Data collection activity	Number of respondents	Frequency	Total responses	Average time per response (minutes)	Burden hours
ETA Form 790	8,356	Occasionally	8,356	60	8,356
ETA Form 795	1,000	Occasionally	1,000	15	250
Total	9,356	Occasionally	9,356	75	8,606

The calculations are based on a normal biweekly work week, as most jobs are 8 hours a day for 10 days bi-weekly. There are 26 bi-weeks in a year. Therefore, 8 hours × 10 days × 26 bi-weeks = 2,080 hours worked in a year. Also the calculations are based on the average median salary of a state worker of $69,992 per year, and the estimated hours expended in completing and processing ETA Form 790 and ETA Form 795 respectively, providing the grand total of burden cost reflected above.

The burden is estimated to be 60 minutes for Form 790 and 15 minutes for Form 795:

- ETA 790—8,356 multiplied by 60 minutes = 501,360 divided by 60 = 8,356;
- ETA 795—1,000 multiplied by 15 minutes = 15,000 divided by 60 = 250;
- The average median salary of a state works is $69,992 divided by 2,080 hours = $33.65 P/Hr.;
- The annual hours of 8,606 multiplied by the hourly rate of $33.65 = $289,592 total annual burden cost.

The estimate above is based on the Bureau of Labor Statistics data provided in the Occupational Employment Statistics (OES) at *www.bls.gov.* In calculating the cost of completing and processing of the forms, the hourly rate of $33.65 per hour was used.

Comments submitted in response to this comment request will be summarized and/or included in the request for OMB approval of the ICR; they will also become a matter of public record.

Dated: Signed in Washington, DC, on this 8th day of May 2012.

Jane Oates,

Assistant Secretary, Employment and Training Administration.

[FR Doc. 2012–11628 Filed 5–14–12; 8:45 am]

BILLING CODE 4510–FN–P

MORRIS K. UDALL AND STEWART L. UDALL FOUNDATION

Sunshine Act Meetings

TIME AND DATE: 8:30 a.m. to 12:30 p.m., Tuesday, May 22, 2012.

PLACE: JW Marriott Starr Pass, 3800 W. Starr Pass Boulevard, Tucson, Arizona 85745.

STATUS: This meeting will be open to the public, unless it is necessary for the Board to consider items in executive session.

MATTERS TO BE CONSIDERED: (1) Program reports; (2) management committee report; (3) Parks in Focus Program report; (4) financial scenarios report; (5) Board governance and (6) personnel matters.

PORTIONS OPEN TO THE PUBLIC: All agenda items except as noted below.

PORTIONS CLOSED TO THE PUBLIC: Executive session to review personnel matters.

CONTACT PERSON FOR MORE INFORMATION: Ellen K. Wheeler, Executive Director, 130 South Scott Avenue, Tucson, AZ 85701, (520) 901–8500.

Dated: May 7, 2012.

Ellen K. Wheeler,

Executive Director, Morris K. Udall and Stewart L. Udall Foundation, and Federal Register Liaison Officer.

[FR Doc. 2012–11455 Filed 5–14–12; 8:45 am]

BILLING CODE 6820–FN–M

NUCLEAR REGULATORY COMMISSION

[NRC–2012–0107]

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

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) 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 or combined license, as applicable, 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 April 19, 2012 to May 2, 2012. The last biweekly notice was published on May 1, 2012 (77 FR 25753).

ADDRESSES: You may access information and comment submissions related to this document, which the NRC possesses and is publicly available, by searching on *http://www.regulations.gov* under Docket ID NRC–2012–0107. You may submit comments by the following methods:

- *Federal Rulemaking Web Site:* Go to *http://www.regulations.gov* and search for Docket ID NRC–2012–0107. Address questions about NRC dockets to Carol Gallagher; telephone: 301–492–3668; email: *Carol.Gallagher@nrc.gov.*
- *Mail comments to:* Cindy Bladey, Chief, Rules, Announcements, and Directives Branch (RADB), Office of Administration, Mail Stop: TWB–05–B01M, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.
- *Fax comments to:* RADB at 301–492–3446.

