Primary Containment and Drywell Isolation Instrumentation,'' Item 1.f, to increase the analytical limit for detected temperature and the resulting TS Allowable Value related to the setpoint for the Main Steam Line Turbine Building Temperature—High system isolation function. Additionally, it authorizes the use of the GOTHIC 7.0 computer program to perform analyses of main steamline leaks in the turbine building for Perry Nuclear Power Plant to replace the currently approved COMPARE computer program for performing the analyses listed above.

Date of issuance: July 9, 2004.

Effective date: As of the date of issuance and shall be implemented within 90 days.

Amendment No.: 130.

Facility Operating License No. NPF–58: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: (69 FR 696) January 6, 2004.

The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 9, 2004.

No significant hazards consideration comments received: No.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: February 27, 2004.

Brief description of amendment: The amendment deletes Technical Specification Section 5.6.2.6, ''Post-Accident Sampling,'' requirements to maintain a Post-Accident Sampling System.

Date of issuance: July 6, 2004.

Effective date: As of the date of issuance and shall be implemented within 60 days of issuance.

Amendment No.: 213.

Facility Operating License No. DPR–72: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register***: April 13, 2004 (69 FR 19571).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 6, 2004.

No significant hazards consideration comments received: No.

Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of application for amendments: October 23, 2002, as supplemented by letters dated August 28, 2003, December 11, 2003, February 3, 2004, and March 25, 2004.

Brief description of amendments: These amendments revised Technical Specification Section 5.6, ''Design Features—Fuel Storage,'' for St. Lucie Units 1 and 2 to include the design of a new cask pit spent fuel storage rack for each unit, and increase each unit's spent fuel storage capacity by combining the cask pit rack and existing spent fuel pool storage rack capacities. The cask pit racks will be used to store spent fuel to allow refueling outage fuel offloads and nonoutage fuel shuffles and, for Unit 1, to store new fuel prior to loading it into the reactor.

Date of Issuance: July 9, 2004.

Effective Date: As of the date of issuance and shall be implemented within 60 days.

Amendment Nos.: 192 and 135.

Renewed Facility Operating License Nos. DPR–67 and NPF–16: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 28, 2003 (68 FR 4244), as corrected March 31, 2003 (68 FR 15487). The August 28, 2003, December 11, 2003, February 3, 2004, and March 25, 2004, supplements did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in an Environmental Assessment dated July 2, 2004 and in a Safety Evaluation dated July 9, 2004.

No significant hazards consideration comments received: No.

Nebraska Public Power District, Docket No. 50–298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: January 29, 2004, as supplement by letter dated April 8, 2004.

Brief description of amendment: The amendment revises Technical Specification 3.4.9 Pressure Temperature (P/T) limit curve Figures 3.4.9–1, 3.4.9–2, and 3.4.9–3.

Date of issuance: July 14, 2004.

Effective date: As of the date of issuance and shall be implemented within 60 days of issuance.

Amendment No.: 204.

Facility Operating License No. DPR–46: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 16, 2004 (69 FR 12371). The April 8, 2004, supplemental letter provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 14, 2004.

No significant hazards consideration comments received: No.

Nebraska Public Power District, Docket No. 50–298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: January 30, 2004, as supplemented by letter dated June 17, 2004.

Brief description of amendment request: The proposed amendment would revise the Cooper Nuclear Station (CNS) Technical Specifications (TSs), by adding a temporary note to allow a one-time extension of a limited number of TS Surveillance Requirements (SRs). The temporary note states that the next required performance of the SRs may be delayed until the current cycle refueling outage, but no later than February 2, 2005, and it expires upon startup from the refueling outage. With the exception of one SR, the period of additional time requested occurs during the next planned refueling outage.

Date of issuance: July 14, 2004.

Effective date: As of the date of issuance and shall be implemented within 30 days of issuance.

Amendment No.: 205.

Facility Operating License No. DPR–46: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 12, 2004 (69 FR 7023). The June 17, 2004, supplemental letter provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 14, 2004.

No significant hazards consideration comments received: No.

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of application for amendments: September 24, 2002, and its supplements dated November 21, 2003, and March 9, 2004.

