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

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, Deborah Chan, at 301–415–7041, TDD: 301–415–2100, or by e-mail at *DLC@nrc.gov*. Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

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: February 9, 2006.

**R. Michelle Schroll,**

*Office of the Secretary.*

[FR Doc. 06–1418 Filed 2–10–06; 1:17 pm]

**BILLING CODE 7590–01–M**

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## 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 January 20, 2006, to February 2, 2006. The last biweekly notice was published on January 31, 2006 (71 FR 5078).

### 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 O1F21, 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 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 PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by email to *pdr@nrc.gov.*

*Dairyland Power Cooperative, Docket No. 50–409, La Crosse Boiling Water Reactor, Genoa, Wisconsin*

*Date of amendment request:* December 13, 2005.

*Description of amendment requests:* The La Crosse Boiling Water Reactor (LACBWR) is currently undergoing limited decommissioning and dismantlement. The proposed license amendment would revise Technical Specifications (TS) to allow waste processing components or fixtures to be handled over the Fuel Element Storage Well (FESW), limiting the weight of such items to 50 tons (the weight of the heavy load drop found acceptable in the cask drop analyses performed for the LACBWR FESW). The proposed wording changes to the TS would allow processing and shipment of Class B and Class C radioactive waste currently stored in the FESW, which will require a cask similar to the spent fuel shipping cask reflected in the current TS.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR Part 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 shipping cask, whether it is a spent fuel shipping cask or a waste shipping cask, will be handled with the same equipment, under essentially the same LACBWR crane operating procedures and precautions, and will be conservatively enveloped by previous accident evaluations that assumed a heavy load drop weighing 50 tons. Allowing the placement of typical waste processing equipment in the FESW and the handling of a waste shipping cask limited to weighing less than 50 tons over the FESW may increase the number of cask movements over the FESW slightly but will not increase the probability nor consequences of an accident previously evaluated during a given cask handling.

(2) Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated? No.

Simply changing the name of the heavy load handled over the FESW from ''spent fuel shipping cask'' to the generic term ''shipping cask,'' as long as the heavy loads are limited to the analyzed drop weight of 50 tons and their methods of handling are essentially equivalent, does not create the possibility of a new or different kind of accident from any accident previously evaluated. Other waste processing equipment will likewise be limited to the analyzed drop weight.

(3) Does the proposed change involve a significant reduction in a margin of safety? No.

Any shipping cask or other waste processing equipment to be handled over the LACBWR FESW will be conservatively enveloped by the load and conditions in the heavy load drop analysis, which assumed a drop weight of 50 tons, performed for the LACBWR FESW and, therefore, the TS change will not involve a significant reduction in a margin of safety.

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

*NRC Section Chief:* Claudia Craig.

*Dominion Energy Kewaunee, Inc. Docket No. 50–305, Kewaunee Power Station, Kewaunee County, Wisconsin*

*Date of amendment request:* January 12, 2006.

*Description of amendment request:* The proposed changes to the Technical Specifications (TSs) are necessary in order to implement the guidance for the industry initiative on NEI 97–06, ''Steam Generator [SG] Program Guidelines.''

*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, via reference to a generic analysis published in the **Federal Register** on March 2, 2005 (70 FR 10298). In addition, the licensee's January 12, 2006, application contains analysis of the issue of no significant hazards consideration associated with those changes to the TS needed to adapt the model, generic, TS ( described in NUREG–1431, Revision 3) addressed in the **Federal Register** on March 2, 2005, to the plant-specific TS applicable to Kewaunee Power Station. The analysis is presented below:

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change requires a SG Program that includes performance criteria that will provide reasonable assurance that the 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.

A SGTR [Steam Generator Tube Rupture] event is one of the design basis accidents that are analyzed as part of a plant's 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 MSLB, [Main Steam Line Break] 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). These analyses typically assume that primary to secondary LEAKAGE for all SGs is 1 gallon per minute or increases to 1 gallon per minute as a result of accident induced stresses. 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 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 SG 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, repair, and leakage monitoring. The proposed changes do not, therefore, significantly increase the probability of an accident previously evaluated.

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 [Iodine 131] in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for DOSE EQUIVALENT I–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than [500 gallons per day or 720 gallons per day] in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I–131 are at the TS 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 TSs 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 TSs.

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 an MSLB, rod ejection, or a reactor coolant pump locked rotor event, or other previously evaluated accident.

The proposed change involves rewording of certain Technical Specification sections to be consistent with NUREG–1431, Revision 3. These modifications involve no technical changes to the existing Technical Specifications. As such, these changes are administrative in nature and do not affect initiators of analyzed events or assumed mitigation of accident or transient events.

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

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The proposed performance based requirements are an improvement over the requirements imposed by the current technical specifications. Implementation of the proposed SG 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 SG 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.

The proposed change involves rewording of certain Technical Specification sections to be consistent with NUREG–1431, Revision 3. The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The changes will not impose any new or different requirements or eliminate any existing requirements from those already approved in the CLIIP.

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

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

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 isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of an 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 SG Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the SG Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TSs.

The proposed change involves rewording of certain Technical Specification sections to be consistent with NUREG–1431, Revision 3. The changes are administrative in nature and will not involve any technical changes. The changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. In addition, since these changes are administrative in nature, no question of safety is involved.

Therefore, the 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:* Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

*Acting NRC Branch Chief:* T. Kobetz.

*Entergy Nuclear Operations, Inc., Docket Nos. 50–247 and 50–286, Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 & IP3), Westchester County, New York*

*Date of amendment request:* December 27, 2005

*Description of amendment request:* The proposed amendment changes consist of:

• Adoption of Technical Specification Task Force (TSTF)–258, Revision 4; regarding changes to Section 5.0, Administrative Controls .

• Adoption of TSTF–308, Revision 1; regarding the determination of cumulative and projected dose contributions in the Radioactive Effluents Control Program (RECP).

• Revision of IP2 definition for dose equivalent 1–131 based on NUREG–1431, Revision 3.

• Revision of IP2 RECP requirements based on NUREG–1431, Revision 3.

• Revision of IP3 Explosive Gas and Storage Tank Radioactivity Monitoring Program requirements based on NUREG–1431.