For additional direction on accessing information and submitting comments, see ''Accessing Information and Submitting Comments'' in the **SUPPLEMENTARY INFORMATION** section of this document.

SUPPLEMENTARY INFORMATION:

I. Accessing Information and Submitting Comments

A. Accessing Information

Please refer to Docket ID NRC–2012–0107 when contacting the NRC about the availability of information regarding this document. You may access information related to this document, which the NRC possesses and is publicly available, by the following methods:

- *Federal Rulemaking Web Site:* Go to *http://www.regulations.gov* and search for Docket ID NRC–2012–0107.
- *NRC's Agencywide Documents Access and Management System (ADAMS):* You may access publicly available documents online in the NRC

Library at *http://www.nrc.gov/reading-rm/adams.html.* To begin the search, select ''ADAMS Public Documents'' and then select ''Begin Web-based ADAMS Search.'' For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1–800–397–4209, 301–415–4737, or by email to *pdr.resource@nrc.gov.* Documents may be viewed in ADAMS by performing a search on the document date and docket number.

• *NRC's PDR:* You may examine and purchase copies of public documents at the NRC's PDR, Room O1–F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

B. Submitting Comments

Please include Docket ID NRC–2012–0107 in the subject line of your comment submission, in order to ensure that the NRC is able to make your comment submission available to the public in this docket.

The NRC cautions you not to include identifying or contact information in comment submissions that you do not want to be publicly disclosed. The NRC posts all comment submissions at *http://www.regulations.gov* as well as entering the comment submissions into ADAMS, and the NRC does not edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information in their comment submissions that they do not want to be publicly disclosed. Your request should state that the NRC will not edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment submissions into ADAMS.

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Combined 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 Title 10 of the Code of Federal Regulations (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.

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.

Within 60 days after the date of publication of this notice, any person(s) whose interest may be affected by this action may file a request for a hearing and a petition to intervene with respect to issuance of the amendment to the subject facility operating license or combined license. 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 person(s) should consult a current copy of 10 CFR 2.309, which is available at the NRC's PDR, located at One White Flint North, Room O1–F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. NRC regulations are accessible electronically from the NRC Library on the NRC's Web site at *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed by the above date, 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 identify the specific contentions which the requestor/petitioner 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 requestor/petitioner 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 requestor/petitioner intends to rely in proving the contention at the hearing. The requestor/petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the requestor/petitioner 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 requestor/petitioner to relief. A requestor/petitioner 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, 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, then any hearing held would take place before the issuance of any amendment.

All documents filed in NRC adjudicatory proceedings, including a request for hearing, a petition for leave to intervene, any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities participating under 10 CFR 2.315(c), must be filed in accordance with the NRC E–Filing rule (72 FR 49139, August 28, 2007). The E–Filing process requires participants to submit and serve all adjudicatory documents over the Internet, or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek an exemption in accordance with the procedures described below.

To comply with the procedural requirements of E–Filing, at least 10 days prior to the filing deadline, the participant should contact the Office of the Secretary by email at *hearing.docket@nrc.gov,* or by telephone at 301–415–1677, to request (1) a digital identification (ID) certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E–Submittal server for any proceeding in which it is participating; and (2) advise the Secretary that the participant will be submitting a request or petition for hearing (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

Information about applying for a digital ID certificate is available on the NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals/apply-certificates.html.* System requirements for accessing the E–Submittal server are detailed in the NRC's ''Guidance for Electronic Submission,'' which is available on the agency's public Web site at *http://www.nrc.gov/site-help/e-submittals.html.* Participants may attempt to use other software not listed on the Web site, but should note that the NRC's E–Filing system does not support unlisted software, and the NRC Meta System Help Desk will not be able to offer assistance in using unlisted software.

