provided information to the NRC to demonstrate that the site meets the license termination criteria in Subpart E of 10 CFR Part 20 for unrestricted use.

The NRC staff has prepared an EA in support of the license amendment. The facility was remediated and surveyed prior to the licensee requesting the license amendment. The NRC staff has reviewed the information and final status survey submitted by Rohm & Haas Company. As discussed in the EA, the staff has determined that the residual radioactivity meets the requirements in Subpart E of 10 CFR Part 20.

III. Finding of No Significant Impact

The staff has prepared the EA (summarized above) in support of the license amendment to release the facility for unrestricted use. The NRC staff has evaluated Rohm & Haas Company's request and the results of the surveys and has concluded that the completed action complies with the criteria in Subpart E of 10 CFR Part 20. The staff has found that the radiological environmental impacts from the action are bounded by the impacts evaluated by NUREG–1496, Volumes 1–3, ''Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Facilities'' (ML042310492, ML042320379, and ML042330385). Additionally, no non-radiological or cumulative impacts were identified. On the basis of the EA, the NRC has concluded that the environmental impacts from the action are expected to be insignificant and has determined not to prepare an environmental impact statement for the action.

IV. Further Information

Documents related to this action, including the application for the license amendment and supporting documentation, are available electronically at the NRC's Electronic Reading Room at *http://www.nrc.gov/reading-rm/adams.html.* From this site, you can access the NRC's Agencywide Document Access and Management System (ADAMS), which provides text and image files of NRC's public documents. The ADAMS accession numbers for the documents related to this Notice are: Environmental Assessment (ML053570288); Final Status Survey and amendment request dated April 26, 2005 [ADAMS Accession No. ML051390274]; Letter dated May 16, 2005 providing additional information [ADAMS Accession No. ML051510089]; Letter dated May 27, 2005 providing additional information [ADAMS Accession No. ML051590269]; Letter dated May 31, 2005 providing additional information [ADAMS Accession No. ML051590359]; and Letter dated June 29, 2005 providing additional information [ADAMS Accession No. ML051880162]. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at (800) 397–4209 or (301) 415–4737, or by e-mail to *pdr@nrc.gov.*

Documents related to operations conducted under this license not specifically referenced in this Notice may not be electronically available and/or may not be publicly available. Persons who have an interest in reviewing these documents should submit a request to NRC under the Freedom of Information Act (FOIA). Instructions for submitting a FOIA request can be found on the NRC's Web site at *http://www.nrc.gov/reading-rm/foia/foia-privacy.html.*

Dated at King of Prussia, Pennsylvania, this 23rd day of December 2005.

For the Nuclear Regulatory Commission.

James P. Dwyer,

Chief, Commercial and Research & Development Branch, Division of Nuclear Materials Safety, Region I.

[FR Doc. E5–8205 Filed 12–30–05; 8:45 am]

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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 December 9, 2005 to December 21, 2005. The last biweekly notice was published on December 20, 2005 (70 FR 75489).

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 e-mail to *pdr@nrc.gov*.

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

Date of amendment request: September 19, 2005.

Description of amendment request: Pursuant to 10 CFR 50.90, Entergy Operations, Inc. hereby requests an Operating License amendment for Arkansas Nuclear One, Unit 2, to replace the existing steam generator (SG) tube surveillance program with that being proposed by the Technical Specifications Task Force (TSTF) in TSTF 449, Revision 4. Specifically, Technical Specification (TS) 1.1, Definitions; TS 3/4.4.5, Steam Generators; TS 3.4.6.2, Reactor Coolant System Leakage; TS 6.5.9, Steam Generator Tube Surveillance Program; and TS 6.6.7, Steam Generator Tube Surveillance Reports are being revised to incorporate the new Steam Generator Program of TSTF 449, Revision 4. The proposed changes are consistent with the Consolidated Line Item Improvement Process provided in the May 6, 2005, **Federal Register** Notice (70 FR 24126).

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

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

Response: No.

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

The structural integrity performance criterion is:

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

The accident induced leakage performance criterion is:

The primary to secondary accident induced leakage rate for any design basis accidents, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm through any one SG.

The operational leakage performance criterion is:

The RCS operational primary to secondary leakage through any one SG shall be limited to ≤150 gallons per day per SG.

A steam generator tube rupture (SGTR) 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 leakage rate associated with a double-ended rupture of a single tube is assumed.

For other design basis accidents such as main steam line break (MSLB) and control element assembly (CEA) ejection, the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). 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 identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the Steam Generator Program required by the proposed change. 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 consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the 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 720 gallons per day in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I–131 are at the technical specification values before the accident.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current technical specifications 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 technical specifications.

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 other design basis events.

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

Response: No.

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

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

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

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

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

Response: No.

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

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

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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao.

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

Date of amendment request: September 19, 2005.

Description of amendment request: Entergy Operations, Inc., proposes to amend Technical Specification (TS) 3.6.2.1, ''Containment Spray System,'' to allow a one-time extension of the allowable outage time (AOT) for the Containment Spray System (CSS) from 72 hours to a maximum of 7 days, to be used once for each train or, at most, two times during fuel cycles 18 and 19. The proposed change is intended to provide flexibility in scheduling CSS maintenance activities, reduce refueling outage duration, and improve the availability of CSS components important to safety during plant shutdowns.

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 TS change does not affect the design, operational characteristics, function or reliability of the CSS.

The CSS is primarily designed to mitigate the consequences of a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). The requested change does not affect the assumption used in the deterministic LOCA or MSLB analyses.

The duration of a TS AOT is determined considering that there is a minimal possibility that an accident will occur while a component is removed from service. A risk informed assessment was performed which concluded that the increase in plant risk is small and consistent with the guidance contained in Regulatory Guide 1.177 [''An Approach for Plant-Specific Risk-Informed Decisionmaking: Technical Specifications''].

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 previously evaluated?

Response: No.

The proposed change does not involve a change in the design, configuration, or method of operation of the plant that could create the possibility of a new or different kind of accident. The proposed change extends the AOT currently allowed by the TS to 7 days.

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 Heat Removal System (CHRS) consists of the CSS and the Containment Cooling System (CCS). The CHRS functions to rapidly reduce the containment pressure and temperature after a postulated LOCA or MSLB accident by removing thermal energy from the containment atmosphere. The CHRS also assists in limiting off-site radiation levels by reducing the pressure differential between the containment atmosphere and the outside atmosphere, thereby reducing the driving force for leakage of fission products from the containment.