*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 changes are administrative in nature and have no affect on accident scenarios previously evaluated. Affected sections include Unit Staff requirements, the Radioactive Effluent Controls Program (RECP), and High Radiation Areas. In addition, a definition is being revised for IP2. The proposed changes will result in consistent wording for the affected sections in the Indian Point 2 and Indian Point 3 Technical Specifications, based on wording used in the latest version of the Standard Technical Specifications. This will facilitate the implementation of common programs and administrative procedures for the Indian Point site. The proposed changes do not affect initiating events for accidents previously evaluated and do not affect modified plant systems or procedures used to mitigate the progression or outcome of those accident scenarios.

Therefore, 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?

*Response:* No.

The proposed changes do not involve the installation of new plant equipment or modification of existing plant equipment. No system or component setpoints are being changed and there are no changes being proposed for the way that the plant is operated. There are no new accident initiators or equipment failure modes resulting from the proposed changes. The proposed changes are administrative in nature and support the implementation of common programs and administrative procedures for the two nuclear units located at the same site.

Therefore, the proposed changes do 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 proposed changes revise a definition and the description of certain administrative control programs. There are no changes proposed to equipment operability requirements, setpoints, or limiting parameters specified in the plant Technical Specifications.

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:* Mr. John Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

*NRC Branch Chief:* Richard J. Laufer.

*Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York and Lancaster Counties, Pennsylvania*

*Date of application for amendments:* April 28, 2005.

*Description of amendment request:* The proposed changes will modify Technical Specifications (TSs) 3.3.4.2, ''End of Cycle Recirculation Pump Trip (EOC–RPT) Instrumentation''; 3.4.1,''Recirculation Loops Operating''; and 3.7.6, ''Main Turbine Bypass System'' to add a requirement for the linear heat generation rate (LHGR) limits specified in the Core Operating Limits Report (COLR).

*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?

*Response:* No.

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location.

Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences, and to ensure that the peak cladding temperature (PCT) during a postulated design basis Loss-of-Coolant Accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

LHGR limits have been established consistent with the NRC-approved GESTAR methodology to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed changes establish a requirement for LHGR limits to be modified, as specified in the COLR, such that the fuel is protected for the conditions of an inoperable EOC–RPT [end-of-cycle recirculation pump trip] instrument function, single recirculation loop operation, or an inoperable Main Turbine Bypass System and during any plant transients or anticipated operational occurrences that may occur while in these conditions. Modifying the LHGR limits for the above three (3) condition[s] does not increase the probability of an evaluated accident. The proposed change[s] [do] not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Therefore, no individual precursors of an accident are affected.

Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences, and to ensure that the PCT during a postulated design basis LOCA does not exceed the limits specified in 10 CFR 50.46. This will ensure that the fuel design safety criteria (i.e., less than 1% plastic strain of the fuel cladding and no fuel centerline melting) are met and that the core remains in a coolable geometry following a postulated design basis LOCA or any anticipated operational occurrence. Since the operability of plant systems designed to mitigate any consequences of accidents has not changed and all fuel design limits continue to be met, the consequences of an accident previously evaluated are not expected to increase.

Therefore, the proposed changes do not involve a significant 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?

*Response:* No.

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in allowable modes of operation. The proposed changes do not involve any modifications of the plant configuration or allowable modes of operation. Requiring the LHGR limits to be modified for the conditions of inoperable EOC–RPT instrument function, single recirculation loop operation, or an inoperable Main Turbine Bypass System ensures that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences and that the assumptions of the LOCA analyses are met.

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

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

*Response:* No.

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed change[s] will not adversely affect operation of plant equipment. The change[s] will not result in a change to the setpoints at which protective actions are initiated. LHGR limits for the conditions of an inoperable EOC–RPT instrument function, single recirculation loop operation, or an inoperable Main Turbine Bypass System are established to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences and that the PCT during a postulated design basis LOCA does not exceed the limits specified in 10 CFR 50.46. This will ensure that the core remains in a coolable geometry following a postulated design basis LOCA. The proposed change will ensure the appropriate level of fuel protection.

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:* Mr. Brad Fewell, Assistant General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

*NRC Branch Chief:* Darrell J. Roberts.

*FPL Energy Seabrook LLC, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire*

*Date of amendment request:* December 19, 2005.

*Description of amendment request:* The requested change will delete those parts of Technical Specification (TS) 6.8.1.2, ''Annual Reports,'' related to occupational radiation exposures and challenges to pressurizer relief and safety valves, and TS 6.8.1.5, ''Monthly Operating Reports.'' The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on June 23, 2004 (69 FR 35067). The licensee affirmed the applicability of the model NSHC determination in its application dated December 19, 2005.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration 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 eliminates the Technical Specifications (TSs) reporting requirements to provide a monthly operating report of shutdown experience and operating statistics if the equivalent data is submitted using an industry electronic database. It also eliminates the TS reporting requirement for an annual occupational radiation exposure report, which provides information beyond that specified in NRC regulations. The proposed change involves no changes to plant systems or accident analyses. As such, the change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accidents or transients. 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?

*Response:* No.

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any 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 previously evaluated.

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

*Response:* No.

This is an administrative change to reporting requirements of plant operating information and occupational radiation exposure data, and has no effect on plant equipment, operating practices or safety analyses assumptions. For these reasons, the proposed change does not involve a significant reduction in the margin of safety.

Based upon the reasoning presented above, the requested change does not involve significance hazards consideration.

*Attorney for licensee:* M.S. Ross, Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408–0420.

*NRC Branch Chief:* Darrell J. Roberts.

*Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant (MNGP), Wright County, Minnesota*

*Date of amendment request:* September 15, 2005.

*Description of amendment request:* The licensee proposed to revise the current licensing basis by incorporating a full-scope application of the Alternative Source Term (AST) methodology (see Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents of Nuclear Power Reactors,'' July 2000) in the analysis of radiological consequences for design-basis accidents. Approval of this amendment by the Nuclear Regulatory Commission (NRC) staff would result in updating various portions of the MNGP Technical Specifications to reflect the assumptions and parameters used in the AST methodology. Also, upon approval of the proposed amendment, the licensee will make conforming changes to the MNGP Updated Final Safely Analysis Report.

