(c) NASA, Glenn Research Center at Lewis Field, Cleveland, OH 44135 (866–404–3642);

(d) NASA, Goddard Space Flight Center, Greenbelt, MD 20771 (301–286–4721);

(e) NASA, Johnson Space Center, Houston, TX 77058 (281–483–8612);

(f) NASA, Kennedy Space Center, FL 32899 (321–867–2745);

(g) NASA, Langley Research Center, Hampton, VA 23681 (757–864–2497);

(h) NASA, Marshall Space Flight Center, Huntsville, AL 35812 (256–544–1837); and

(i) NASA, Stennis Space Center, MS 39529 (228–688–2118).

NASA published a Notice of Availability (NOA) of the Draft EIS (DEIS) for the MSL mission in the **Federal Register** on September 5, 2006, (71 FR 52347) and made the DEIS available in electronic format on its Web site. The EPA published its NOA in the **Federal Register** on September 8, 2006, (71 FR 53093). In addition, NASA published its NOA in local newspapers in the Cape Canaveral, Florida regional area, and in Washington, DC, and held public meetings in Cocoa, Florida on September 27, 2006, and in Washington, DC on October 10, 2006, during which attendees were invited to present both oral and written comments on the DEIS. Three comments relevant to the DEIS were presented at these meetings. NASA received 44 written comment submissions, both hardcopy and electronic, during the comment period ending October 23, 2006. The comments are addressed in the FEIS.

Olga M. Dominguez,

Assistant Administrator for Infrastructure and Administration.

[FR Doc. E6–19610 Filed 11–20–06; 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 October 27, 2006, to November 8, 2006. The last biweekly notice was published on November 7, 2006 (71 FR 65139).

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, Rulemaking, Directives and Editing 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.*

AmerGen Energy Company, LLC, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station (Oyster Creek), Ocean County, New Jersey

Date of amendment request: September 28, 2006.

Description of amendment request: The amendment would revise the Oyster Creek Technical Specifications definition of Channel Calibration, Channel Check, and Channel Functional Test in accordance with the NUREG–1433, Revision 3, ''Standard Technical Specifications, General Electric Plants—BWR [boiling water reactor]/4.''

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. Will operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The definitions of Channel Check, Channel Calibration[,] and Channel Functional Test specified in Technical Specifications (TS) provide basic information regarding what the test involves, the components involved in the test, and general information regarding how the test is to be performed. Instrument channel checking, calibrating, and testing are not initiators of any accident previously evaluated. Furthermore, the proposed changes will not affect the ability of the channel being checked, calibrated[,] or tested to respond as assumed in any accident previously evaluated. Therefore, these revised definitions result in no increase in the probability of an accident previously evaluated.

The proposed revisions of these definitions, corresponding administrative changes (capitalization of definitions), and the proposed alternate testing and calibrating methodology using sequential, overlapping testing, and/or actual channel input signals and/or in place qualitative assessments of resistance temperature detectors (RTD's) and thermocouples (TC's) involve no changes to plant design, equipment, or operation related to mitigation of accidents. The qualitative evaluation of sensor behavior for non-adjustable sensors will provide an accurate indication of sensor operation and will assure that [the evaluated] portion of the channel is operating properly, ensuring that the consequences of an accident will remain as previously evaluated. Therefore, these revised definitions result in no increase in the consequences of an accident previously identified.

Based on the above, AmerGen concludes that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance of the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed revisions of the instrument surveillance definitions, corresponding administrative changes (capitalization of definitions), and the proposed alternate testing and calibrating methodology using sequential, overlapping testing, and/or actual channel input signals and/or in place qualitative assessments of RTD's and TC's do not involve a physical alteration of the plant or a change in the methods governing normal plant operation. No new or different type[s] of equipment will be installed. The proposed changes also do not adversely affect the operation or operability of existing plant equipment. The proposed revisions will allow a change in testing and calibrating methodology. Allowing an alternate testing and calibrating methodology will not change how the plant is operated. Each instrument channel will be tested one sub channel at a time, as is currently performed, and will not create the possibility of a new or different kind of accident.

Based on the above discussion, AmerGen concludes that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The affected definitions involve checking, calibrating[,] and testing of instrumentation used in the mitigation of accidents to ensure that the instrumentation will perform as assumed in safety analyses. The proposed revisions of these definitions, corresponding administrative changes (capitalization of definitions), and the proposed alternate testing and calibrating methodology using sequential, overlapping testing, and/or actual channel input signals and/or in place qualitative assessments of RTD's and TC's does not alter the ability of the instrument channel to respond as designed or assumed in the safety analyses. As a result[,] the ability of the plant to respond to[,] and mitigate[,] accidents is unchanged by the revised definitions. Therefore, this 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: Thomas S. O'Neill, Associate General Counsel, Exelon Generation Company, LCC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Harold K. Chernoff.

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

Date of amendment request: June 16, 2006, as supplemented by letter dated September 14, 2006.

Description of amendment request: The proposed amendment would revise the Byron Station Updated Final Safety Analysis Report (UFSAR) to incorporate changes concerning the requirements for physical protection from tornado-generated missiles (TGM) for safety-related and non-safety related systems and components.

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

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

Response: No.

The probability of occurrence of the design basis tornado remains the same as originally established in the Byron Station Updated Final Safety Analysis Report (UFSAR). The request involves the use of a probability-based assessment of the need for physical tornado missile protection of specific existing features at Byron Station.

The request is to utilize an NRC approved methodology (*i.e.*, the Electric Power Research Institute (EPRI) Topical Report ''Tornado Missile Risk Evaluation Methodology'') to conclude that the acceptance criteria of NUREG–0800, ''Standard Review Plan,'' (SRP) Section 2.2.3, ''Evaluation of Potential Accidents,'' Revision 2, July 1981, has been met for Byron Station and that tornado missile damage of selected components at Byron Station need not be considered as a credible event.

Per Item 2 in Section III of SRP 3.5.1.4, probability methods can be used to accept tornado missile effects provided damage to all important structures, systems and components, as discussed in Regulatory Guide 1.117 are considered. Per Section II of the SRP, the acceptance criterion of SRP 2.2.3 is applicable. Section II of SRP 2.2.3 states that the expected rate of occurrence of potential exposure in excess of 10 CFR Part 100, ''Reactor Site Criteria,'' guidelines of approximately 1.0E–06 per reactor year is acceptable, if when combined with reasonable qualitative arguments, that the realistic probability can be shown to be smaller.