The CHRS is designed so that either both trains of the CSS, or one train of CSS and one train of CCS will provide adequate heat removal to attenuate the post-accident pressure and temperature conditions imposed upon the containment following a LOCA or MSLB.

The proposed change includes administrative controls that will be established to ensure one train of CSS and one train of CCS will be available during the extended CSS AOT.

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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao.

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

Date of amendment request: October 18, 2005.

Description of amendment request: The proposed amendment would revise applicability requirements related to single control rod withdrawal allowances in shutdown modes.

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 special operation allowances do not involve the modification of any plant equipment or affect basic plant operation. The relevant design basis analyses are associated with refueling operations. The refueling interlocks are designed to back up procedural core reactivity controls during refueling operations to prevent an inadvertent criticality during refueling operations. The relaxations proposed in relocating and revising single controlrod withdrawal allowances during the Refueling MODE with the reactor vesselhead fully tensioned, to the proposed special operations allowances consistent with NUREG–1433 recommendations, will not increase the probability of an accident compared to a withdrawal of a rod while in Refueling MODE with the reactor vessel head removed. This is because the proposed special operations will allow the withdrawal of only one control rod at a time while requiring the one-rod-out interlock to be OPERABLE and other requirements imposed to ensure that all other rods remain fully inserted. This requirement coupled with the reactivity margin requirement for the most reactive rod fully withdrawn or removed, is adequate to prevent inadvertent criticality when a single rod is withdrawn for maintenance or testing. As such, there is no significant increase in the probability of an accident previously evaluated. Since no criticality is assumed to occur, the consequences of analyzed events are therefore not affected. Therefore, the proposed change does not involve a significant increase in the 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 any physical alteration of existing plant equipment or the installation of new equipment. The basic operation of installed equipment is unchanged and no new accident initiators or failure modes are introduced as a result of these changes. The methods governing plant operation and testing remain consistent with current safety analysis assumptions. These changes do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. The requirements imposed during these Special Operations ensure the existing analyses and equipment operating conditions remain bounding. 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. The margin of safety is not reduced because the proposed requirements offer similar protection to those imposed during normal refueling activities. The proposed special operation allowances do not involve the modification of any plant equipment or affect basic plant operation. The proposed allowances limit the withdrawal of only one control rod at a time. This allowance is controlled by the reactor mode switch in the refuel position, or other precautions to prevent the withdrawal or removal of more than one rod and the requirement that adequate reactivity margin be maintained. These requirements are adequate to prevent an inadvertent criticality. These changes do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: J.M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts, 02360–5599.

NRC Branch Chief: Richard Lauder.

Exelon Generation Company, LLC, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of amendment request: January 10, 2005.

Description of amendment request: The proposed change will delete the License Conditions concerning emergency core cooling system pump suction strainers from Appendix C of the Limerick Generating Station, Unit No. 1 Facility Operating License that were added by Amendment No. 128.

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 is administrative in nature. The proposed change does not involve the modification of any plant equipment nor does it affect basic plant operation. The proposed change will have no impact on any safety related structures, systems or components. The License Conditions proposed for deletion pertain to actions that have been completed and are obsolete, or involve activities that are controlled in accordance with other regulatory processes, i.e., 10 CFR 50.59 and 10 CFR 50.65.

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 is administrative in nature. The proposed change has no impact on the design, function or operation of any plant structure, system or component and does not affect any accident analyses. The License Conditions in Appendix C can be deleted because they are obsolete or involve activities that are controlled in accordance with other regulatory processes.

Therefore, the proposed change 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 a margin of safety?

Response: No. The proposed change is administrative in nature, does not negate any existing requirement, and does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there is no change being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by deletion of the License Conditions.

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

NRC Branch Chief: Darrell J. Roberts.

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

Date of amendment request: September 29, 2005.

Description of amendment request: The proposed amendment would eliminate operability requirements for Secondary Containment, Secondary Containment Isolation Valves, the Standby Gas Treatment System, and Secondary Containment Isolation Instrumentation when handling irradiated fuel that has decayed for 24 hours since critical reactor operations and when performing Core Alterations. Similar technical specification relaxations are proposed for the Control Room Emergency Filter System and its initiation instrumentation after a decay period of 7 days.

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

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

Response: No.

The proposed amendment involves implementation of the Alternative Source Term (AST) for the fuel handling accident (FHA) at Cooper Nuclear Station (CNS). There are no physical design modifications to the plant associated with the proposed amendment. The FHA AST calculation does not impact the initiators of an FHA in any way.

The changes also do not impact the initiators for any other design[-]basis accident (DBA) or events. Therefore, because DBA initiators are not being altered by adoption of the AST analyses the probability of an accident previously evaluated is not affected.

With respect to consequences, the only previously evaluated accident that could be affected is the FHA. The AST is an input to calculations used to evaluate the consequences of the accident, and does not, in and of itself, affect the plant response or the actual pathways to the environment utilized by the radiation/activity released by the fuel. It does, however, better represent the physical characteristics of the release, so that appropriate mitigation techniques may be applied. For the FHA, the AST analyses demonstrate acceptable doses that are within regulatory limits after 24 hours of radioactive decay since reactor shutdown, without credit for Secondary Containment, the Standby Gas Treatment System, Secondary Containment Isolation Valves, or Secondary Containment Isolation Instrumentation, and that the Control Room Emergency Filter System (CREFS) and CREFS Instrumentation need not be credited after a 7[-]day period of decay. Therefore, the consequences of an accident previously evaluated are not significantly increased.

Based on the above conclusions, this proposed amendment does 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.

The proposed amendment does not involve a physical alteration of the plant. No new or different types of equipment will be installed and there are no physical modifications to existing equipment associated with the proposed changes. The proposed changes to the control of Engineered Safety Features during handling of irradiated fuel do not create new initiators or precursors of a new or different kind of accident. New equipment or personnel failure modes that might initiate a new type of accident are not created as a result of the proposed amendment.

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

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

Response: No.

The proposed amendment is associated with the implementation of a new licensing basis for the CNS FHA. Approval of this change from the original source term to an AST derived in accordance with the guidance of Regulatory Guide (RG) 1.183 is being requested. The results of the FHA analysis, revised in support of the proposed license amendment, are subject to revised acceptance criteria. The AST FHA analysis has been performed using conservative methodologies, as specified in RG 1.183. Safety margins have been evaluated and analytical conservatism has been utilized to ensure that the analysis adequately bounds the postulated limiting event scenario. The dose consequences of the limiting FHA remain within the acceptance criteria presented in 10 CFR 50.67, the Standard Review Plan, and RG 1.183.