*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's own analysis is presented below:

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

No. The licensee's proposed application of AST methodology to the licensing basis is analytical in nature (i.e., in Chapter 14 of the MNGP Updated Final Safety Analysis Report), and does not lead to nor is it a result of modifications to plant equipment or method of operation. Since there is no change to plant equipment or method of operation, there can thus be no change in the probability of occurrence of an accident, and no change to the accident scenarios documented in the MNGP licensing basis and previously evaluated by the NRC staff. Consequently, the actual accident radiological consequences would not be any different whether or not AST methodology is used in predicting radiological consequences.

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

No. The proposed amendment does not introduce new equipment operating modes, nor does it alter existing system and component design. Accordingly, the proposed amendment to apply AST methodology does not introduce new failure modes, nor does it alter the equipment required for accident mitigation. The postulated accident scenarios previously evaluated are not changed in any way. Therefore, the proposed amendment will 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 the margin of safety?

No. The proposed amendment would approve the licensee's application of AST methodology to predict radiological consequences for various postulated accident scenarios. The AST methodology is an NRC-approved alternative for this purpose. Other than this change, which will be reviewed by the NRC staff, the licensee is proposing no other changes to other analytical models, assumptions, parameters, or acceptance criteria. Accordingly, the proposed amendment does not involve a significant reduction in the margin of safety.

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

*Attorney for licensee:* Jonathan Rogoff, Esquire, Vice President, Counsel & Secretary, Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

*NRC Acting Branch Chief:* T. Kobetz.

*Nuclear Management Company, LLC, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota*

*Date of amendment request:* November 9, 2005.

*Description of amendment request:* The proposed amendments would revise Technical Specifications (TS) for the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2, to clarify which TS Surveillance Requirements (SRs) shall be met for TS systems which include more components (installed spare components) than are required to satisfy the TS Limiting Conditions for Operation (LCO).

*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.

This license amendment proposes to revise Technical Specification Surveillance Requirements for event monitoring instrumentation, containment ventilation isolation instrumentation, cooling water system, AC sources during plant operations and nuclear instrumentation during refueling. The affected Surveillance Requirements may require all possible components in their associated Technical Specifications to meet the Surveillance Requirements even though the Technical Specifications Limiting Conditions for Operation only require some of the possible components to be operable to satisfy the Limiting Conditions for Operation. Consistent with industry guidance, the affected Surveillance Requirements were revised to include some form of ''required'' as a descriptor of the components which shall meet the Surveillance Requirements. Minor format and error corrections are also proposed for some of these Technical Specifications.

The instrumentation and systems which are the subject of the affected Technical Specifications mitigate accidents or monitor plant conditions. The instrumentation and systems are not accident initiators, thus the proposed changes do not involve a significant increase in the probability of a previously evaluated accident. With the proposed changes, the Technical Specification Limiting Conditions for Operation will continue to be met, thus the proposed changes do not involve a significant increase in the consequences of a previously evaluated accident. Therefore, these changes do not involve a significant increase in the probability or 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.

This license amendment proposes to revise Technical Specification Surveillance Requirements for event monitoring instrumentation, containment ventilation isolation instrumentation, cooling water system, AC sources during plant operations and nuclear instrumentation during refueling. The affected Surveillance Requirements may require all possible components in their associated Technical Specifications to meet the Surveillance Requirements even though the Technical Specifications Limiting Conditions for Operation only require some of the possible components to be operable to satisfy the Limiting Conditions for Operation. Consistent with industry guidance, the affected Surveillance Requirements were revised to include some form of ''required'' as a descriptor of the components which shall meet the Surveillance Requirements. Minor format and error corrections are also proposed for some of these Technical Specifications.

The proposed Technical Specification changes do not involve a change in the instrumentation or systems' operation, or the use of the instrumentation or systems. The Limiting Conditions for Operation will continue to be met and the instrumentation and systems will continue to provide their same monitoring or mitigation function. There are no new failure modes or mechanisms created through the clarifications of which components must meet the Surveillance Requirements. There are no new accident precursors generated by clarifying which components must meet the Surveillance Requirements. The minor format and error corrections do not create new failure modes or mechanisms and do not generate new accident precursors. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

*Response:* No.

This license amendment proposes to revise Technical Specification Surveillance Requirements for event monitoring instrumentation, containment ventilation isolation instrumentation, cooling water system, AC sources during plant operations and nuclear instrumentation during refueling. The affected Surveillance Requirements may require all possible components in their associated Technical Specifications to meet the Surveillance Requirements even though the Technical Specifications Limiting Conditions for Operation only require some of the possible components to be operable to satisfy the Limiting Conditions for Operation. Consistent with industry guidance, the affected Surveillance Requirements were revised to include some form of ''required'' as a descriptor of the components which shall meet the Surveillance Requirements. Minor format and error corrections are also proposed for some of these Technical Specifications.

The Technical Specification changes proposed in this License Amendment

Request are administrative, that is, they do not involve any substantive changes in plant systems, structures or components and they do not involve any changes in plant operations. Currently the affected Technical Specification Limiting Conditions for Operation do not require all possible components addressed by the Technical Specifications to be operable. This License Amendment Request clarifies that the components not required to be operable are not required to meet the Surveillance Requirements. The Limiting Conditions for Operation will continue to be met as required by the Technical Specifications. Minor format and error corrections are also proposed. Since these changes are administrative, they do not involve a significant reduction in a margin of safety.

Therefore, based on the considerations given above, 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 requests involve no significant hazards consideration.

*Attorney for licensee:* Jonathan Rogoff, Esquire, Vice President, Counsel & Secretary, Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

*NRC Acting Branch Chief:* Timothy Kobetz.

*Pacific Gas and Electric Company, Docket No. 50–275, Diablo Canyon Nuclear Power Plant, Unit No. 1, San Luis Obispo County, California*

*Date of amendment requests:* December 16, 2005.