[The licensee in its September 14, 2006, letter stated the following in regards to the consequences of an accident previously evaluated:

The acceptance criteria for the TORMIS analysis has been established as 1.0 E–06 per year cumulative probability of a TGM striking/damaging an unprotected essential SSC [system, structure or component] required for safe shutdown in the event of a tornado, which is the same value found to be acceptable by the NRC based on the accepted rates of occurrence of potential exposures in excess of 10 CFR 100 guidelines. This criteria in combination with conservative qualitative assumptions show that the realistic probability of a potential exposure in excess of the 10 CFR Part 100 guidelines is lower than 1.0 E–06 per year. The conservative qualitative assumptions are the same as previously found to be acceptable by the NRC as described below:

It is assumed that an essential SSC being struck/damaged by a tornado missile will result in damage sufficient to preclude it from performing its safety function.

It is assumed that the damage to the essential SSC results in damage to fuel sufficient to result in conservatively calculated radiological release values in excess of 10 CFR 100 guidelines.

There are no missiles that can directly impact irradiated fuel, even the spent fuel stored in the Spent Fuel Pool.]

The proposed change is not considered to constitute a significant increase in the probability or occurrence or the consequences of an accident due to the extremely low probability of damage due to tornado-generated missiles and therefore an extremely low probability of a radiological release. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of previously evaluated accidents.

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

Response: No.

This change involves the use of an alternative methodology to assess the need for tornado missile protection on selected Byron Station components. The use of this methodology and the changes to the Byron Station UFSAR will be limited to design basis tornado applications and do not contribute to the possibility of a new or different kind of accident from those previously analyzed.

No new or different system interactions are created and no new processes are introduced. The proposed change does not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases. Based on this evaluation, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The changes, allowing for no additional physical protection for tornado-generated missiles for certain Byron Sation components, is based on successfully meeting the acceptance criteria of NUREG–0800, ''Standard Review Plan,'' (SRP) Section 2.2.3, ''Evaluation of Potential Accidents,'' Revision 2, July 1981. Because of the extremely low probability of damage to these components from tornado-generated missiles, the change is not considered to constitute a significant decrease in the margin of safety.

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

Attorney for licensee: Mr. Bradley J. Fewell, Assistant General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

NRC Branch Chief: Daniel S. Collins.

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

Date of amendment request: October 13, 2006.

Description of amendment request: The proposed amendment would eliminate License Condition 2.F, which requires reporting violations of Operating License Section 2.C, and eliminates Technical Specification 5.6.6, which contains a reporting condition similar to Operating License Section 2.C.(6).

The availability of this operating license improvement was announced in the **Federal Register** on November 4, 2005 (70 FR 67202), as part of the consolidated line item improvement process (CLIIP). The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 29, 2005 (70 FR 51098), on possible amendments concerning this CLIIP, including a model safety evaluation and a model no significant hazards consideration (NSHC) determination. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on November 4, 2005 (70 FR 67202). In its application dated October 13, 2006, the licensee affirmed the applicability of the following determination.

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

Response: No.

The proposed change involves the deletion of a reporting requirement. The change does not affect plant equipment or operating practices and therefore does not significantly increase the probability or consequences of an accident previously evaluated.

2. Does the 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 that it deletes a reporting requirement. The change does not add new plant equipment, change existing plant equipment, or affect the operating practices of the facility. Therefore, the 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 proposed change deletes a reporting requirement. The change does not affect plant equipment or operating practices and therefore does not involve a significant reduction in a margin of safety.

The NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: David W. Jenkins, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Branch Chief: Daniel S. Collins.

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

Date of amendment request: October 5, 2006.

Description of amendment request: The proposed amendment to the Improved Technical Specification will revise the defined pool burnup-enrichment requirements, storage configuration for fresh fuel and low burnup/high enriched fuel, the definition of a peripheral assembly, and will include minor editorial changes.

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

The LAR proposes to revise the fresh fuel loading configuration. PEF [Progress Energy Florida, Inc.] has reanalyzed the criticality of the revised storage configuration for fresh fuel checkerboarded with spent fuel in Pool A, and surrounded by empty water cells in Pool B. Similarly, storage of spent fuel in peripheral storage locations, given the new definition, was also reanalyzed. The revised fuel storage configuration does not affect any structure, system, component or process related to the operation of Crystal River Unit 3 (CR–3). As a result, the proposed LAR will not change the probability or consequences of any accidents previously evaluated that are related to operation of the plant. Thus, only those accidents that are related to movement and storage of fuel assemblies could be potentially affected by the proposed LAR.

Fuel Handling Accidents (FHAs) are analyzed in Section 14.2.2.3 of the CR–3 Final Safety Analysis Report (FSAR). These include a FHA inside the Reactor Building (RB) and outside the RB. This LAR involves storage of fuel assemblies, an activity conducted outside the RB only. Therefore, only the FHA outside the RB event needs to be considered.

The FHA outside the RB event is described as the dropping of a fuel assembly into the spent fuel storage pool that results in damage to a fuel assembly and the release of the gaseous fission products. The current FHA assumes all 208 fuel pins in the dropped assembly are damaged and the gas gap activity released. The results of that analysis demonstrate that the applicable dose acceptance criteria, 10 CFR 50.67 and Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,'' are satisfied. Thus, the consequences of a FHA are not increased by the allowed change in the fresh fuel configuration. The fresh fuel storage configurations permit more effective use of already existing storage locations. They do not change the frequency or method for handling fuel assemblies. Fuel handling equipment is unaffected. As such, the probability of a FHA has not increased. Since only one fuel assembly is handled at a time, the consequences of a FHA have not increased.

The current limiting heat load for the spent fuel pool is from the combined impact of stored spent fuel and a full core off-load. These changes do not increase spent fuel storage capacity over that for which the racks are currently analyzed and it does not increase the amount of heat ejected from an off-loaded core. Consequently, current analyses for spent fuel pool cooling remain valid. The configuration change allows fresh fuel to be checkerboarded with spent fuel. Since these changes do not increase the storage capacity over that already analyzed for the racks, filling the empty water cells in the checkerboard pattern with spent fuel will not increase the heat load over that already analyzed. The Pool B allowance to surround a higher enriched/lower burnup fuel assembly in Pool B with empty water cells or changing the definition of a periphery rack cell does not increase the number of spent fuel assembly rack locations over that previously analyzed. Therefore, there is no increase in the pool heat load over that already analyzed.

A change in storage configurations in storage Pools A and B does not increase the probability of a full core off-load or the frequency of establishing maximum heat load conditions.