The proposed changes continue to ensure that the doses at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) boundary, as well as the Control Room, are within the corresponding regulatory limits. For the FHA, RG 1.183 conservatively sets the EAB and LPZ limits below the 10 CFR 50.67 limit, and sets the Control Room limit consistent with 10 CFR 50.67.

Since the proposed amendment continues to ensure the doses at the EAB, LPZ and Control Room are within corresponding regulatory limits, the proposed license amendment 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: Mr. John C. McClure, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Branch Chief: David Terao.

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

Date of amendment request: October 12, 2005.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 3.4.9, ''RCS [reactor coolant system] Pressure and Temperature (P/T) Limits,'' curves 3.4.9–1, ''Pressure/Temperature Limits for Non-Nuclear Heatup or Cooldown Following Nuclear Shutdown,'' 3.4.9–2, ''Pressure/Temperature Limits for Inservice Hydrostatic and Inservice Leakage Tests, and 3.4.9–3, ''Pressure/Temperature Limits for Criticality,'' to remove the cycle operating restriction and replace it with a limitation of 30 effective full-power years (EFPY).

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

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

Response: No.

The proposed revisions to the Cooper Nuclear Station (CNS) P/T curves are based on the recommendations in Regulatory Guide (RG) 1.99, Revision 2, and are, therefore, in accordance with the latest Nuclear Regulatory Commission (NRC) guidance. The fluence evaluation for the P/T curves for 30 EFPY was performed using the NRC-approved Radiation Analysis Modeling Application (RAMA) fluence methodology. The curves generated from this method provide guidance to ensure that the P/T limits will not be exceeded during any phase of reactor operation. Accordingly, the proposed revision to the CNS P/T curves is based on an NRC accepted means of ensuring protection against brittle reactor vessel fracture, and compliance with 10 CFR 50 Appendix G. The curves are the same as approved in Amendment Number 204, CNS is only requesting to remove the one cycle limitation and limit their use to 30 EFPY based on the shift in the Adjusted Reference Temperature (ART) using the new fluence values. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Based on the above, NPPD [Nebraska Public Power District] concludes that the proposed TS change to TS 3.4.9[,] P/T curves, Figures 3.4.9–1, 3.4.9–2, and 3.4.9–3 does not significantly increase 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.

The proposed change updates existing P/T operating limits to correspond to the current NRC guidance. The proposed TS change extends the use of the current, NRC-approved P/T curves beyond the end of Cycle 23 to 30 EFPY. The proposed change does not involve a physical change to the plant, add any new equipment or any new mode of operation. These TS changes demonstrate compliance with the brittle fracture requirements of 10 CFR 50 Appendix G and, therefore, do not create the possibility for a new or different kind of accident from any accident previously evaluated.

Based on the above, NPPD concludes that the proposed TS change to TS 3.4.9[,] P/T curves, Figures 3.4.9–1, 3.4.9–2, and 3.4.9–3 does 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.

The proposed change revises the existing CNS P/T curves to limit their use to 30 EFPY based on fluence calculation using the NRC-approved Radiation Analysis Modeling Application (RAMA) fluence methodology. The curves have not been recalculated. Limiting the use of the P/T curves to 30 EFPY, based on the recalculation of the fluence per the NRC-approved (RAMA) fluence methodology does not affect a margin of safety. These changes do not affect any system used to mitigate accidents or transients.

Based on the above, NPPD concludes that the proposed TS change to TS 3.4.9[,] P/T curves, Figures 3.4.9–1, 3.4.9–2, and 3.4.9–3 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. John C. McClure, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Branch Chief: David Terao.

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

Date of amendment request: September 16, 2005.

Description of amendment request: The proposed amendment would revise the surveillance requirements (SRs) for the emergency Diesel Generators (EDGs) to provide more margin to the acceptance criterion. The new SR acceptance criterion will allow the EDG frequency to be within ±2 percent of the rated value. The current acceptance limit is nominally ±1 percent of rated frequency.

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 proposed change. The EDG are not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The consequences of any accident previously evaluated are not increased, as the EDG will continue to meet their safety function, as specified in the accident analysis, in a highly reliable manner.

Therefore, the proposed change does 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.

No new or different accidents result from utilizing the proposed change. The changes do 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. The changes do not alter assumptions made in the safety analysis for the EDG performance. The proposed changes remain consistent with the safety analysis assumptions (e.g., UFSAR Section 8.3.1.4).

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 amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed change revises the acceptance criterion for EDG Surveillances to match that in the NRC's guidelines (Safety Guide 9) and the Improved Standard Technical Specifications (NUREG–1433, Rev 3). Because the EDG can perform to the specified acceptance criterion as stated in the UFSAR Section 8.3.1.4; the EDG will continue to meet their specified safety function in the safety analysis, in a highly reliable manner.

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

NRC Branch Chief: L. Raghavan.

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 21, 2005.

Description of amendment request: The proposed amendments to Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2 Operating Licenses, would allow extension of the Completion Time associated with Technical Specification (TS) 3.8.1 Required Action B4, from 7 days to 14 days and for concomitant TS changes. The proposed amendment would also allow online performance of emergency diesel generator maintenance activities that are currently performed during refueling outages, to provide additional flexibility.

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 request proposes Technical Specification changes to extend the Technical Specification 3.8.1, ''AC Sources-Operating,'' Completion Time for an inoperable emergency diesel generator to 14 days. These changes allow an emergency diesel generator to be inoperable for 7 days more than Technical Specification 3.8.1 currently provides. A minor format correction on the Technical Specification 3.8.1 Actions Table is also proposed.

The emergency diesel generators are safety related components which provide backup electrical power supply to the onsite Safeguards Distribution System. The emergency diesel generators are not accident initiators, thus allowing an emergency diesel generator to be inoperable for an additional 7 days for performance of maintenance or testing does not increase the probability of a previously evaluated accident.

Deterministic and probabilistic risk assessments evaluated the effect of the proposed Technical Specification changes on the availability of an electrical power supply to the plant emergency safeguards features systems. These assessments concluded that the proposed Technical Specification changes do not involve a significant increase in the risk of power supply unavailability.