*Description of amendment requests:* The proposed amendment would revise Technical Specification 5.6.5, ''Core Operating Limits Report (COLR),'' by adding WCAP–12945–P–A, Addendum 1–A, Revision 0, ''Method for Satisfying 10 CFR 50.46 [Section 50.46 of Title 10 of the Code of Federal Regulations] Reanalysis Requirements for Best Estimate LOCA [Loss-of-Coolant Accident] Evaluation Models,'' dated December 2004, as an approved analytical method for determining core operating limits for Unit 1. Pacific Gas and Electric is performing a plant-specific best-estimate loss-of-coolant accident analysis for Unit 2 using a methodology different than the methodology presented in Addendum 1–A to WCAP–12945–P–A. Therefore, this license amendment applies only to Unit 1.

*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 to allow the use of the abbreviated best estimate loss-of-coolant accident (LOCA) analysis methodology does not involve a physical alteration of any plant equipment or change operating practice at Unit 1 of Diablo Canyon Power Plant (DCPP). Therefore, there will be no increase in the probability of a LOCA. The consequences of a LOCA are not being increased.

The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in Unit 1. That is, it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, paragraph b. No other accident is potentially affected by this change.

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 accident from any accident previously evaluated?

*Response:* No.

The proposed change would not result in any physical alteration to any Unit 1 system, and there would not be a change in the method by which any safety related system performs its function. The parameters assumed in the analysis are within the design limits of existing plant equipment.

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

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

*Response:* No.

It has been shown that the analytic technique used in the analysis realistically describes the expected behavior of the DCPP Unit 1 reactor system during a postulated LOCA. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of LOCAs with different break sizes, different locations, and other variations in properties have been analyzed to provide assurance that the most severe postulated LOCAs were analyzed. It has been shown by the analysis that there is a high level of probability that all criteria contained in 10 CFR 50.46, paragraph b, are met.

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 requests involve no significant hazards consideration.

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

*NRC Branch Chief:* David Terao.

*PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2 (SSES 1 and 2), Luzerne County, Pennsylvania*

*Date of amendment request:* November 18, 2005.

*Description of amendment request:* The proposed amendment would change the SSES 1 and 2 Technical Specifications (TSs) to implement the Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA). Specifically, the average power range monitor (APRM) flow-biased scram and rod block trip setpoints would be revised to permit operation in the MELLLA region. The current flow-biased rod block monitor (RBM) would also be replaced by a power dependent RBM implemented through the referenced proposed upgrade to a digital power range neutron monitor system (PRNMS). The change from the flow-biased RBM to the power-dependent RBM would also require new trip setpoints. In addition, the flow-biased APRM scram and rod block trip setdown requirement would be replaced by more direct power and flow-dependent thermal limits to reduce the need for APRM gain adjustments, and to allow more direct thermal limits administration during operation other than rated conditions. Finally, the proposed amendment would change the methods used to evaluate the annulus pressurization (AP), mass blowdown, and early release resulting from the postulated recirculation suction line break (RSLB).

*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.

Proposed Change No. 1: The proposed change eliminates the Average Power Range Monitor (APRM) flow-biased scram and rod block trip setpoint setdown requirements and substitutes power and flow dependent adjustments to the Minimum Critical Power Ratio (MCPR) and Linear Heat Generation Rate (LHGR) thermal limits. Thermal limits will be determined using NRC approved analytical methods. The proposed change will have no effect upon any accident initiating mechanism. The power and flow dependent adjustments will ensure that the MCPR safety limit will not be violated as a result of any Anticipated Operational Occurrence (AOO), and that the fuel thermal and mechanical design bases will be maintained. Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Proposed Change No. 2: The proposed change expands the power and flow operating domain by relaxing the restrictions imposed by the formulation of the APRM flow-biased scram and rod block trip setpoints and the replacement of the current flow-biased RBM with a new power dependent RBM, which will be implemented using a digital Power Range Neutron Monitoring System (PRNMS). The APRM and RBM are not involved in the initiation of any accident; and the APRM flow-biased scram and rod block functions are not credited in any PPL safety licensing analyses.

The analysis of the instrument line break event resulted in an insignificant change in the radiological consequences. The change for the instrument line break was an insignificant increase of 0.1 Rem.

Since the proposed changes will not affect any accident initiator, or introduce and initial conditions that would result in NRC approved criteria being exceeded, and since the APRM and RBM will remain capable of performing their design functions, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Proposed Change No. 3: The methods used to evaluate Annulus Pressurization (AP) and mass blowdown and energy releases resulting from the postulated Recirculation Suction Line Break (RSLB) at the MELLLA conditions are changed to use more realistic, but still conservative, methods of analysis to determine an AP mass and energy release profile for AP loads resulting from the postulated RSLB. The releases resulting from the RSLB at off-rated conditions have been demonstrated to be bounded by the current design basis loads. Since the proposed changes do not affect any accident initiator and since the RSLB AP releases remain bounded by the current design basis, the proposed changes do not involve a significant increase in the probability or radiological consequences of an accident previously evaluated. Therefore the proposed changes do not involve a significant increase in the probability or consequences of any 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.

Proposed Change No. 1: The proposed change eliminates the Average Power Range Monitor (APRM) flow-biased scram and rod block setpoint setdown requirements and substitutes power and flow dependent adjustments to the Minimum Critical Power Ratio (MCPR) and Linear Heat Generation Rate (LHGR) thermal limits. Because the thermal limits will continue to be met, no analyzed transient event will escalate into a new or different type of accident due to the initial starting conditions permitted by the adjusted thermal limits. Therefore, the proposed change will not create the possibility of a new or different kind of accident previously evaluated.

Proposed Change No. 2: The proposed change expands the power and flow operating domain by relaxing the restrictions imposed by the formulation of the APRM flow-biased scram and rod block trip setpoints and the replacement of the current flow-biased RBM with a new power dependent RBM, which will be implemented using a digital Power Range Neutron Monitoring System (PRNMS). Changing the formulation for the APRM flow-biased scram and rod block trip setpoints and from a flow-biased RBM to a power dependent RBM does not change their respective functions and manner of operation. The change does not introduce a sequence of events or introduce a new failure mode that would create a new or different type of accident. The APRM flow-biased rod block trip setpoint will continue to block control rod withdrawal when core power significantly exceeds normal limits and approaches the scram level. The APRM flow-biased scram trip setpoint will continue to initiate a scram if the increasing power/flow condition continue beyond the APRM flow-biased rod block setpoint. The power dependent RBM will prevent rod withdrawal when the power dependent RBM rod block setpoint is reached. No new failure mechanisms, malfunctions, or accident initiators are being introduced by the proposed changes. In addition, operating within the expanded power flow map will not require any systems, structures or components to function differently than previously evaluated and will not create initial conditions that would result in a new or different kind of accident from any accident previously evaluated.