The FSAR specifies the normal upper limit of the fuel pool cooling system as 160 °F. Administrative controls are implemented to control when fuel may be moved from the reactor to the fuel pool to prevent reaching this limit.

Because neither the probability nor the consequences of a FHA are increased, and because there is not additional heat input to the spent fuel pools, it is concluded that the LAR does not involve a significant increase in 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 evaluated?

Onsite storage of spent fuel assemblies in the spent fuel pools is a normal activity for which CR–3 has been designed and licensed. As part of assuring that this normal activity can be performed without endangering public health and safety, the ability of CR–3 to safely accommodate different possible accidents in the spent fuel pools, such as dropping a fuel assembly or the misloading of a fuel assembly, have been analyzed. The revised fuel storage configurations proposed by the LAR does not change the methods of fuel movement or fuel storage. No structural or mechanical change to racks or fuel handling equipment is being proposed. The proposed revisions allow for more effective use of existing, unmodified rack locations when fresh or highly enriched, low burnup fuel is stored in the pool. The proposed revisions are a modification to the criticality analysis only, and therefore the proposed LAR does not create any new or different kind of accident from those previously evaluated.

(3) Involve a significant reduction in a margin of safety?

The CR–3 Improved Technical Specification (ITS) ensures the effective neutron multiplication factor, Keff, of the spent fuel storage racks is maintained less than or equal to 0.95 when fully loaded and flooded with unborated water. The revisions proposed by the LAR likewise ensure Keff is maintained less than this requirement.

Analyses for the proposed fuel storage configurations have shown that sufficient margin exists for fuel enriched to the maximum allowed by the CR–3 license, and for all fuel that is or has been in use at CR–3. Maintaining this margin is assured by remaining within the limits on initial enrichment and fuel burnup that are specified in the CR–3 ITS and, in the case of highly enriched, low burnup fuel in Pool B, by water hole spacing. The LAR proposes allowing fresh fuel to be checkerboarded with Category B type fuel in Pool A rather than with empty water cells. It also allows fresh fuel with high initial enrichment which does not meet current burnup requirements to be placed in Pool B if surrounded by eight empty water cells. It also proposes to change the definition of a periphery rack location for storing Category BP type fuel. Analyses show that the new proposed limits ensure that Keff remains less than 0.95. Attachment E [not included in this FR notice] provides an analysis summary.

The current CR–3 licensing basis allows the use of administrative controls, *e.g.*, curves of initial fuel assembly enrichment versus burnup, as a means of preventing criticality in the spent fuel pools. The use of these curves would be continued under this proposed amendment. The changes to these curves proposed by this LAR consist of revising the values of burnup and adding notes to restrict loading of certain fuel assemblies to specific configurations. These types of curves and administrative controls have been included in the CR–3 operating license and their use implemented by site procedures for many operating cycles. From this previous use, CR–3 personnel are familiar with the practice of using administrative controls, such as curves of fuel assembly enrichment versus burnup, to prevent criticality events when placing fuel assemblies in the spent fuel pool.

Misloaded and mislocated fuel assemblies were analyzed. The analysis demonstrated that misloading of a fresh fuel assembly, assuming no soluble poison (boron) in the water does result in exceeding the criticality margin regulatory limit of Keff = 0.95. The analysis further shows that a concentration of 165 ppm boron in the Pool A and a concentration of 46 ppm boron in Pool B is sufficient to ensure Keff < 0.95. LCO 3.7.14 currently requires a minimum boron concentration of 1925 ppm in the spent fuel pools until fuel is verified as having been loaded in accordance with the enrichment and burnup requirements of LCO 3.7.15. The soluble boron assumed in the analysis for this proposed change is substantially less than the 1925 ppm required by the existing license. Therefore, existing license requirements for soluble boron remain conservative.

The NRC staff has reviewed the analysis provided for Florida Power Corporation 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: David T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Branch Chief (Acting): L. Raghavan.

FPL Energy Duane Arnold, LLC, Docket No. 50–331, Duane Arnold Energy Center (DAEC), Linn County, Iowa

Date of amendment request: July 17, 2006.

Description of amendment request: The proposed amendment would revise the Limiting Condition for Operation (LCO) 3.6.3.1 to eliminate the requirement for the Containment Atmospheric Dilution (CAD) system, allowing its removal from the DAEC. LCO 3.6.3.2 would also be revised to allow an additional 48 hours on plant start-up or shutdown sequences for the primary containment to be de-inerted.

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 Containment Atmosphere Dilution (CAD) system and primary containment oxygen concentration are not initiators to any accident previously evaluated in the DAEC Updated Final Safety Analysis Report (UFSAR). The CAD system and containment oxygen concentration were previously relied upon to mitigate the consequences of a design basis accident (DBA) combustible gas mixture. However, the revised 10 CFR 50.44 (68 FR 54123) no longer defines a DBA hydrogen release (*i.e.*, combustible gas mixture) and the Commission has subsequently found that the DBA loss of coolant accident (LOCA) hydrogen release is not risk significant. In addition, hydrogen control systems, such as CAD, have been determined to be ineffective at mitigating hydrogen releases from the more risk significant beyond design basis accidents that could threaten containment integrity. Therefore, elimination of the CAD system will not significantly increase the consequences of any accident previously evaluated. The consequences of an accident while relying on the revised Required Actions for primary containment oxygen concentration are no different than the consequences of the same accidents under the current Required Actions. As a result, the consequences of any accident previously evaluated are not significantly 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 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, except for the elimination of the CAD system (*i.e.*, no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The CAD system is not considered an accident precursor, nor does its 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 from any DBA. In addition, the changes do not impose any new or different requirements. The changes to the Technical Specifications for oxygen concentration do not alter assumptions made in the safety analysis, but reflect changes to the safety analysis requirements allowed under the revised 10 CFR 50.44. Specifically that an inerted containment is no[t] required to mitigate any DBA, but has been found to be helpful in mitigating certain beyond design basis events (*i.e.*, severe accidents) that could generate combustible levels of hydrogen.

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 installation of combustible gas control systems, such as CAD, required by the original § 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 Commission 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. (68 FR 54123). The proposed changes to CAD and primary containment oxygen concentration reflect this new regulatory position and, in light of the remaining plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, including postulated beyond design basis events, does not result in a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mr. R. E. Helfrich, Florida Power & Light Company, P. O. Box 14000, Juno Beach, FL 33408–0420.

NRC Branch Chief: L. Raghavan.