The plant emergency safeguards features systems consist of two trains for 100% redundancy within each unit. Accident analyses demonstrate that only one emergency safeguards features train is required for accident mitigation. Thus, with one train inoperable the other train is capable of performing the required safety function. Design basis analyses are not required to be performed assuming extended loss of all power supplies to the plant emergency safeguards features systems. Thus this change does not involve a significant increase in the consequences of a previously analyzed accident.

The Technical Specification format correction is an administrative change and does not involve a significant increase in the probability or consequences of an accident.

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

2. 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 request proposes Technical Specification changes to extend the Technical Specification 3.8.1, ''AC Sources-Operating,'' Completion Time for an inoperable emergency diesel generator to 14 days. These changes allow an emergency diesel generator to be inoperable for 7 days more than Technical Specification 3.8.1 currently provides. A minor format correction on the Technical Specification 3.8.1 Actions Table is also proposed.

The proposed Technical Specification changes do not involve a change in the plant design, system operation, or procedures involved with the emergency diesel generators. The proposed changes allow an emergency diesel generator to be inoperable for additional time. There are no new failure modes or mechanisms created due to plant operation for an extended period to perform emergency diesel generator maintenance or testing. Extended operation with an inoperable emergency diesel generator does not involve any modification in the operational limits or physical design of plant systems. There are no new accident precursors generated due to the extended allowed Completion Time.

The Technical Specification format correction is an administrative change and does not create the possibility of a new or different kind of accident.

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 request proposes Technical Specification changes to extend the Technical Specification 3.8.1, ''AC Sources-Operating,'' Completion Time for an inoperable emergency diesel generator to 14 days. These changes allow an emergency diesel generator to be inoperable for 7 days more than Technical Specification 3.8.1 currently provides. A minor format correction on the Technical Specification 3.8.1 Actions Table is also proposed.

Currently, if an inoperable emergency diesel generator is not restored to operable status within 7 days, Technical Specification 3.8.1 will require unit shutdown to MODE 3 within 6 hours and MODE 5 within 36 hours. The proposed Technical Specification changes will allow steady state plant operation at 100% power for an additional 7 days.

There is some risk associated with continued operation for an additional 7 days with one emergency diesel generator inoperable. This risk is judged to be small and reasonable consistent with the risk associated with operations for 7 days with one emergency diesel generator inoperable as allowed by the current Technical Specifications. Specifically, the remaining operable emergency diesel generator and paths are adequate to supply electrical power to the onsite Safeguards Distribution System. An emergency diesel generator is required to operate only if both offsite power sources fail and there is an event which requires operation of the plant emergency safeguards features such as a design basis accident. The probability of a design basis accident occurring during this period is low.

Deterministic and probabilistic risk assessments evaluated the effect of the proposed Technical Specification changes on the availability of an electrical power supply to the plant emergency safeguards features systems. These assessments concluded that the proposed Technical Specification changes do not involve a significant increase in the risk of power supply unavailability.

There is also some risk associated with the Technical Specification unit shutdown evolutions. Plant load change evolutions require additional plant operations activities which introduce equipment challenges, increase the risk of plant trip and increase the risk for operational errors. Also unit shutdown does not remove the desirability of having emergency diesel generator backup for the 4 kV safeguards buses, but rather places dependence on the operable 4 kV bus by requiring operation of the residual heat removal system. Thus, possible additional risk associated with continuing operation an additional 7 days with an inoperable emergency diesel generator may be offset by avoiding the additional risk associated with unit shutdown.

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 Branch Chief: L. Raghavan.

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: September 30, 2005.

Description of amendment request: Omaha Public Power District (OPPD) proposes to change the licensing basis by replacing EMF–2087(P)(A), Revision 0, ''SEM/PWR–98: ECCS [Emergency Core Cooling System] Evaluation Model for PWR [pressurized-water reactor] LBLOCA [large break loss-of-coolant accident] Applications,'' Siemens Power Corporation, June 1999, with the AREVA Topical Report EMF–2103(P)(A), ''Realistic Large Break LOCA Methodology,'' Framatome ANP, Inc. in the Fort Calhoun Station, Unit 1 (FCS) Core Operating Limit Report (COLR). Currently, fuel for the FCS is supplied by AREVA. AREVA has performed an FCS-specific LBLOCA analysis using their realistic LBLOCA methodology for Cycle 24 and beyond.

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 amendment replaces EMF–2087(P)(A), Revision 0, ''SEM/PWR–98: ECCS Evaluation Model for PWR LBLOCA Applications,'' Siemens Power Corporation, June 1999 (Reference 8.6 [of the licensee's amendment request]), with the AREVA Topical Report EMF–2103(P)(A), ''Realistic Large Break LOCA Methodology,'' Framatome ANP, Inc. (Reference 8.1 [of the licensee's amendment request]) in the FCS COLR. AREVA Topical Report EMF–2103(P)(A) will also replace EMF–2087(P)(A) in OPPD topical report OPPD–NA–8303 (Reference 8.5 [of the licensee's amendment request]). This amendment will allow the use of the RLBLOCA [realistic large break loss-of-coolant accident] methodology to perform the FCS LBLOCA analysis. The proposed amendment will not affect any previously evaluated accidents because they are analyzed using applicable NRC[-]approved methodologies to ensure all required safety limits are met.

The proposed amendment does not affect any acceptance criteria for any postulated accidents or anticipated operational occurrences (AOOs) analyzed and listed in the FCS Updated Safety Analysis Report (USAR). The proposed change will not increase the likelihood of a malfunction of a structure, system or components (SSC) since the change does not involve operation of SSCs in a manner or configuration different from those previously evaluated.

The results from the FCS RLBLOCA analysis have demonstrated the adequacy of the ECCS, and these results satisfy the regulatory criteria set forth in 10 CFR 50.46(b).

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 result in changes in the operation or overall configuration of the facility. The proposed amendment does not involve a change in the design function or the operation of SSCs involved. The proposed amendment does not involve the operation or configuration of the SSCs different from those previously analyzed. The proposed amendment to add the RLBLOCA methodology to the FCS COLR and OPPD topical report OPPD–NA–8303 (Reference 8.5 [of the licensee's amendment request]) does not create any new or different kind of accident.

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.

AREVA has performed the RLBLOCA analysis for FCS and demonstrated that the Emergency Core Cooling System (ECCS) is adequate to mitigate the consequences of a[n] LBLOCA. The analysis has concluded that the acceptance criteria for the ECCS are met with significantly increased margins.