Proposed Change No. 3: The methods used to evaluate Annulus Pressurization (AP) and mass blowdown and energy releases resulting from the postulated Recirculation Suction Line Break (RSLB) at the MELLLA conditions are changed to use more realistic, but still conservative, methods of analysis to determine an AP mass and energy release profile for AP loads resulting from the postulated RSLB. The proposed changes to the methods of analysis to determine AP mass and energy releases resulting from the postulated RSLB do not change the design function or operation of any plant equipment. No new failure mechanisms, malfunctions, or accident initiators are being introduced by the proposed changes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

*Response:* No.

Proposed Change No. 1: The proposed change eliminates the Average Power Range Monitor (APRM) flow-biased scram and rod block setpoint setdown requirements and substitutes power and flow dependent adjustments to the Minimum Critical Power Ratio (MCPR) and Linear Heat Generation Rate (LHGR) thermal limits. Replacement of the APRM setpoint setdown requirement with power and flow dependent adjustments to the MPR and LHGR thermal limits will ensure that margins to the fuel cladding Safety Limit are preserved during operation at other than rated conditions. Thermal limits will be determined using NRC approved analytical methods. The power and flow dependent adjustments will ensure that the MPR safety limit will not be violated as a result of any Anticipated Operational Occurrence (AOO), and that the fuel thermal and mechanical design bases will be maintained. The 10 CFR 50.46 acceptance criteria for the performance of the Emergency Core Cooling System (ECCS) following postulated Loss-Of-Coolant Accidents (LOCAs) will continue to be met. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

Proposed Change No. 2: The proposed change expands the power and flow operating domain by relaxing the restrictions imposed by the formulation of the APRM flow-biased scram and rod block trip setpoints and the replacement of the current flow-biased RBM with a new power dependent RBM, which will be implemented using a digital Power Range Neutron Monitoring System (PRNMS). The APRM flow-biased rod block trip setpoint will continue to block control rod withdrawal when core power significantly exceeds normal limits and approaches the scram level. The APRM flow-biased scram trip setpoint will continue to initiate a scram if the increasing power/flow condition continues beyond the APRM flow-biased rod block setpoint. The RBM will continue to prevent rod withdrawal when the power dependent RBM rod block setpoint is reached. The MPR and LHGR thermal limits will be developed to ensure that fuel thermal mechanical design bases shall remain within the licensing limits during a rod withdrawal error event and to ensure that the MPR safety limit will not be violated as a result of a rod withdrawal error event. Operation in the expanded operating domain will not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. Anticipated operational occurrences and postulated accident within the expanded operating domain will be evaluated using NRC approved methods. Therefore, the proposed change will not involve a significant reduction in the margin of safety.

Proposed Change No. 3: The methods used to evaluate Annulus Pressurization (AP) and mass blowdown and energy releases resulting from the postulated Recirculation Suction Line Break (RSLB) at the MELLLA conditions are changed to use more realistic, but still conservative, methods of analysis to determine an AP mass and energy release profile for AP loads resulting from the postulated RSLB. Mass and energy releases for AP loads resulting from the postulated RSLB remain bounded by the current design basis releases. 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:* Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

*NRC Branch Chief:* Richard J. Laufer.

*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:* November 30, 2005.

*Description of amendment requests:* The proposed amendment would revise the Technical Specification (TS) requirements related to steam generator (SG) tube integrity, based on the NRC-approved Revision 4 to TS Task Force (TSTF)-449, ''Steam Generator Tube Integrity.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on March 2, 2005 (70 FR 10298), on possible amendments adopting TSTF–449, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 6, 2005 (70 FR 24126). The licensee affirmed the applicability of the following NSHC determination in its application dated November 30, 2005.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change requires a[n] SG Program that includes performance criteria that will provide reasonable assurance that the 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.

A[n] SGTR [SG Tube Rupture] event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. In the analysis of a[n] 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 MSLB [main steamline break], 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). These analyses typically assume that primary to secondary LEAKAGE for all SGs is 1 gallon per minute or increases to 1 gallon per minute as a result of accident induced stresses. 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 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 SG 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, repair, and leakage monitoring. The proposed changes do not, therefore, significantly increase the probability of an accident previously evaluated.

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT 1–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for DOSE EQUIVALENT 1–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than 720 gallons per day in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT 1–131 are at the TS 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 TSs 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 TSs.

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

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The proposed performance based requirements are an improvement over the requirements imposed by the current technical specifications. Implementation of the proposed SG 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 SG 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.

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

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 isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of an 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 SG Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the SG Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TSs.

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 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 Branch Chief:* David Terao.

*Southern Nuclear Operating Company, Inc., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia*

*Date of amendment request:* December 16, 2005.

*Description of amendment request:* The proposed amendment would revise the ACTIONS NOTE for TS 3.7.5, ''Auxiliary Feedwater (AFW) System,'' based on Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF–359, Revision 9, ''Increased Flexibility in Mode Restraints.''

*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 does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed change does not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational public radiation exposures. The proposed change is consistent with safety analysis assumptions and resultant consequences.

Therefore, the proposed change does not increase 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 previously evaluated?

No. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the change does not impose any new or different requirements or eliminate any existing requirements. The change does not alter assumptions made in the safety analysis. The proposed change is consistent with the safety analysis assumptions and current plant operating practice.

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 change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by this change. The proposed change will not result in plant operation in a configuration outside the design basis.

Therefore, the proposed change 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 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. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

*NRC Branch Chief:* Evangelos C. Marinos.

*Tennessee Valley Authority (TVA), Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant (SQN), Units 1 and 2, Hamilton County, Tennessee*

*Date of amendment request:* December 19, 2005 (TS–05–11).