Indiana Michigan Power Company (I&M), Docket No. 50–316, Donald C. Cook Nuclear Plant, Unit 2, Berrien County, Michigan

Date of amendment request: September 15, 2006.

Description of amendment request: The proposed amendment would replace the current control system and it will increase the nominal control fluid oil operating pressure from 114 pounds per square inch gauge (psig) to 1600 psig. The control fluid oil pressure provides an input to the reactor protection system via three pressure switches connected to the control fluid header. Due to the change in the operating pressure, I&M is proposing a revision to the allowable low fluid oil pressure value from greater than or equal to 57 psig to greater than or equal to 750 psig.

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 reflects a design change to the turbine control system that increases the control oil pressure, necessitating a change to the value at which a low fluid oil pressure initiates a reactor trip. The turbine control oil pressure is an input to the reactor trip instrumentation, and the reactor trip is a response to an event that trips the turbine. A change in the nominal control oil pressure does not introduce any mechanisms that would increase the probability of an accident previously analyzed. The reactor trip on turbine trip function is an anticipatory trip, and the safety analysis does not credit this trip for protecting the reactor core. Thus, the consequences of previously analyzed accidents are not impacted.

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.

The control fluid oil pressure decreases in response to a turbine trip. The value at which the low control fluid oil initiates a reactor trip is not an accident initiator. The change in the value reflects the higher pressure of the turbine control system that will be installed during the Unit 2 Cycle 17 refueling outage.

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 change involves a parameter that initiates an anticipatory reactor trip following a turbine trip. The safety analyses do not credit this anticipatory trip for reactor core protection.

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

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

Attorney for licensee: James M. Petro, Jr., Esquire, One Cook Place, Bridgman, MI 49106.

NRC Acting Branch Chief: Martin C. Murphy.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Unit 1 and 2, Berrien County, Michigan

Date of amendment request: September 15, 2006.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TS) to change Required Action Notes in TS 3.3.1, ''Reactor Trip System Instrumentation,'' and TS 3.3.2, ''Engineered Safety Features Actuation System Instrumentation,'' to reflect installed bypass test capability, as well as correct one administrative error in TS 3.3.1 Condition Q. The proposed changes to the Required Action Notes are consistent with wording in Standard Technical Specifications (NUREG–1431, Revision 3) for plants with installed bypass test capability.

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 of occurrence or consequences of an accident previously evaluated?

Response: No.

The proposed change reflects NUREG–1431, Revision 3, ''Standard Technical Specifications, Westinghouse Plants,'' (STS) wording for plants with installed bypass test capability and aligns Technical Specification (TS) Condition entry requirements with other portions of the TS. The proposed changes do not modify how the reactor trip system (RTS) and engineered safety features actuation systems (ESFAS) functions respond to an accident condition. The proposed changes to the TS Required Action Notes prevent unnecessary TS Action entry during performance of surveillance testing. The probability of accidents previously evaluated remains unchanged since the proposed change does not affect any accident initiators. The consequences of accidents previously evaluated are unaffected by this change because no change to any accident mitigation scenario has resulted and there are no additional challenges to fission product barrier integrity.

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.

No changes are being made to the plant that would introduce any new accident causal mechanisms. The proposed change to the Required Action Notes and Condition entry requirements does not adversely affect previously identified accident initiators and does not create any new accident initiators. The change does not affect how the RTS and ESFAS functions operate. No new single failure or accident scenarios are created by the proposed change and the proposed change does not result in any event previously deemed incredible being made credible.

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.

No safety analyses were changed or modified as a result of the proposed TS changes to reflect STS wording for plants with installed bypass test capability or for aligning TS Condition entry requirements. All margins associated with the current safety analyses acceptance criteria are unaffected. The current safety analyses remain bounding. The safety systems credited in the safety analyses will continue to be available to perform their mitigation functions. The proposed change does not affect the availability or operability of safety-related systems and components.

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

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

Attorney for licensee: James M. Petro, Jr., Esquire, One Cook Place, Bridgman, MI 49106

NRC Acting Branch Chief: M. Murphy.

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: August 14, 2006.

Description of amendment request: The proposed amendments would make miscellaneous improvements to the Technical Specifications (TS) for Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2. The proposed amendments would revise TS 1.3, ''Completion Times''; TS 3.1.4, ''Rod Group Alignment Limits''; TS 3.3.7, ''Spent Fuel Pool Special Ventilation System (SFPSVS) Actuation Instrumentation''; TS 3.7.10, ''Control Room Special Ventilation System (CRSVS)''; and TS Chapter 4.0, ''Design Features''.

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 changes to the Prairie Island Nuclear Generating Plant Technical Specifications as follows: Technical Specification 1.3, ''Completion Times'', revise a text header and add a new text header; Technical Specification 3.1.4, ''Rod Group Alignment Limits'', remove a Surveillance Note which cross-references another Technical Specification and may cause confusion; Technical Specification 3.3.7, ''Spent Fuel Pool Special Ventilation System (SFPSVS) Actuation Instrumentation'', revises the Modes of Applicability consistent with plant design and the Technical Specifications for the Spent Fuel Pool Special Ventilation System, the supported system; Technical Specification 3.7.10, ''Control Room Special Ventilation System (CRSVS)'', revises the applicability of Condition C and clarifies the requirements of the Surveillance to verify train filtration flow; and Technical Specification Chapter 4.0, ''Design Features'', revises Reference 1 to the most recent version of the document.

Revising and adding text headers in Technical Specification 1.3 are administrative changes because the revised document does not change any basis for the current Technical Specifications. Since these are administrative changes, they do not involve a significant increase in the probability or consequences of a previously evaluated accident. Technical Specification 3.1.4 assures that the control rod positions are within the limits assumed in the safety analysis and that the assumed shutdown margin is available when needed. This license amendment request proposes to remove a Note from a surveillance requirement that cross-references to Technical Specification 3.1.7. Removal of this Note does not change plant operations, testing or maintenance; therefore the proposed change does not involve a significant increase in the probability of an accident. Since plant operations, testing and maintenance are not changed, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

The Spent Fuel Pool Special Ventilation System filters radioactive materials in the fuel pool enclosure atmosphere released following a fuel handling accident. This license amendment request proposes to revise the Modes and Other Specified Conditions of Applicability for the actuation instrumentation.