All required safety limits will continue to be analyzed using methodologies approved by the Nuclear Regulatory Commission.

Therefore, the proposed amendment 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: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Branch Chief: David Terao.

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

Date of amendment request: October 5, 2005.

Description of amendment request: The proposed amendment would delete requirements from the Technical Specifications (TSs) to maintain hydrogen recombiners and hydrogen and oxygen monitors. A notice of availability for this TS improvement using the consolidated line item improvement process was published in the **Federal Register** on September 25, 2003 (68 FR 55416).

Licensees were generally required to implement upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide (RG) 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2 in 1979. Requirements related to combustible gas control were imposed by order for many facilities and were added to, or included in, the TSs for nuclear power reactors currently licensed to operate. The revised Title 10 of the Code of Federal Regulations (10 CFR) Section 50.44, ''Combustible gas control for nuclear power reactors,'' eliminated the requirements for hydrogen recombiners and relaxed safety classifications and licensee commitments to certain design and qualification criteria for hydrogen and oxygen monitors.

The Nuclear Regulatory Commission (NRC) staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated October 5, 2005.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC 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 revised 10 CFR 50.44 no longer defines a design-basis loss-of-coolant accident (LOCA) hydrogen release, and eliminates requirements for hydrogen control systems to mitigate such a release. The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The NRC has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant accident sequences that could threaten containment integrity.

With the elimination of the design-basis LOCA hydrogen release, hydrogen and oxygen monitors are no longer required to mitigate design-basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG 1.97 Category 1, is intended for key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen and oxygen monitors no longer meet the definition of Category 1 in RG 1.97. As part of the rulemaking to revise 10 CFR 50.44, the NRC found that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of beyond design-basis accidents. Also, as part of the rulemaking to revise 10 CFR 50.44, the NRC found that Category 2, as defined in RG 1.97, is an appropriate categorization for the oxygen monitors, because the monitors are required to verify the status of the inert containment.

The regulatory requirements for the hydrogen and oxygen monitors can be relaxed without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Classification of the hydrogen monitors as Category 3, [classification of the oxygen monitors as Category 2,] and removal of the hydrogen and oxygen monitors from TSs will not prevent an accident management strategy through the use of the severe accident management guidelines, the emergency plan, the emergency operating procedures, and site survey monitoring that support modification of emergency plan protective action recommendations.

Therefore, the elimination of the hydrogen recombiner requirements and relaxation of the hydrogen and oxygen monitor requirements, including removal of these requirements from TSs, does not involve a significant increase in the probability or the consequences of any 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 elimination of the hydrogen recombiner requirements and relaxation of the hydrogen and oxygen monitor requirements, including removal of these requirements from TSs, will not result in any failure mode not previously analyzed. The hydrogen recombiner and hydrogen and oxygen monitor equipment was intended to mitigate a design-basis hydrogen release. The hydrogen recombiner and hydrogen and oxygen monitor equipment are not considered accident precursors, nor does their existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

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

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen and oxygen monitor requirements, including removal of these requirements from TSs, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety.

The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The NRC has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage.

Category 3 hydrogen monitors are adequate to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related hydrogen monitors.

Category 2 oxygen monitors are adequate to verify the status of an inserted containment.

Therefore, this change does not involve a significant reduction in the margin of safety. The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related oxygen monitors. Removal of hydrogen and oxygen monitoring from TSs will not result in a significant reduction in their functionality, reliability, and availability.

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. Lauder.

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

Date of amendment request: October 5, 2005.

Description of amendment request: The requested change will delete Technical Specification (TS) 5.6.1, ''Occupational Radiation Exposure Report,'' and TS 5.6.4, ''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 October 5, 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.

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. Lauder.

R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: November 7, 2005.

Description of amendment request: The proposed amendment would revise Technical Specification 3.9.3, ''Containment Penetrations,'' to allow an emergency egress door, access door, or roll up door, as associated with the equipment hatch penetration, to be open, but capable of being closed, during core alterations or movement of irradiated fuel within containment.

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 change has no impact on the probability of a FHA [fuel-handling accident] inside containment. It merely allows the transfer of equipment and personnel through the equipment hatch, and allows parallel activities. The refueling operations have spatial separation from the open hatch precluding interaction with refueling. Having the equipment hatch open will not impact the operation or operability of refueling equipment or the performance of the refueling crew.

Per [Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors''], the analysis was performed assuming a two hour release of radioactivity with the hatch open for the entire duration. An analysis assuming a closed hatch was not performed for comparison. This change merely allows plant conditions to exist that are assumed in the analysis. The relatively small off-site dose values shown in Section 4 [of the November 7 application], and the additional conservatism provided by the requirement for administrative closure capability, demonstrates that any consequence to the public resulting from this change would be minimal.

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 change more closely aligns the allowed plant conditions with those conditions assumed in an existing (analyzed) accident. Allowing movement of equipment through the equipment hatch during core alterations does not create any new accident initiators. Given the plant conditions, it does not affect system operation or the functions they perform. 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 change does not create conditions different from or less conservative than, those assumed in the analysis, and is consistent with the regulatory guidance for performing that analysis. 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: Daniel F. Stenger, Ballard Spahr Andrews & Ingersoll, LLP, 601 13th Street, NW., Suite 1000 South, Washington, DC 20005.

NRC Branch Chief: Richard J. Lauder.

R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: November 18, 2005.

Description of amendment request: The proposed amendment would revise the frequency in Technical Specification Surveillance Requirement (SR) 3.6.6.15, which verifies that each containment spray nozzle is unobstructed. The frequency would be changed from ''10 years'' to ''following maintenance which could result in nozzle blockage.''