*Description of amendment request:* The proposed amendment would revise the Technical Specifications (TSs) for consistency with the requirements of 10 CFR 50.55a(f)(4). Title 10 CFR 50.55a(f)(4) provides reference to the applicable American Society of Mechanical Engineers (ASME) code for testing pumps and valves that are classified as ASME Code Class 1, 2, and 3. The proposed change provides consistency with the 10 CFR 50.55a(f)(4) requirement by replacing the TS reference to ASME Boiler and Pressure Vessel Code, Section XI, with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) as it applies to the Inservice Test program. This change is based on TSTF–479, Revision 0, ''Changes to Reflect Revision of 10 CFR 50.55a.''

*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.

TVA's proposed change revises TS Surveillance Requirement (SR) 4.0.5 for SQN Units 1 and 2 to conform to the requirements of 10 CFR 50.55a(f) regarding inservice testing of pumps and valves for the third 10-Year interval. The current TSs reference the ASME Boiler and Pressure Vessel Code, Section XI, as the requirements for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes would replace current reference to Section XI of the Boiler and Pressure Vessel Code to the ASME OM Code, which is consistent with 10 CFR 50.55a(f) and accepted for use by the Nuclear Regulatory Commission (NRC). The proposed change incorporates updates to ASME code requirements that result in a net improvement in the measures for testing pumps and valves.

The proposed change does not involve any hardware changes, nor does it affect the probability of any event initiators. There will be no change to normal plant operating parameters, engineered safety feature actuation setpoints, accident mitigation capabilities, or accident analysis assumptions or inputs.

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?

*Response:* No.

The proposed change incorporates ASME code requirements that result in a net improvement for testing pumps and valves. The proposed change does not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure.

Equipment important to safety will continue to operate as designed. The changes to not result in any event previously deemed incredible being made credible. The changes do not result in adverse conditions or result in any increase in the challenges to safety systems.

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?

*Response:* No.

The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures of testing.

The safety function of the affected components will be maintained.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences. The proposed amendment will not otherwise affect the plant protective boundaries, will not cause a release of fission products to the public, nor will it degrade the performance of any other structures, systems, or components important to safety.

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 Branch 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:* December 13, 2005 (TS–05–06).

*Description of amendment request:* The proposed amendment would change the steam generator (SG) level requirement for Limiting Condition for Operation (LCO) 3.4.7.b and Surveillance Requirements (SRs) 3.4.5.2, 3.4.6.3 and 3.4.7.2 from greater than or equal to (≥) 6 percent to ≥ 32 percent following replacement of the SGs during the Unit 1 Cycle 7 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?

*Response:* No.

The accidents and transients of interest are those that may occur in MODE 3, 4 or 5 and that rely upon one or two of the SGs to be OPERABLE to provide a heat sink for the removal of decay heat from the reactor vessel. These events include an accidental control rod withdrawal from subcritical, ejection of a control rod, and accidental boron dilution. TS [Technical Specification] SRs provide verification of SG water level which demonstrates that the SG is OPERABLE and able to act as a heat sink.

The proposed revision to TSs 3.4.5, 3.4.6, and 3.4.7 reflects the change to the required minimum SG water level necessary to demonstrate OPERABILITY of the RSGs [Replacement SGs]. Therefore, since no initiating event mechanisms or OPERABILITY requirements are being changed, 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?

*Response:* No.

Operation in MODE 3, 4 or 5 with a SG water level of less than 32% of span is not an initiator of any of the accidents and transients described in the UFSAR [updated final safety analysis report]. This situation puts the plant into a LCO [limiting condition for operation] situation and requires that the plant initiate actions within a specified timeframe if SG OPERABILITY cannot be restored within the specified timeframe. The change in the value of the SG water level reflects the differences between the OSGs [Old Steam Generators] and the RSGs. The new value will be used in the same manner as the old one to assess the OPERABILITY of the SGs.

Therefore, operation in MODE 3, 4 or 5 with a SG water level of less than 32% of span will not initiate an accident nor create any new failure mechanisms. The changes to the TSs do not result in any event previously deemed incredible being made credible. The change will not result in more adverse conditions and is not expected to result in any increase in the challenges to safety systems.

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?

*Response:* No.

The proposed changes to the affected TSs revise the value of SG narrow range water level that is needed to demonstrate that OPERABILITY of the SG to support operation with the RSGs. The change in the value of the SG water level reflects the differences between the OSGs and the RSGs. These changes assure that the required numbers of SGs are OPERABLE with a secondary side narrow range water level indication high enough to cover the tubes. Therefore, the acceptance criterion is to provide an indicated level that will ensure the tubes are covered. Since the same acceptance criteria is being used for the RSGs as was used for the OSGs, there is no reduction in the margin of safety.

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 Branch Chief:* Michael L. Marshall, Jr.

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

*Date of amendment request:* December 15, 2005 (TS–05–09).

*Description of amendment request:* The proposed amendment would revise the Technical Specification Surveillance Requirements to increase the minimum required average ice basket weight, and thus the corresponding total weight of the stored ice in the WBN ice condenser. The changes to the ice basket and total ice weights are due to the additional energy associated with the Replacement Steam Generators.

*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 primary purpose of the ice bed is to provide a large heat sink to limit peak containment pressure in the event of a release of energy from a design basis loss-of-coolant [accident] (LOCA) or high energy line break (HELB) in containment. The LOCA requires the greatest amount of ice compared to other accident scenarios; therefore the increase in ice weight is based on the LOCA analysis. The amount of ice in the bed has no impact on the initiation of an accident, but rather on the mitigation of the accident.

The containment integrity analysis shows that the proposed increased ice weight is sufficient to maintain the peak containment pressure below the containment design pressure, and that the containment heat removal systems function to rapidly reduce the containment pressure and temperature in the event of a LOCA. 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?

*Response:* No.

The ice condenser serves to limit the peak pressure inside containment following a LOCA. The revised containment pressure analysis determined that sufficient ice would be present to maintain the peak containment pressure below the containment design pressure. The increased ice weight does not create the possibility of an accident that is different from any already evaluated in the WBN Updated Final Safety [Analysis Report]

(UFSAR). No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of 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 a margin of safety?