Technical Specification to be consistent with the Modes and Other Specified Conditions of Applicability in the Technical Specification for the supported system. The Spent Fuel Pool Special Ventilation System and its actuation instrumentation are not accident initiators; therefore, the proposed changes do not affect the probability of an accident. With the proposed change, the Technical Specifications will continue to require the system actuation instrumentation to be operable when irradiated fuel is moved in the fuel pool enclosure which is also the required Applicability in the supported system Technical Specification. Since the instrumentation will be required to actuate the supported system when it is required to operate, the accident consequences will continue to be mitigated with this proposed Technical Specification change. Thus, the proposed Technical Specification change does not involve a significant increase in the consequences of an accident previously evaluated.

The Control Room Special Ventilation System provides an enclosed control room environment from which the plant can be operated following an uncontrolled release of radioactivity. This system is not an accident initiator, thus the proposed changes do not increase the probability of an accident. This license amendment proposes changes which will: (1) Reduce the time to shut down the plant when Technical Specification required actions or completion time is not met; and (2) clarifies surveillance requirements to assure that the system performs as designed. These changes do not impact the performance of the system; thus this change does not involve a significant increase in the consequences of an accident previously evaluated.

Updating the reference in Technical Specification Chapter 4.0 is an administrative change because the revised document does not change any basis for the current Technical Specifications. Since this is an administrative change, it does not involve a significant increase in the probability or consequences of a previously evaluated accident.

The changes proposed in this license amendment do not involve a significant 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.

This license amendment request proposes changes to the Prairie Island Nuclear Generating Plant Technical Specifications as follows: Technical Specification 1.3, ''Completion Times'', revise a text header and add a new text header; Technical Specification 3.1.4, ''Rod Group Alignment Limits'', remove a Surveillance Note which cross-references another Technical Specification and may cause confusion; Technical Specification 3.3.7, ''Spent Fuel Pool Special Ventilation System (SFPSVS) Actuation Instrumentation'', revises the Modes of Applicability consistent with plant design and the Technical Specifications for the Spent Fuel Pool Special Ventilation System, the supported system; Technical Specification 3.7.10, ''Control Room Special Ventilation System (CRSVS)'', revises the applicability of Condition C and clarifies the requirements of the Surveillance to verify train filtration flow; and Technical Specification Chapter 4.0, ''Design Features'', revises Reference 1 to the most recent version of the document.

Revising and adding text headers in Technical Specification 1.3 are administrative changes because the revised document does not change any basis for the current Technical Specifications. Since these are administrative changes, they do not create the possibility of a new or different kind of accident.

Removal of a surveillance note from Technical Specification 3.1.4 that cross-references another Technical Specification does not change any plant operations, maintenance activities or testing requirements. The Limiting Conditions for Operation will continue to be met and the proper control rod positions will continue to be maintained. There are no new failure modes or mechanisms created through the removal of the Surveillance Requirements Note, nor are new accident precursors generated by this change. This proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed revision of Modes of Applicability for the Spent Fuel Pool Special Ventilation System actuation instrumentation makes operation of the actuation instrumentation consistent with the Technical Specification requirements for the supported system and does not change the operation of the supported system for accident mitigation. The Limiting Conditions for Operation will continue to be met, no new failure modes or mechanisms are created and no new accident precursors are generated by this change. This proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

The changes proposed for the Control Room Special Ventilation System Technical Specifications do not change any the system operations, maintenance activities or testing requirements. The Limiting Conditions for Operation will continue to be met, no new failure modes or mechanisms are created and no new accident precursors are generated by this change. This proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

Updating the reference in Technical Specification Chapter 4.0 is an administrative change because the revised document does not change any basis for the current Technical Specifications. Since this is an administrative change, it does not create the possibility of a new or different kind of accident.

The Technical Specification changes proposed in this license amendment 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 changes to the Prairie Island Nuclear Generating Plant Technical Specifications as follows: Technical Specification 1.3, ''Completion Times'', revise a text header and add a new text header; Technical Specification 3.1.4, ''Rod Group Alignment Limits'', remove a Surveillance Note which cross-references another Technical Specification and may cause confusion; Technical Specification 3.3.7, ''Spent Fuel Pool Special Ventilation System (SFPSVS) Actuation Instrumentation'', revises the Modes of Applicability consistent with plant design and the Technical Specifications for the Spent Fuel Pool Special Ventilation System, the supported system; Technical Specification 3.7.10, ''Control Room Special Ventilation System (CRSVS)'', revises the applicability of Condition C and clarifies the requirements of the Surveillance to verify train filtration flow; and Technical Specification Chapter 4.0, ''Design Features'', revises Reference 1 to the most recent version of the document.

Revising and adding text headers in Technical Specification 1.3 are administrative changes because the revised document does not change any basis for the current Technical Specifications. Since these are administrative changes, they do not involve a significant reduction in a margin of safety.

Plant operations are required to meet all Technical Specifications for which the Applicability is met; therefore, removal of the cross-reference Note from a Technical Specification 3.1.4 surveillance requirement does not change how the plant is operated and therefore, this change does not involve a significant reduction in a margin of safety.

Technical Specification 3.3.7 provides requirements for actuation instrument which supports the operation of the Spent Fuel Pool Special Ventilation System as required by Technical Specification 3.7.13. The current Applicability for Technical Specification 3.3.7 requires the actuation instrumentation to be operable in Modes which are not required by Technical Specification 3.7.13. This license amendment proposes to make Technical Specification 3.3.7 Applicability the same as Technical Specification 3.7.13. This change does not reduce the conditions or Modes when the Spent Fuel Pool Special Ventilation System will operate and perform its accident mitigation function; thus this change does not involve a significant reduction in a margin of safety.

This license amendment proposes changes to the Control Room Special Ventilation System Technical Specifications which will: (1) Reduce the time to shut down the plant when Technical Specification required actions or completion time is not met; and (2) clarifies surveillance requirements to assure that the system performs as designed. The proposed time to shut down the plant is consistent with other Technical Specifications for shutting down the plant and allows adequate time for an orderly shut down of the plant; thus this change does not involve a significant reduction in a margin of safety. The surveillance requirement clarifications do not reduce any testing requirements and will continue to demonstrate that the system can perform its required safety function and satisfy the Limiting Conditions for Operation. Thus this change does not involve a significant reduction in a margin of safety.

Updating the reference in Technical Specification Chapter 4.0 is an administrative change because the revised document does not change any basis for the current Technical Specifications. Since this is an administrative change, it does not involve a significant reduction in a margin of safety.

The Technical Specification changes proposed in this license amendment 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: M. Murphy (A).

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

Date of amendment request: October 26, 2006.