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 modifies the SR to verify that the Containment Spray System nozzles are unobstructed after maintenance that could introduce material that could result in nozzle blockage. The spray nozzles are not assumed to be initiators of any previously analyzed accident. Therefore, the change does not increase the probability of any accident previously evaluated. The spray nozzles are assumed in the accident analyses to mitigate design basis accidents. The revised SR to verify system OPERABILITY following maintenance is considered adequate to ensure OPERABILITY of the Containment Spray System. Since the system will still be able to perform its accident mitigation function, the consequences of accidents previously evaluated are not increased. 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 revises the SR to verify that the Containment Spray System nozzles are unobstructed after maintenance that could result in nozzle blockage. The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators or impact the assumptions made in the safety analysis. 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 revises the frequency for performance of the SR to verify that the Containment Spray System nozzles are unobstructed. The frequency is changed from every 10 years to following maintenance that could result in nozzle blockage. This requirement, along with foreign material exclusion programs and the remote physical location of the spray nozzles, provides assurance that the spray nozzles will remain unobstructed. As the spray nozzles are expected to remain unobstructed and able to perform their post-accident mitigation function, plant safety is not significantly affected. 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: Daniel F. Stenger, Ballard Spahr Andrews & Ingersoll, LLP, 601 13th Street, NW., Suite 1000 South, Washington, DC 20005.

NRC Branch Chief: Richard J. Lauder.

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: December 6, 2005.

Description of amendment requests: The proposed amendment will delete Technical Specification (TS) Limiting Condition for Operation (LCO) 3.3.10, ''Fuel Handling Isolation Signal (FHIS),'' and TS LCO 3.7.14, ''Fuel Handling Building Post-Accident Cleanup Filter System,'' and their associated Surveillance Requirements. The proposed amendment will also delete the Fuel Handling Building Post-Accident Cleanup Filter Systems from the Ventilation Filter Testing Program in administrative TS 5.5.2.12.

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 Fuel Handling Building (FHB) Post-Accident Cleanup Filter System (PACFS) and its initiating radiation monitors are not involved in the initiation of any accidents. The PACFS is not credited with providing any supplemental filtration of releases from an accident occurring in the FHB. The PACFS was designed to provide an accident mitigation function by isolating the system and filtering the radioiodines that may be released from a damaged fuel assembly in the event of a Fuel Handling Accident (FHA). The charcoal adsorber was the primary component that supported this filtration function. However, the FHA dose consequences analysis has demonstrated that doses due to the FHA, to both the public and the control room operators, remain well within regulatory acceptance limits even assuming no credit for either isolation or filtration. The charcoal filtration function is not required and need not be tested. Thus, there is no required safety function provided by either the ventilation system or the airborne radiation monitor in the event of a fuel handling accident.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of any 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.

The FHB PACFS and its initiating radiation monitors do not initiate any accidents. The PACFS was designed to provide an accident mitigation function by isolating the system and filtering the radioiodines that may be released from a damaged fuel assembly in the event of a Fuel Handling Accident. Analysis shows that the isolation and filtration functions are not required. The charcoal adsorber cannot influence any accident initiators. The deletion of the Technical Specification requirements does not impact this conclusion and does not influence any new potential accident scenarios in any way.

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 FHB PACFS and its initiating radiation monitors were designed to provide an accident mitigation function by filtering the radioiodines that may be released from a damaged fuel assembly in the event of a Fuel Handling Accident. Analysis of the FHA in the FHB demonstrates that the margin of safety provided by the Technical Specification requirement will not change. Since the control room charcoal adsorber is capable of accommodating the design[-]basis loss[-]of[-]coolant accident fission product halogen loadings, which are more limiting than the fuel handling accident loadings, [a] more than adequate design margin is available with respect to postulated FHA releases. The margin of safety, in terms of the dose limitations of 10 CFR part 100 and 10 CFR part 50[,] Appendix A, General Design Criterion 19, has not been significantly reduced.

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

NRC Branch Chief: David Terao.

Virginia Electric and Power Company, Docket Nos. 50–280 and 50–281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia

Date of amendment request: July 21, 2005.

Description of amendment request: The proposed change would revise the accident monitoring instrumentation listing, the allowed outage times (AOTs) to be consistent with the requirements of the Improved Technical Specifications (ITS) for post accident monitoring instrumentation. TS 3.7E, TS Table 3.7–6, and TS Table 4.1–2 would be affected by this change.

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. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change revises the [AOTs] and requirements for accident monitoring instrumentation. The proposed change expands the instrumentation listing in the Technical Specifications to include the Category 1 RG [Regulatory Guide] 1.97 variables and deletes the Category 2 RG 1.97 variables, which are addressed in a licensee controlled document. The revise requirements continue to require the accident monitoring instrumentation to be operable. The required operability will continue to ensure that sufficient information is available on selected unit parameters to monitor and assess unit status and response during and following an accident. Accident monitoring instrumentation is not an initiator of any accident previously evaluated. The consequences of an accident during the extended [AOTs] would be the same as the consequences during the current [AOTs]. Therefore, the proposed change does not involve a significant increase in either the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously identified.

The proposed change involves no physical changes to the plant, nor is there any impact on the design of the plant or the accident monitoring instrumentation. There is also no impact on the capability of the instrumentation to provide post accident data for plant operator use, the accident monitoring instrumentation initiates no automatic action, and there is no change in the likelihood that the instrumentation will fail since surveillance tests will continue to be performed. Therefore, the proposed change does not introduce any new failures that could create the possibility of a new or different kind of accident from any accident previously identified.

3. Involve a significant reduction in a margin of safety.

The proposed change provides more appropriate times to restore inoperable accident monitoring instrumentation to operable status and does not impact the level of assurance that the instrumentation will be available to perform its function. Accident monitoring instrumentation has been screened out of the probabilistic risk analysis (PRA) model due to its low risk significance, so the proposed change has no risk impact from a PRA perspective. The proposed change does not alter the condition or performance of equipment or systems used in accident mitigation or assumed in any accident analysis. Therefore, this 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: Lillian M. Cuoco, Esq., Senior Counsel, Dominion Resources Services, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Branch Chief: Evangelos C. Marinos.

Virginia Electric and Power Company, Docket Nos. 50–280 and 50–281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia

Date of amendment request: September 13, 2005.

Description of amendment request: The proposed change would change the exclusion area boundary (EAB), reduce the design-basis accident (DBA) Atmospheric Dispersion Factor (X/Q), and reduce the calculated EAB dose consequences for accidents described in Chapter 14 of the Updated Final Safety Analysis Report (UFSAR).

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. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed redefinition of the EAB will significantly reduce the design basis accident X/Q, which will result in an increase in margin to the dose consequence limits for future accident analyses. The dose consequence accident analyses were not reanalyzed with this change because the EAB results currently documented in the UFSAR are conservative with respect to consequences that would be calculated using this redefined EAB. The EAB redefinition is not an initiator of any accident previously evaluated and has no impact on radiation levels, airborne activity, DBA source terms, or releases.