*Response:* No.

The containment integrity analysis for increased ice weight results in a peak containment pressure that is slightly greater than that in the previous analysis of record, but still less than design pressure. This increase in peak pressure, along with the ice weight increase, is due to an increase in RCS [reactor coolant system] inventory and stored residual heat in the replacement Steam Generators that will be installed in the Unit 1 Cycle 7 Refueling Outage.

The revised technical specification ice weight surveillance limits are based on the ice weight assumed in the containment integrity analysis, with margins included for sublimation that is based on actual sublimation data from the first six refueling cycles at WBN. The analysis further demonstrates that the existing relationship between ice bed melt-out and containment spray switchover has been conservatively maintained. With the increased ice inventory, melt-out of the ice bed following a worst case large break LOCA has been determined to occur after the switchover of containment spray to the recirculation mode. Thus, the greater ice bed mass does not result in a reduction in the margin for operator action to initiate the switchover.

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 Branch 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 (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 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 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. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois*

*Date of application for amendment:* April 1, 2005, as supplemented September 23, 2005.

*Brief description of amendment:* The amendment revised the Technical Specifications (TSs) to support the implementation of Oscillation Power Range Monitor.

*Date of issuance:* January 26, 2006.

*Effective date:* As of the date of issuance and shall be implemented within 30 days following restart from the February 2006 refueling outage.

*Amendment No.:* 171.

*Facility Operating License No. NPF–62:* The amendment revised the TSs.

*Date of initial notice in* ***Federal Register:***April 26, 2005 (70 FR 21452). The supplement dated September 23, 2005, 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 amendment is contained in a Safety Evaluation dated January 26, 2006.

*No significant hazards consideration comments received:* No.

*Calvert Cliffs Nuclear Power Plant, Inc., Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland*

*Date of application for amendments:* June 7, 2005, as supplemented on September 16, 2005.

*Brief description of amendments:* The amendments revised Technical Specification (TS) 3.1.1, ''Shutdown Margin,'' to modify the restrictions in Required Action B.1 to allow positive reactivity additions as long as the shutdown margin requirements in Limiting Condition for Operations 3.1.1 are maintained. The amendments also corrected an administrative error regarding an incorrect TS reference in TS 3.4.17, ''Special Test Exception RCS [reactor coolant system] Loops—Modes 4 and 5.''

*Date of issuance:* January 19, 2006.

*Effective date:* As of the date of issuance to be implemented within 30 days.

*Amendment Nos.:* 277 and 254.

*Renewed Facility Operating License Nos. DPR–53 and DPR–69:* Amendments revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** July 5, 2005 (70 FR 38716).

The September 16, 2005, 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 these amendments is contained in a Safety Evaluation dated January 19, 2006.

*No significant hazards consideration comments received:* No.

*Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan*

*Date of application for amendment:* March 17, 2005, as supplemented by letter dated April 15, 2005.

*Brief description of amendment:* The amendment revised Technical Specification

(TS) 3.4.10, ''RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits.'' Specifically, the amendment revised the P/T curves for the hydrostatic pressure test, non-nuclear heatup and cooldown, and nuclear (core critical) limits illustrated in TS Figure 3.4.10–1 with six recalculated separate curves for 24 and 32 effective full power years of reactor operation. In addition, the amendment revised associated surveillance requirements.

*Date of issuance:* January 25, 2006.

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

*Amendment No.:* 168.

*Facility Operating License No. NPF–43:* Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** April 26, 2005 (70 FR 21453).

The supplement dated April 15, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards determination as published in the **Federal Register** on April 26, 2005 (70 FR 21453).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 25, 2006.

*No significant hazards consideration comments received:* No.

*Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan*

*Date of application for amendment:* May 18, 2005, as supplemented by letter dated August 8, 2005.

*Brief description of amendment:* The amendment revised the Fermi 2 Technical Specifications to add Actions to limiting condition for operation [LCO] 3.8.1, ''AC [alternating current] Sources—Operating,'' for one offsite circuit inoperable, for two offsite circuits inoperable, and for one offsite circuit and one or both emergency diesel generators in one division inoperable.

*Date of issuance:* January 31, 2006.

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

*Amendment No.:* 170.

*Facility Operating License No. NPF–43:* Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** June 7, 2005 (70 FR 33212).

The supplement dated August 8, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally notice, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register** on June 7, 2005 (70 FR 33212).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 31, 2006.

*No significant hazards consideration comments received:* No.

**Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan**

*Date of application for amendment:* June 29, 2005.

*Brief description of amendment:* The amendment revised Surveillance Requirements (SR) 3.6.1.3.11 and 3.6.1.3.12 in TS 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs).'' Specifically, the proposed amendment revised the combined secondary containment bypass leakage rate limit for all bypass leakage paths in SR 3.6.1.3.11 from 0.05 to 0.10 $L_a$ (the maximum allowable containment leakage rate) and the combined main steam isolation valve (MSIV) leakage rate limit for all four main steam lines in SR 3.6.1.3.12 from 150 to 250 standard cubic feet per hour.

*Date of issuance:* January 25, 2006.

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

*Amendment No.:* 169.

*Facility Operating License No. NPF–43:* Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 16, 2005 (70 FR 48203).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 25, 2006.

*No significant hazards consideration comments received:* No.

*Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas*

*Date of application for amendment:* January 31, 2005.

*Brief description of amendment:* The amendment changed Technical Specifications (TS) 3.8.2.5, ''ELECTRICAL POWER SYSTEMS—Containment Penetration Conductor Overcurrent Protective Devices.'' The change relocated the requirements for containment penetration conductor overcurrent protective devices from the TSs to the licensee's Technical Requirements Manual (TRM). The Bases for this TS were also relocated to the TRM.

*Date of issuance:* January 23, 2006.

*Effective date:* As of the date of issuance to be implemented within 60 days from the date of issuance.

*Amendment No.:* 263.

*Renewed Facility Operating License No. NPF–6:* Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 2, 2005 (70 FR 44401).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 23, 2006.