Description of amendment request: The proposed request would revise the Units 2 and 3 emergency diesel generator (EDG) Technical Specification (TS) Completion Time (CT) from 14 days to 7 days for restoration of an inoperable EDG. The current 14-day CT was based on the assumption that Unit 1 was shut down. The near-term restart of Unit 1 will invalidate this assumption, therefore, the affected CTs are being returned to their original duration 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. Does the proposed Technical Specification change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The EDGs are designed as backup alternating current (AC) power sources in the event of a loss of offsite power. The proposed restoration of the EDG CT to its original TS duration does not change the conditions, operating configurations, or minimum amount of operating equipment assumed in the safety analysis for accident mitigation. No changes are proposed in the manner in which the EDGs provide plant protection or which create new modes of plant operation. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed Technical Specification change create the possibility of

a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment does not introduce new equipment which could create a new or different kind of accident. Existing equipment will not be operated in any new modes or for purposes different than it is now utilized. No new external threats, release pathways, or equipment failure modes are created. Therefore, the implementation of the proposed amendment will not create a possibility for an accident of a new or different type than those previously evaluated.

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

Response: No.

BFN's emergency AC [alternating current] system is designed with sufficient redundancy such that an EDG may be removed from service for maintenance or testing. The remaining EDGs are capable of carrying sufficient electrical loads to satisfy the UFSAR [Updated Final Safety Analysis Report] requirements for accident mitigation or unit safe shutdown. The proposed change does not impact the redundancy or availability requirements of offsite power supplies or change the ability of the plant to cope with station blackout events.

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

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

NRC Branch Chief: L. Raghavan.

U.S. Department of Transportation (USDOT), United States Maritime Administration (MARAD), License No. NS–1, Docket No. 50–238, Nuclear Ship Savannah (NSS)

Date of amendment request: August 7, 2006.

Description of amendment request: The proposed license amendment would modify the Technical Specification (TS) requirements to prepare for decommissioning the NSS. Five TS changes are proposed. Three of the proposed changes are related to allowing the NSS to be berthed at locations other than the James River Reserve Fleet (JRRF), Newport News, Virginia. The fourth proposed change eliminates the need to utilize administrative controls to remove the Containment Vessel (CV) Entry Shield Plugs to perform activities such as surveys, system walkdowns and inspections required for developing a detailed decommissioning plan, schedule and cost estimate.

The fifth proposed change clarifies the TS and eliminates redundancies, subtle differences and inefficiencies in the current TS regarding preventing unauthorized access into the Reactor Compartment and Radiation Control Areas. In addition, MARAD is enhancing the numbering of the TSs to remove ambiguities that exist in the current numbering (*e.g.*, TS 2.2 is found on pages 3 and 11 of the current TSs).

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

Response: No.

Proposed changes (1) Ship's Location, (2) Review and Audit Committee Membership, (3) Qualification to perform Surveys and Surveillances, (4) CV Entry Shield Plugs and (5) RC and RCA Entrances are administrative in nature and do not involve the modification of any plant equipment or affect basic plant operation.

The NSS's reactor is not operational and the level of radioactivity in the NSS has significantly decreased from the levels that existed when the 1976 Possession-only License was issued. No aspect of any of proposed changes is an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased.

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

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

Response: No.

Proposed changes (1) Ship's Location, (2) Review and Audit Committee Membership, (3) Qualification to perform Surveys and Surveillances, (4) CV Entry Shield Plugs and (5) RC and RCA Entrances are administrative and do not involve any physical alteration of plant equipment that was not previously allowed by Technical Specifications. These proposed changes do not change the method by which any safety-related system performs its function. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. The methods governing plant operation and testing remain consistent with current safety analysis assumptions.

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

(3) Does the change involve a significant reduction in a margin of safety?

Response: No.

Proposed changes (1) Ship's Location, (2) Review and Audit Committee Membership, (3) Qualification to perform Surveys and Surveillances, (4) CV Entry Shield Plugs and (5) RC and RCA Entrances are administrative in nature. No margins of safety exist that are relevant to the ship's defueled and partially dismantled reactor. 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 changes. The proposed changes involve movement of the ship, changes in the performance of responsibilities and significantly improved radiological conditions since 1976.

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 upon the staff's review of the licensee's analysis, as well as the staff's own evaluation, the staff concludes 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.

Senior Technical Advisor, N.S. Savannah: Erhard W. Koehler, MARAD, Office of Ship Operations.

NRC Branch Chief: Claudia Craig.

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*.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units Nos. 1, 2, and 3, Maricopa County, Arizona

Date of application for amendments: September 29, 2005, as supplemented by letter dated July 5, 2006.

Brief description of amendments: These amendments modified the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Security Program.

Date of issuance: October 31, 2006.

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

Amendment Nos.: Unit 1–162, Unit 2–162, Unit 3–162.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Operating Licenses for all three units.

Date of initial notice in ***Federal Register:*** August 1, 2006 (71 FR 43530).

The July 5, 2006, letter contained the no significant hazards consideration determination for the September 29, 2005, letter that was published in the August 1, 2006, notice. The July 5, 2006, supplemental letter provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Letter contained the no significant hazards consideration determination for the September 29, 2005, letter that was published in the August 1, 2006, notice. The July 5, 2006, supplemental letter provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: January 4, 2006.

Brief description of amendment: The proposed amendment changed the Millstone Power Station, Unit No. 2 Technical Specification (TS) 3/4 3.3.8, ''Instrumentation, Accident Monitoring,'' to modify the description of the pressurizer power operated relief valves and pressurizer safety valves position indicators.

Date of issuance: November 7, 2006.

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

Amendment No.: 294.

Facility Operating License No. DPR–65: The amendment revised the TSs.

Date of initial notice in ***Federal Register:*** February 28, 2006 (71 FR 10073).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 7, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 15, 2005.

Brief description of amendment: The amendment modified the technical specifications to clarify the wording of the emergency closed cooling water (ECCW) Surveillance Requirement 3.7.10.2 that verified actuation of the entire ECCW system rather than just verifying ''valve'' actuation.

Date of issuance: October 27, 2006.

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

Amendment No.: 139.

Facility Operating License No. NPF–58: This amendment revised the Technical Specification Surveillance Requirements and License.

Date of initial notice in ***Federal Register***: January 31, 2006 (71 FR 5081).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 27, 2006.

No significant hazards consideration comments received: No.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of application for amendments: April 27, 2006, as supplemented October 3, 2006.