Brief description of amendments: The amendments revise Technical Specification (TS) Section 3.4.11, ''Pressurizer Power Operated Relief Valves (PORVs),'' to credit the automatic actuation of the pressurizer PORVs for mitigating the plant transient of inadvertent actuation of the safety injection (SI) system. The amendments also modify the wording in Criteria A, B, and E of TS 3.4.11 to reflect the new requirement of ensuring automatic function of PORVs and adds two new surveillance requirements. The licensee withdrew the changes to TS 3.4.10, ''Pressurizer Safety Valves,'' in its letter dated March 9, 2004.

Date of issuance: July 2, 2004.

Effective date: July 2, 2004, and shall be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 1—171; Unit 2—172.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 24, 2002 (67 FR 78522)

The November 21, 2003, and March 9, 2004, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 2, 2004.

No significant hazards consideration comments received: No.

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of application for amendment: September 19, 2003.

Brief description of amendment: This amendment revised Surveillance Requirement 4.2.4.2 to specifically identify the Power Distribution Monitoring System being used in determining the Quadrant Power Tilt Ratio with one inoperable Power Range Channel.

Date of issuance: July 6, 2004.

Effective date: As of the date of issuance and shall be implemented within 30 days from the date of issuance.

Amendment No.: 168.

Renewed Facility Operating License No. NPF–12: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register***: March 30, 2004 (69 FR 16623).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 6, 2004.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendment: March 5, 2004.

Brief description of amendment: The amendment revises the reactor coolant pump flywheel inspection interval from 10 years to 20 years.

Date of issuance: July 8, 2004.

Effective date: As of the date of issuance and shall be implemented within 45 days of issuance.

Amendment Nos.: 293 and 283.

Facility Operating License No. DPR–77 and DPR–79: Amendment revises the technical specifications.

*Date of initial notice in **Federal Register***: April 13, 2004 (69 FR 19577).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 8, 2004.

No significant hazards consideration comments received: No.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: April 8, 2004.

Brief description of amendment: The amendment revises TS 5.5.7, ''Reactor Coolant Pump Flywheel Inspection Program,'' to increase the inspection interval from 10 years to 20 years.

Date of issuance: July 12, 2004.

Effective date: July 12, 2004, and shall be implemented within 90 days from the date of issuance.

Amendment No.: 163.

Facility Operating License No. NPF–30: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register***: May 11, 2004 (69 FR 26193).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 12, 2004.

No significant hazards consideration comments received: No.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: April 30, 2003, as supplemented by letters dated December 18, 2003, and April 13, 2004.

Brief description of amendment: The amendment revises several surveillance requirements (SRs) in Technical Specification (TS) 3.8.1 on alternating current sources for plant operation. The revised SRs have notes deleted or modified to allow the SRs to be performed, or partially performed, in reactor modes that previously were not allowed by the TSs. The proposed changes to SRs 3.8.4.7 and 3.8.4.8 for direct current sources were withdrawn by letter dated April 13, 2004.

Date of issuance: July 12, 2004.

Effective date: July 12, 2004, and shall be implemented within 90 days of the date of issuance including the incorporation of the changes to the TS Bases for TS 3.8.1 as described in the licensee's letters dated April 30 and December 18, 2003, and April 13, 2004.

Amendment No.: 154.

Facility Operating License No. NPF–42. The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register***: June 10, 2003 (68 FR 34673).

The December 18, 2003, and April 13, 2004, supplemental letters provided additional clarifying information, did not expand the scope of the application as noticed and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 12, 2004.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 26th day of July 2004.

For the Nuclear Regulatory Commission.

James E. Lyons,

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

[FR Doc. 04–17346 Filed 8–2–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Draft Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission has issued for public comment a proposed revision of a guide in its Regulatory Guide Series. Regulatory Guides are developed to describe and make available to the public such information as methods acceptable to the NRC for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

The draft guide is temporarily identified by its task number, DG–1124, which should be mentioned in all correspondence concerning this draft guide. Draft regulatory guide DG–1124, ''Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,'' is proposed Revision 33 of Regulatory Guide 1.84. The regulation in 10 CFR 50.55a(c), ''Reactor Coolant Pressure Boundary,'' requires, in part, that components of the reactor coolant pressure boundary must be designed, fabricated, erected, and tested in accordance with the requirements for Class 1 components of Section III, ''Rules for Construction of Nuclear Power Plant Components,'' of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code or equivalent quality standards. The ASME publishes a new edition of the B&PV Code, which includes Section III, every three years, and new addenda every year. The latest