If a participant is electronically submitting a document to the NRC in accordance with the E–Filing rule, the participant must file the document using the NRC's online, Web-based submission form. In order to serve documents through the Electronic Information Exchange System, users will be required to install a Web browser plug-in from the NRC's Web site. Further information on the Web-based submission form, including the installation of the Web browser plug-in, is available on the NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals.html.*

Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with the NRC guidance available on the NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals.html.* A filing is considered complete at the time the documents are submitted through the NRC's E–Filing system. To be timely, an electronic filing must be submitted to the E–Filing system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E–Filing system time-stamps the document and sends the submitter an email notice confirming receipt of the document. The E–Filing system also distributes an email notice that provides access to the document to the NRC's Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E–Filing system.

A person filing electronically using the agency's adjudicatory E–Filing system may seek assistance by contacting the NRC Meta System Help Desk through the ''Contact Us'' link located on the NRC Web site at *http://www.nrc.gov/site-help/e-submittals.html,* by email at *MSHD.Resource@nrc.gov,* or by a toll-free call at 1–866 672–7640. The NRC Meta System Help Desk is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted 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; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service. A presiding officer, having granted an exemption request from using E–Filing, may require a participant or party to use E–Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E–Filing no longer exists.

Documents submitted in adjudicatory proceedings will appear in the NRC's electronic hearing docket which is available to the public at *http://ehd1.nrc.gov/ehd/,* unless excluded pursuant to an order of the Commission, or the presiding officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

Petitions for leave to intervene must be filed no later than 60 days from the date of publication of this notice. Nontimely filings will not be entertained absent a determination by the presiding officer that the petition or request should be granted or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)–(viii).

For further details with respect to this license amendment application, see the application for amendment which is available for public inspection at the NRC's PDR, located at One White Flint North, Room O1–F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available documents created or received at the NRC are accessible electronically through ADAMS in the NRC Library at *http://www.nrc.gov/reading-rm/adams.html.* Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS should contact the NRC's PDR Reference staff at 1–800–397–4209, 301–415–4737, or by email to *pdr.resource@nrc.gov.*

Dominion Nuclear Connecticut, Inc., Docket No. 50–423, Millstone Power Station, Unit 3, New London County, Connecticut

Date of amendment request: November 17, 2011.

Description of amendment request: The proposed amendment would add Optimized ZIRLO™ as an allowable fuel rod cladding material and add the Westinghouse topical report on Optimized ZIRLO™ to the Millstone Power Station, Unit 3 (MPS3) Technical Specifications. In addition, a typographical error would be corrected.

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

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

Response: No.

The proposed Technical Specifications changes are: (1) Adding Optimized ZIRLO™ to the allowable or approved cladding materials to be used at MPS3, and (2) correcting a typographical error in the title of Reference 8 in Technical Specification (TS) 6.9.1.6.b. The proposed change of adding a cladding material does not result in an increase to the probability or consequences of an accident previously evaluated. Technical Specification 5.3.1 addresses the fuel assembly design, and currently specifies that ''Each assembly shall consist of a matrix of Zircaloy or ZIRLO® fuel rods * * *''. The proposed change will add Optimized ZIRLO™ to the approved fuel rod cladding materials listed in this technical specification. In addition, a reference to the topical report for Optimized ZIRLO™, WCAP–12610–P–A and CENPD–404–P–A, Addendum 1–A, will be added to the listing of approved methods used to determine the core operating limits for MPS3 provided in Technical Specification 6.9.1.6.b.

Westinghouse topical report WCAP–12610–P–A & CENPD–404–P–A, Addendum 1–A, ''Optimized ZIRLOTM,'' provides the details and results of material testing of Optimized ZIRLO™ compared to standard ZIRLO®, as well as the material properties to be used in various models and methodologies when analyzing Optimized ZIRLO™. As the nuclear industry pursues longer operating cycles with increased fuel discharge burnup and fuel duty, the corrosion performance requirements for the nuclear fuel cladding become more demanding. Optimized ZIRLO™ was developed to meet these industry needs by providing a reduced corrosion rate while maintaining the composition and physical properties, such as mechanical strength, similar to standard ZIRLO®. In addition, margin to the fuel rod design criterion on fuel rod internal pressure has been impacted by increased fuel duty, use of integral fuel burnable absorbers, and corrosion/temperature feedback effects. Reducing the associated corrosion buildup reduces temperature feedback effects, providing additional margin to the fuel rod internal pressure design criterion. The fuel will continue to satisfy the pertinent design basis operating limits, so cladding integrity is maintained. There are no changes that will adversely affect the ability of existing components and systems to mitigate the consequences of any accident. Addition of Optimized ZIRLO™ to the allowable cladding materials for MPS3 therefore does not result in a significant increase in the probability or consequences of an accident previously evaluated.