Brief description of amendments: The amendments revise, on a one-time basis, Technical Specification 3/4.4.5, Steam Generator (SG) Surveillance Requirements, to exclude the region of the SG tubes below 17 inches from the top of the hot leg tube sheet from the inspection requirements. The amendments also permanently revise the limit for primary-to-secondary leakage in TS 3/4.4.6, Reactor Coolant System Leakage.

Date of issuance: November 1, 2006.

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

Amendment Nos: 231 and 226.

Renewed Facility Operating License Nos. DPR–31 and DPR–41: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 1, 2006 (71 FR 43532).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 1, 2006.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: March 7, 2006, as supplemented by letter dated August 3, 2006.

Brief description of amendments: The amendment revised Section 3.3.1, ''Reactor Trip System (RTS) Instrumentation,'' of the DCCNP–1 and DCCNP–2 Technical Specifications, changing the reactor trip on turbine trip interlock from the P–7 setpoint (10 percent rated thermal power) to the P–8 setpoint (31 percent rated thermal power).

Date of issuance: October 30, 2006.

Effective date: As of the date of issuance and shall be implemented prior to entry into Mode 1 from the Cycle 21 refueling outage for DCCNP–1, and prior to entry into Mode 1 from the Cycle 17 refueling outage for DCCNP–2.

Amendment Nos.: 297 and 298.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revise the Technical Specifications.

Date of initial notice in ***Federal Register***: April 25, 2006 (71 FR 23956).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 30, 2006.

No significant hazards consideration comments received: No.

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

Date of amendment request: March 15, 2006.

Brief description of amendment: The amendment revised the Cooper Nuclear Station Technical Specification 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' by adding two sub-paragraphs to note exemptions from Section III.A and Section lII.B of 10 CFR Part 50, Appendix J, Option B. These two sub-paragraphs allow the leakage contribution from the four main steam line penetrations, referred to as the Main Steam Isolation Valve leakage, to be excluded.

Date of issuance: October 31, 2006.

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

Amendment No.: 226.

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

Date of initial notice in ***Federal Register***: April 25, 2006 (71 FR 23958).

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

No significant hazards consideration comments received: No.

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

Date of amendment request: October 31, 2005, as supplemented on July 25, 2006.

Brief description of amendment: The amendment revised the FCS Updated Safety Analysis Report Sections related to the radiological consequences of events affected by the planned 2006 replacement of the steam generators and pressurizer.

Date of issuance: October 27, 2006.

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

Amendment No.: 243.

Renewed Facility Operating License No. DPR–40: The amendment revised the Updated Safety Analysis Report.

Date of initial notice in ***Federal Register***: December 20, 2005 (70 FR 75493).

The July 25, 2006, supplemental letter provided information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a safety evaluation dated October 27, 2006.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 19, 2005, as supplemented on May 30, 2006.

Brief description of amendment: The amendment modified Fort Calhoun Station, Unit No. 1's Technical Specification 2.4, ''Containment Cooling,'' (and the associated Bases) to reduce the required number of operable containment spray (CS) pumps from three to two in order to enhance net positive suction head margins. The proposed change was implemented by disabling the CS actuation signal automatic start feature of one of the two CS pumps that share the same diesel generator and a common suction line.

Date of issuance: October 27, 2006.

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

Amendment No.: 244.

Renewed Facility Operating License No. DPR–40: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: February 28, 2006 (71 FR 10075).

The May 30, 2006, supplemental letter provided information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a safety evaluation dated October 27, 2006.

No significant hazards consideration comments received: No.

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

Date of amendment request: September 30, 2005, as supplemented by letters dated May 23 and August 16, 2006.

Brief description of amendment: Omaha Public Power District proposed 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 NP, Inc. 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). This change is necessary since the EMF–2087(P)(A) methodology is not approved for analyzing M5 clad fuel, which will be used in the FCS reactor core starting in Cycle 24. As part of this approval, the NRC staff reviewed the AREVA NP, Inc. FCS-specific LBLOCA analysis using EMF–2103(P)(A). EMF–2103(P)(A) will be used for Cycle 24 and beyond.

Date of issuance: November 3, 2006.

Effective date: Effective as of its date of issuance and shall be implemented within 90 days of issuance.

Amendment No.: 245.

Renewed Facility Operating License No. DPR–40: The amendment revised the COLR.

Date of initial notice in ***Federal Register***: January 3, 2006 (71 FR 152).

The May 23 and August 16, 2006, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a safety evaluation dated November 3, 2006.

No significant hazards consideration comments received: No.

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

Date of amendment request: May 30, 2006, as supplemented by two letters dated on August 30, 2006.

Brief description of amendment: The amendment revised the Fort Calhoun Station, Unit No. 1 (FCS) Technical Specification (TS) requirements related to steam generator tube integrity. The change is consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF–449, ''Steam Generator Tube Integrity.'' The availability of this TS improvement was announced in the **Federal Register** on May 6, 2005 (70 FR 24126) as part of the consolidated line item improvement process (CLIIP).

OPPD also changed the FCS TS by deleting the sleeving repair alternative to plugging for steam generator tubes. The FCS replacement steam generators (RSGs) to be installed during the fall of 2006 are manufactured by Mitsubishi Heavy Industries, Ltd. (MHI). OPPD has stated that the sleeving repair alternative to plugging will not be used for the MHI RSGs.

Date of issuance: November 7, 2006.

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

Amendment No.: 246.

Renewed Facility Operating License No. DPR–40: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register***: July 18, 2006 (71 FR 40750).

The two August 30, 2006, supplemental letters provided information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a safety evaluation dated November 7, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 7, 2005, as supplemented by letter dated September 8, 2006.

Brief description of amendment: The proposed amendment revised the Technical Specifications (TSs) to clarify certain requirements during fuel movement, core alterations, and operations with the potential for draining the reactor vessel. The amendment better aligns the TSs with the NRC-approved Revision 2 to Technical Specification Task Force (TSTF) Traveler TSTF–51, ''Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations,'' and NUREG–1433, ''Standard Technical Specifications General Electric Plants, BWR [boiling water reactor]/4.''

Date of issuance: October 31, 2006.

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

Amendment No.: 170.

Facility Operating License No. NPF–57: This amendment revised the TSs.

*Date of initial notice in **Federal Register***: May 9, 2006 (71 FR 27002).

The licensee's September 8, 2006, supplement provided clarifying information that did not change the scope of the proposed amendment as described in the original notice of proposed action published in the **Federal Register**, and did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: April 25, 2006.