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

2. Create the possibility of a new or different kind of accident from any accident previously identified.

The proposed change involves no physical changes to the plant, nor is there any impact on the design or operation of the plant. There is also no impact on any equipment relied upon to mitigate an accident. Therefore, the proposed change does not introduce any new failures that could create the possibility of a new or different kind of accident from any accident previously identified.

3. Involve a significant reduction in a margin of safety.

The proposed change does not alter the condition or performance of equipment or systems used in accident mitigation or assumed in any accident analysis. The EAB redefinition has no impact on radiation levels, airborne activity, DBA source terms, or releases. Therefore, this proposed change does not involve a significant reduction in the [a] margin of safety. However, the proposed redefinition of the EAB will significantly reduce the design basis accident X/Q, which will result in an increase in margin to the dose consequence limits for future accident analyses.

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: Lillian M. Cuoco, Esq., Senior Counsel, Dominion Resources Services, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: October 27, 2005.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) 1.1, ''Definitions,'' and 3.4.16, ''RCS [reactor coolant system] Specific Activity.'' The revisions would replace the current Limiting Condition for Operation (LCO) 3.4.16 limit on RCS gross specific activity with limits on RCS Dose Equivalent I–131 and Dose Equivalent XE–133 (DEX). The conditions and required actions for LCO 3.4.16 not being met, and surveillance requirements for LCO 3.4.16, are being revised. The modes of applicability for LCO 3.4.16 would be extended. The current definition of $\bar{E}$—Average Disintegration Energy in TS 1.1 would be replaced by the definition of DEX. In addition, the current definition of Dose Equivalent I–131 in TS 1.1 would be revised to allow alternate, NRC-approved thyroid dose conversion factors.

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

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

Response: No.

The proposed changes would add new thyroid dose conversion factor reference[s] to the definition of DOSE EQUIVALENT I–131, eliminate the definition of Ē–AVERAGE DISINTEGRATION ENERGY, add a new definition of DOSE EQUIVALENT XE–133, replace the Technical Specification (TS) 3.4.16 limit on reactor coolant system (RCS) gross specific activity with a limit on noble gas specific activity in the form of a Limiting Condition for Operation (LCO) on DOSE EQUIVALENT XE–133, replace TS Figure 3.4.16–1 with a maximum limit on DOSE EQUIVALENT I–131, extend the Applicability of LCO 3.4.16, and make corresponding changes to TS 3.4.16 to reflect all of the above. The proposed changes are not accident initiators and have no impact on the probability of occurrence of any design[-]basis accidents.

The proposed changes will have no impact on the consequences of a design[-]basis accident because they will limit the RCS noble gas specific activity to be consistent with the values assumed in the radiological consequence analyses. The changes will also limit the potential RCS [radio]iodine concentration excursion to the value currently associated with full power operation, which is more restrictive on plant operation than the existing allowable RCS [radio]iodine specific activity at lower power levels.

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

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

Response: No.

The proposed changes do not alter any physical part of the plant nor do they affect any plant operating parameters besides the allowable specific activity in the RCS. The changes which impact the allowable specific activity in the RCS are consistent with the assumptions assumed in the current radiological consequence analyses. [The proposed changes are also not accident initiators.]

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

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

Response: No.

The acceptance criteria related to the proposed changes involve the allowable control room and offsite radiological consequences following a design[-]basis accident. The proposed changes will have no impact on the radiological consequences of a design[-]basis accident because they will limit the RCS noble gas specific activity to be consistent with the values assumed in the radiological consequence analyses. The changes will also limit the potential RCS [radio]iodine specific activity excursion to the value currently associated with full power operation, which is more restrictive on plant operation than the existing allowable RCS [radio]iodine specific activity at lower power levels.

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: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Branch Chief: David Terao.

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.*

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of application for amendment: October 12, 2004, as supplemented by March 4 and August 4, 2005.

Brief description of amendment: The license amendment changes the Final Safety Analysis Report (FSAR) to reflect that the reactor core isolation cooling (RCIC) system is not required to mitigate the consequences of the control rod drop accident (CRDA). The FSAR revision clarifies that although the RCIC system is designed to initiate and inject into the reactor pressure vessel (RPV) at a low water level (L2), the additional RPV inventory is not required to prevent the accident or to mitigate the consequences of the CRDA.

Date of issuance: December 14, 2005.

Effective date: This license amendment is effective as of the date of its issuance, and shall be implemented within 60 days.

Amendment No.: 196.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 9, 2004 (69 FR 64987).

The supplemental letters dated March 4 and August 4, 2005, provided 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 consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 14, 2005.

No significant hazards consideration comments received: No.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of application for amendment: August 17, 2005.

Brief description of amendment: The amendment allows a one-time extension of the 72-hour Completion Time (CT) for the required action of Condition B of Technical Specification (TS) 3.7.1, ''Standby Service Water (SW) System and Ultimate Heat Sink (UHS),'' and of TS 3.8.1, ''AC Sources—Operating.''

Specifically, the proposed one-time extension request is for an additional 72 hours to the CT and would result in a 144-hour CT for an inoperable SW subsystem. This would allow extensive maintenance, not capable of being completed in the current 72-hour CT, to be conducted on the SW train B pump.

Date of issuance: December 8, 2005.

Effective date: The license amendment is effective as of its date of issuance.

Amendment No.: 195.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 27, 2005 (70 FR 56501)

The November 15 and 30, 2005, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 8, 2005.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2, Beaver County, Pennsylvania; FirstEnergy Nuclear Operating Company, et al., Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio; FirstEnergy Nuclear Operating Company, et al., Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of application for amendments: May 18 and June 1, 2005, as supplemented by letters dated July 15 and October 31, 2005.

Brief description of amendments: The conforming amendments implement the direct license transfers of the Facility Operating Licenses for Beaver Valley Power Station, Units 1 and 2, Davis-Besse Nuclear Power Station, Unit 1, and Perry Nuclear Power Plant, Unit 1, to the extent held by Pennsylvania Power Company, Ohio Edison Company, OES Nuclear, Inc., the Cleveland Electric Illuminating Company, and the Toledo Edison Company, with respect to their current ownership interests, to FirstEnergy Nuclear Generation Corporation, a new nuclear generation subsidiary of FirstEnergy Corporation.

Date of issuance: December 16, 2005.

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

Amendment Nos. for License Nos. DPR–66 and NPF–73: 269 and 151.