The NRC has previously approved use of Optimized ZIRLO™ fuel cladding material in Westinghouse fueled reactors provided that licensees ensure compliance with the Conditions and Limitations set forth in the NRC Safety Evaluation for the topical report. Confirmation that these Conditions are satisfied is performed under 10 CFR 50.59 as part of the normal core reload process.

The change to the title of Reference 8 in Technical Specification 6.9.1.6.b is administrative in nature and does not alter any of the requirements of the affected TS. The proposed change does not modify any plant equipment and does not impact any failure modes that could lead to an accident. Additionally, the proposed change has no effect on the consequence of any analyzed accident since the change does not affect any equipment related to accident mitigation.

Based on this discussion, the proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed Technical Specifications change adds Optimized ZIRLO™ to the approved fuel rod cladding materials that may be used at MPS3. Optimized ZIRLO™ was developed to provide a reduced cladding corrosion rate while maintaining the benefits of mechanical strength and resistance to accelerated corrosion from potential abnormal chemistry conditions. The fuel rod design bases are established to satisfy the general and specific safety criteria addressed in the MPS3 Final Safety Analysis Report (FSAR), Chapter 15 (Accident Analyses). The fuel rods are designed to prevent excessive fuel temperatures, excessive fuel rod internal gas pressures due to fission gas releases, and excessive cladding stresses and strains. Westinghouse topical report WCAP–12610–P–A & CENPD–404–P–A, Addendum 1–A, ''Optimized ZIRLO™,'' provides the details and results of material testing of Optimized ZIRLO™ compared to standard ZIRLO®, as well as the material properties to be used in various models and methodologies when analyzing Optimized ZIRLO™. The original fuel design basis requirements have been maintained. No new single failure mechanisms will be created, and there are no alterations to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors.

Therefore, the proposed changes to the MPS3 TSs related to the use of Optimized ZIRLO™ do not create the possibility of a new or different kind of accident or malfunction from those previously evaluated within the FSAR.

The change to the title of Reference 8 in Technical Specification 6.9.1.6.b is administrative in nature. It does not modify any plant equipment and there is no impact on the capability of the existing equipment to perform their intended functions. No system setpoints are being modified and no changes are being made to the method in which plant operations are conducted. No new failure modes are introduced by the proposed changes. The proposed change does not introduce accident initiators or malfunctions that would cause a new or different kind of accident.

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

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

Response: No.

The cladding materials used for fuel rods are designed and tested to prevent excessive fuel temperatures, excessive fuel rod internal gas pressures due to fission .as releases, and excessive cladding stresses and strains. Optimized ZIRLO™ was developed to meet these needs while providing a reduced cladding corrosion rate and maintaining the benefits of mechanical strength and resistance to accelerated corrosion from potential abnormal chemistry conditions. Reducing the associated corrosion buildup reduces temperature feedback effects, providing additional margin to the fuel rod internal pressure design criterion. Westinghouse topical report WCAP–12610–P–A & CENPD–404–P–A, Addendum 1–A, ''Optimized ZIRLO™'' provides the details and results of material testing of Optimized ZIRLO™ compared to standard ZIRLO®, as well as the material properties to be used in various models and methodologies when analyzing Optimized ZIRLO™. The NRC has previously approved use of the Optimized ZIRLO™ fuel cladding material as detailed in their Safety Evaluation for this topical report. The original fuel design basis requirements have been maintained, and evaluations will be performed under 10 CFR 50.59 for each reload cycle that incorporates Optimized ZIRLO™ cladding to confirm that design and safety limits are satisfied. Therefore, inclusion of Optimized ZIRLO™ as an additional fuel rod cladding material for MPS3 does not result in a significant reduction in margin required to preclude or reduce the effects of an accident or malfunction previously evaluated in the FSAR.