*No significant hazards consideration comments received:* No.

*FirstEnergy Nuclear Operating Company, et al., Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio*

*Date of application for amendment:* July 27, 2005.

*Brief description of amendment:* The amendment revised Technical Specification (TS) 3/4.10.2, ''Special Test Exceptions—Physics Tests,'' to increase the allowed time between the flux channel Channel Functional Tests and the beginning of Mode 2 Physics Tests from 12 hours to 24 hours.

*Date of issuance:* January 31, 2006.

*Effective date:* As of the date of issuance and shall be implemented within 120 days.

*Amendment No.:* 271.

*Facility Operating License No. NPF–3:* Amendment revised the Technical Specifications and Surveillance Requirements.

*Date of initial notice in **Federal Register:*** September 27, 2005 (70 FR 56502).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 31, 2006.

*No significant hazards consideration comments received:* No.

*Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa*

*Date of application for amendment:* August 1, 2005, as supplemented by letters dated October 11, November 1, November 2, and November 28, 2005.

*Brief description of amendment:* The amendment conforms the license to reflect the transfer of Facility Operating License No. DPR–49 to FPL Energy Duane Arnold, LLC, as approved by order of the Commission dated December 23, 2005.

*Date of issuance:* January 27, 2006.

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

*Amendment No.:* 260.

*Facility Operating License No. DPR–49:* The amendment revised the Operating License. Date of initial notice in **Federal Register:** September 20, 2005 (70 FR 55175).

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 December 23, 2005.

*No significant hazards consideration comments received:* No.

*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 application for amendments:* August 23, 2004, as supplemented by letter dated May 20, 2005.

*Brief description of amendments:* The amendments revised the Technical Specifications Surveillance Requirements for certain containment purge valves. The amendments replace requirements for valve seat replacement every 24 months with a requirement to perform an Appendix J leakage rate test of the valves at a frequency of at least once every 30 months.

*Date of issuance:* January 20, 2006.

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

*Amendment Nos.:* 248/192.

*Renewed Facility Operating License Nos. DPR–57 and NPF–5:* Amendments revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** January 4, 2005 (70 FR 405).

The supplemental letter 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 amendments is contained in a Safety Evaluation dated January 20, 2006.

*No significant hazards consideration comments received:* No.

Dated at Rockville, Maryland, this 2nd day of February 2006.

For the Nuclear Regulatory Commission.

**Catherine Haney,**

*Director, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation.*

[FR Doc. 06–1162 Filed 2–13–06; 8:45 am]

**BILLING CODE 7590–01–P**

## SECURITIES AND EXCHANGE COMMISSION

**[File No. 1–05084]**

### Issuer Delisting; Notice of Application of Tasty Baking Company To Withdraw Its Common Stock, $.50 Par Value, and Common Stock Purchase Rights From Listing and Registration on the New York Stock Exchange, Inc.

February 7, 2006.

On October 19, 2005, Tasty Baking Company, a Pennsylvania corporation (''Issuer''), filed an application with the Securities and Exchange Commission (''Commission''), pursuant to Section 12(d) of the Securities Exchange Act of 1934 (''Act'') [1] and Rule 12d2–2(d) thereunder,[2] to withdraw its common stock, $.50 par value, and common stock purchase rights (collectively ''Securities''), from listing and registration on the New York Stock Exchange, Inc. (''NYSE'').

The Board of Directors (''Board'') of the Issuer approved resolutions on October 6, 2005 to withdraw the Securities from listing and registration on the NYSE and to list the Securities on the Nasdaq National Market (''Nasdaq''). The Board determined that it is in the best interests of the Issuer to list the Securities on Nasdaq.

The Issuer stated in its application that it has complied with NYSE's rules governing an issuer's voluntary withdrawal of a security from listing and registration by providing NYSE with the required documents governing the removal of securities from listing and registration on NYSE.

The Issuer's application relates solely to the withdrawal of the Securities from listing on the NYSE and from registration under Section 12(b) of the Act,[3] and shall not affect its obligation to be registered under Section 12(g) of the Act.[4]

Any interested person may, on or before March 6, 2006, comment on the facts bearing upon whether the application has been made in accordance with the rules of NYSE, and what terms, if any, should be imposed by the Commission for the protection of investors. All comment letters may be submitted by either of the following methods:

*Electronic Comments*

- Use the Commission's Internet comment form (*http://www.sec.gov/rules/delist.shtml*); or
- Send an e-mail to *rule-comments@sec.gov.* Please include the File Number 1–05084 or;

*Paper Comments*

- Send paper comments in triplicate to Nancy M. Morris, Secretary, Securities and Exchange Commission, 100 F Street, NE., Washington, DC 20549–1090.

All submissions should refer to File Number 1–05084. This file number should be included on the subject line if e-mail is used. To help us process and review your comments more efficiently, please use only one method. The Commission will post all comments on the Commission's Internet Web site (*http://www.sec.gov/rules/delist.shtml*). Comments are also available for public inspection and copying in the Commission's Public Reference Room. All comments received will be posted without change; we do not edit personal identifying information from submissions. You should submit only information that you wish to make available publicly.

The Commission, based on the information submitted to it, will issue an order granting the application after the date mentioned above, unless the Commission determines to order a hearing on the matter.

For the Commission, by the Division of Market Regulation, pursuant to delegated authority.[5]

**Nancy M. Morris,**

*Secretary.*

[FR Doc. E6–2012 Filed 2–13–06; 8:45 am]

**BILLING CODE 8010–01–P**

[1] 15 U.S.C. 78*l*(d).

[2] 17 CFR 240.12d2–2(d).

[3] 15 U.S.C. 78*l*(b).

[4] 15 U.S.C. 78*l*(g).

[5] 17 CFR 200.30–3(a)(1).

## SECURITIES AND EXCHANGE COMMISSION

**[Release No. 34–53234; File No. SR–Amex–2006–009]**

### Self-Regulatory Organizations; American Stock Exchange LLC; Notice of Filing and Immediate Effectiveness of a Proposed Rule Change Relating to ''All or None'' Orders

February 6, 2006.

Pursuant to section 19(b)(1) of the Securities Exchange Act of 1934