Amendment Nos. for License No. NPF–3: 270.

Amendment Nos. for License No. NPF–58: 137.

Facility Operating License Nos. DPR–66, NPF–73, NPF–3, and NPF–58: Amendments revised the Licenses.

Date of initial notice in **Federal Register:** August 2, 2005 (70 FR 44390–44395).

The supplements dated July 15 and October 31, 2005 clarified the application, did not expand the scope of the application as originally noticed.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 16, 2005.

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

Date of amendment request: June 20, 2005.

Brief description of amendment: The amendment revises Cooper Nuclear Station TS 5.3, Unit Staff Qualifications, to upgrade the qualification standard for the shift manager, senior operator, licensed operator, and shift technical engineer from Regulatory Guide 1.8, ''Qualification and Training of Personnel for Nuclear Power Plants,'' Revision 2, April 1987, to Regulatory Guide 1.8, Revision 3, May 2000. It also clarifies qualification requirements applicable to the operations manager position.

Date of issuance: December 15, 2005.

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

Amendment No.: 214.

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

Date of initial notice in **Federal Register:** October 11, 2005 (70 FR 59085).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 15, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 9, 2004, as supplemented by letters dated July 9, 2004, August 17, 2004, and June 3, 2005.

Brief description of amendment: The amendment authorizes the use of the Holtec davit crane in the refueling building for cask handling operations.

Date of issuance: December 15, 2005.

Effective date: December 15, 2005, and shall be implemented within 60 days of issuance.

Amendment No.: 37.

Facility Operating License No. DPR–7: This amendment revises the licensing basis.

Date of initial notice in **Federal Register:** December 7, 2004 (69 FR 70721).

The July 9, 2004, August 17, 2004, and June 3, 2005, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff original no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 15, 2005.

No significant hazards consideration comments received: No.

R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of application for amendment: March 10, 2005, as supplemented on June 8 and August 31, 2005.

Brief description of amendment: The amendment revises Technical Specification 5.5.15, ''Containment Leakage Rate Testing Program,'' to extend, on a one-time basis, the interval for completing the next containment integrated leakage rate test, pursuant to Appendix J to Part 50 of Title 10 of the Code of Federal Regulations, from 10 years to 15 years since the last test. Therefore, the first test performed after the May 31, 1996, test shall be performed by May 31, 2011.

Date of issuance: December 8, 2005.

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

Amendment No.: 93.

Renewed Facility Operating License No. DPR–18: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 7, 2005 (70 FR 33217).

The June 8 and August 31, 2005, letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a

Safety Evaluation dated December 8, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 22, 2005.

Brief description of amendment: This amendment for Virgil C. Summer replaces the current reactor coolant system pressure-temperature limits for 32 effective full power years with the proposed limits for 56 effective full power years.

Date of issuance: December 13, 2005.

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

Amendment No.: 174.

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

Date of initial notice in **Federal Register:** September 27, 2005 (70 FR 56504).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 13, 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: May 25, 2005.

Brief description of amendments: The amendments revised the Technical Specifications to adopt the provisions of Industry/TS Task Force (TSTF) change TSTF–359, ''Increased Flexibility in Mode Restraints.''

Date of issuance: December 13, 2005.

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

Amendment Nos.: 246/190.

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

Date of initial notice in **Federal Register:** August 16, 2005 (70 FR 48207).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 13, 2005.

No significant hazards consideration comments received: No.

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 application for amendments: April 26, 2004, as supplemented by letters dated April 18 and July 22, 2005.

Brief description of amendments: The amendments revised the Units 1 and 2 Technical Specifications Limiting Condition for Operation 3.7.9, ''Ultimate Heat Sink (UHS),'' to allow plant operation with three fans and four spray cells in the Nuclear Service Cooling Water system under certain atmospheric conditions.

Date of issuance: December 2, 2005.

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

Amendment Nos.: 140 and 119.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 20, 2004 (69 FR 43462).

The supplements dated April 18 and July 22, 2005, provided clarifying information that did not change the scope of the April 26, 2004, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 2, 2005.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 23rd day of December, 2005.

For the Nuclear Regulatory Commission.

Edwin M. Hackett,

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

[FR Doc. 05–24669 Filed 12–30–05; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Sunshine Act Meeting

Notice is hereby given, pursuant to the provisions of the Government in the Sunshine Act, Public Law 94–409, that the Securities and Exchange Commission will hold the following meeting during the week of January 2, 2006:

A Closed Meeting will be held on Thursday, January 5, 2006 at 2 p.m.

Commissioners, Counsel to the Commissioners, the Secretary to the Commission, and recording secretaries will attend the Closed Meeting. Certain staff members who have an interest in the matters may also be present.

The General Counsel of the Commission, or his designee, has certified that, in his opinion, one or more of the exemptions set forth in 5 U.S.C. 552b(c)(3), (5), (7), (9)(B), and (10) and 17 CFR 200.402(a), (3), (5), (7), 9(ii) and (10) permit consideration of the scheduled matters at the Closed Meeting.

Commissioner Atkins, as duty officer, voted to consider the items listed for the closed meeting in closed session.

The subject matter of the Closed Meeting scheduled for Thursday, January 5, 2006 will be:

Formal orders of investigations;

Institution and settlement of injunctive actions;

Institution and settlement of administrative proceedings of an enforcement nature;

Regulatory matter involving a financial institution;

Amicus consideration; and an

Opinion.

At times, changes in Commission priorities require alterations in the scheduling of meeting items.

For further information and to ascertain what, if any, matters have been added, deleted or postponed, please contact: The Office of the Secretary at (202) 551–5400.

Dated: December 29, 2005.

Nancy M. Morris,

Secretary.

[FR Doc. 05–24702 Filed 12–29–05; 3:49 pm]

BILLING CODE 8010–01–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–53024; File No. SR–NASD–2005–095]

Self-Regulatory Organizations; National Association of Securities Dealers, Inc.; Notice of Filing of Proposed Rule Change and Amendment No. 2 Thereto Relating to Sub-Penny Restrictions for Non-Nasdaq Over-the-Counter Equity Securities

December 27, 2005.

Pursuant to section 19(b)(1) of the Securities Exchange Act of 1934 (''Act'') [1] and Rule 19b–4 thereunder,[2] notice is hereby given that on July 28, 2005, the National Association of

[1] 15 U.S.C. 78s(b)(1).

[2] 17 CFR 240.19b–4.