The change to the title of Reference 8 in Technical Specification 6.9.1.6.b is administrative in nature. This change does not alter any of the requirements of the affected TS. The proposed change does not affect any of the assumptions used in the accident analysis, nor does it affect any operability requirements for equipment important to plant safety.

Therefore, the proposed change will not result in a significant reduction in the margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Senior Counsel, Dominion Resources Services, Inc., 120 Tredegar Street, RS–2, Richmond, VA 23219.

NRC Branch Chief: George A. Wilson.

Duke Energy Carolinas, LLC, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: March 5, 2012.

Description of amendment request: The proposed amendments would implement a measurement uncertainty recapture power uprate at the McGuire Nuclear Station, Units 1 and 2.

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

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

Response: No.

The proposed amendment changes the rated thermal power from 3411 megawatts thermal (MWt) to 3469 MWt; an increase of approximately 1.7% Rated Thermal Power. Duke Energy's evaluations have shown that all structures, systems and components (SSCs) are capable of performing their design function at the uprated power of 3469 MWt. A review of station accident analyses found that all acceptance criteria are still met at the uprated power of 3469 MWt.

The radiological consequences of operation at the uprated power conditions have been assessed. The proposed power uprate does not affect release paths, frequency of release, or the analyzed reactor core fission product inventory for any accidents previously evaluated in the Final Safety Analysis Report. Analyses performed to assess the effects of mass and energy releases remain valid. All acceptance criteria for radiological consequences continue to be met at the uprated power level.

As summarized in Sections IV, V, and VI of Enclosure 2, the proposed change does not involve any change to the design or functional requirements of the safety and support systems. That is, the increased power level neither degrades the performance of, nor increases the challenges to any safety systems assumed to function in the plant safety analysis.

While power level is an input to accident analyses, it is not an initiator of accidents. The proposed change does not affect any accident precursors and does not introduce any accident initiators. The proposed change does not impact the usefulness of the Surveillance Requirements (SRs) in evaluating the operability of required systems and components.

In addition, evaluation of the proposed TS change demonstrates that the availability of equipment and systems required to prevent or mitigate the radiological consequences of an accident is not significantly affected. Since the impact on the systems is minimal, it is concluded that the overall impact on the plant safety analysis is negligible.

Therefore, the proposed TS changes do not significantly increase 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.

A Failure Modes and Effects Analysis of the new system was performed and the possible effects of failures of the new equipment and the increased power level on the overall plant systems were reviewed. This review found that no new or different accidents were created by the new equipment or the uprated power levels.

No installed equipment is being operated in a different manner. The proposed changes have no significant adverse affect on any safety-related structures, systems or components and do not significantly change the performance or integrity of any safety-related system.

The proposed changes do not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. The uprated power does not create any new accident initiators. Credible malfunctions are bounded by the current accident analyses of record or recent evaluations demonstrating that applicable criteria are still met with the proposed changes.

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

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

Response: No.

Although the proposed amendment increases the operating power level of the plants, it retains the margin of safety because it is only increasing power by the amount equal to the reduction in uncertainty in the heat balance calculation. The margins of safety associated with the power uprate are those pertaining to core thermal power. These include fuel cladding, reactor coolant system pressure boundary, and containment barriers. Analyses demonstrate that the design basis continues to be met after the measurement uncertainty recapture (MUR) power uprate. Components associated with the reactor coolant system pressure boundary structural integrity, including pressure-temperature limits, vessel fluence, and pressurized thermal shock are bounded by the current analyses. Systems will continue to operate within their design parameters and remain capable of performing their intended safety functions.