Brief description of amendments: The amendments revised the Technical Specifications to adopt the provisions in Technical Specification Task Force (TSTF) Traveler TSTF–359, ''Increased Flexibility in Mode Restraints,'' Revision 9. The availability of TSTF–359 for adoption by licensees was announced in the **Federal Register** on April 4, 2003 (68 FR 16579).

Date of issuance: October 27, 2006.

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

Amendment Nos.: 276, 258.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Technical Specifications and License.

*Date of initial notice in **Federal Register***: July 5, 2006 (71 FR 38185).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 27, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 28, 2005, as supplemented on April 2, June 15, and August 31, 2006.

Brief description of amendment: The amendment revises the Virgil C. Summer Nuclear Station Technical Specifications and provides associated Bases to permit the implementation of an alternate alternating current power supply.

Date of issuance: November 2, 2006.

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

Amendment No. 178.

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

*Date of initial notice in **Federal Register***: March 14, 2006 (71 FR 13176).

The supplemental letter provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 2, 2006.

No significant hazards consideration comments received: No.

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

Date of amendment request: October 27, 2005.

Brief description of amendment: The amendment revised Technical Specifications (TSs) 1.1, ''Definitions,'' and 3.4.16, ''RCS [reactor coolant system] Specific Activity,'' to replace the current Limiting Condition for Operation (LCO) 3.4.16 limits on RCS specific activity with limits on RCS Dose Equivalent I–131 (DEI) and Dose Equivalent Xe-133 (DEX). In TS 1.1, the definition of (1) $\bar{E}$—Average Disintegration Energy is replaced by the definition of DEX and (2) DEI is revised to allow the use of alternate thyroid dose conversion factors. The modes of applicability, conditions and required actions, and surveillance requirements for TS 3.4.16 are revised.

Date of issuance: October 31, 2006.

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

Amendment No.: 170.

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

*Date of initial notice in **Federal Register***: January 3, 2006 (71 FR 156).

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

No significant hazards consideration comments received: No.

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

Date of amendment request: August 25, 2006, as supplemented by letter dated October 25, 2006.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.7.2, ''Main Steam Isolation Valves (MSIVs),'' and TS 3.7.3, ''Main Feedwater Isolation Valves (MFIVs),'' to add the associated actuator trains to (1) the limiting condition for operation (LCO), (2) the conditions, required actions, and completion times for the LCO, and (3) the surveillance requirements. The Table of Contents for the TSs is changed to account for the resulting renumbering of TS pages.

Date of issuance: November 7, 2006.

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

Amendment No.: 171.

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

Date of initial notice in ***Federal Register***: September 1, 2006 (71 FR 52173).

The supplemental letter dated October 25, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 7, 2006.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 9th day of November, 2006.

For The Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E6–19434 Filed 11–20–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[NUREG–1852]

Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire, Draft Report for Comment

AGENCY: Nuclear Regulatory Commission.

ACTION: Extension of comment period for NUREG–1852, ''Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire, Draft Report for Comment.''

SUMMARY: On October 12, 2006 (71 FR 60200), the Nuclear Regulatory Commission (NRC) issued for public comment NUREG 1852, ''Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire, Draft Report for Comment.'' Due to an error in the previous notice of comment period extension, a request has been made to extend the public comment period to allow the public 60 days to review the document. Currently, the **Federal Register** specifies that the public comment period ends on December 12, 2006.

DATES: The comment period has been extended and now expires on January 30, 2007. Comments received after this date will be considered if it is practical to do so, but the Commission is able to ensure consideration only for comments received before this date.

ADDRESSES: Members of the public are invited and encouraged to submit written comments to Michael Lesar, Chief, Rules and Directives Branch, Office of Administration, Mail Stop T6–D59, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Hand-deliver comments attention to Michael Lesar, 11545 Rockville Pike, Rockville, MD, between 7:30 a.m. and 4:15 p.m. on Federal workdays. Comments may also be sent electronically to *NRCREP@nrc.gov.*

This document, NUREG–1852, is available at the Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* under Accession No. ML062350292; on the NRC Web site at *http://www.nrc.gov/reading-rm/doc-collections/nuregs/docs4comment.html;* and at the NRC Public Document Room, 11555 Rockville Pike, Rockville, MD. The PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415–4737 or (800) 397–4205; fax (301) 415–3548; e-mail *PDR@NRC.GOV.*

FOR FURTHER INFORMATION CONTACT: Erasmia Lois, Human Factors and Reliability Branch, Office of Nuclear Regulatory Research, telephone: (301) 415–6560; e-mail: *exl1@nrc.gov.*

Dated at Rockville, Maryland, this 15th day of November, 2006.

For the Nuclear Regulatory Commission.

Jose Ibarra,

Chief, Human Factors and Reliability Branch, Probabilistic Risk and Applications, Division of Risk Assessment and Special Projects, Office of Nuclear Regulatory Research.

[FR Doc. E6–19626 Filed 11–20–06; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

General Schedule Locality Pay Areas

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: On behalf of the President's Pay Agent, the Office of Personnel Management (OPM) is providing notice about two changes in locality pay area boundaries in 2007 under the locality pay program for General Schedule and certain other employees. Grayson County, TX, will be added to the Dallas locality pay area, and Berks County, PA, will be added to the Philadelphia locality pay area. These changes will occur automatically under existing regulations. OPM also plans to issue a notice later about changes in the regulations needed to update the official descriptions of the Boston-Worcester-Manchester, MA-NH-ME-RI locality pay area and the Denver-Aurora-Boulder, CO locality pay area. As required by OPM regulations, the additions to locality pay areas are effective as of the first pay period beginning on or after January 1, 2007. Both the additions and the planned description changes are the result of changes made by the Office of Management and Budget in Metropolitan Statistical Areas and Combined Statistical Areas.

DATES: The additions to locality pay areas are applicable on the first day of the first pay period beginning on or after January 1, 2007.

FOR FURTHER INFORMATION CONTACT: Allan Hearne, (202) 606–2838; *FAX:* (202) 606–4264; *e-mail: pay-performance-policy@opm.gov.*

Section 5304 of title 5, United States Code, authorizes locality pay for General Schedule (GS) employees with duty stations in the contiguous United States and the District of Columbia.

Section 5304(f) of title 5, United States Code, authorizes the President's Pay Agent (the Secretary of Labor, the Director of the Office of Management and Budget (OMB), and the Director of the Office of Personnel Management (OPM) to determine locality pay areas. The boundaries of locality pay areas must be based on appropriate factors,