The current McGuire safety analyses including the revised design basis radiological accident dose calculations, bound the power uprate.

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: Lara S. Nichols, Associate General Counsel, Duke Energy Corporation, 526 South Church Street—EC07H, Charlotte, NC 28202.

NRC Branch Chief: Nancy L. Salgado.

Duke Energy Carolinas, LLC, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: November 22, 2011.

Description of amendment request: The amendments would revise the Technical Specifications (TSs) to allow single discharge header operation of the nuclear service water system (NSWS) for a time period of 14 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:

First Standard

Does 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 proposed single discharge header operation configuration for NSWS operation and the associated proposed TS and Bases changes have been evaluated to assess their impact on plant operation and to ensure that the design basis safety functions of safety related systems are not adversely impacted. During single discharge header operation, the operating NSWS header will be able to discharge all required NSWS flow from safety related components. [Probabilistic risk assessment] PRA has demonstrated that due to the limited proposed time in the single discharge header configuration, the resultant plant risk remains acceptable.

The purpose of this amendment request is to ultimately facilitate inspection and maintenance of the Unit 2 NSWS discharge headers within the Auxiliary Building. Therefore, NRC approval of this request will ultimately help to enhance the long-term structural integrity of the NSWS and will help to ensure the system's reliability for many years.

In general, the NSWS serves as an accident mitigation system and cannot by itself initiate an accident or transient situation. The only exception is that the NSWS piping can serve as a source of floodwater to safety related equipment in the Auxiliary Building or in the diesel generator buildings in the event of a leak or a break in the system piping. The probability of such an event is not significantly increased as a result of this proposed request. Safety related NSWS piping is tested and inspected in accordance with all applicable inservice testing and inservice inspection requirements. Given the negligible influence of flooding events on the NSWS for the submittal configuration (i.e., no dominant contribution from floods), it is judged that the analyses assessing the influence of these events provide an acceptable evaluation of the contribution of the flood risk for the requested Completion Time of 14 days.

The proposed 14 day TS Required Action Completion Time has been evaluated for risk significance and the results of this evaluation have been found acceptable. The probabilities of occurrence of accidents presented in the [updated final safety analysis report] UFSAR will not increase as a result of implementation of this change. Because the PRA analysis supporting the proposed change yielded acceptable results, the NSWS will maintain its required availability in response to accident situations. Since NSWS availability is maintained, the response of the plant to accident situations will remain acceptable and the consequences of accidents presented in the UFSAR will not increase.

Second Standard

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

Response: No.

Implementation of this amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed request does not affect the basic operation of the NSWS or any of the systems that it supports. These include the Emergency Core Cooling System, the Containment Spray System, the Containment Valve Injection Water System, the Auxiliary Feedwater System, the Component Cooling Water System, the Control Room Area Ventilation System, the Control Room Area Chilled Water System, the Auxiliary Building Filtered Ventilation Exhaust System, or the Diesel Generators. During proposed single discharge header operation, the NSWS will remain capable of fulfilling all of its design basis requirements. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No changes are being made to the plant which will introduce any new type of accident outside those assumed in the UFSAR.

Third Standard

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

Response: No.

Implementation of this amendment will not involve a significant reduction in any margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these fission product barriers will not be impacted by implementation of this proposed TS amendment. During single discharge header operation, the NSWS and its supported systems will remain capable of performing their required functions. No safety margins will be impacted.

The PRA conducted for this proposed amendment demonstrated that the impact on overall plant risk remains acceptable during single discharge header operation. Therefore, there is not a significant reduction in the margin of safety.

Based upon the preceding discussion, Duke Energy has concluded that the proposed amendment does not involve a significant hazards consideration.

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: Lara S. Nichols, Associate General Counsel, Duke Energy Corporation, 526 South Church Street—EC07H, Charlotte, NC 28202.

NRC Branch Chief: Nancy L. Salgado.

Exelon Generation Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of amendment request: March 26, 2012, as supplemented by letter dated April 2, 2012.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) Limiting Condition for Operation 3.1.1.2, TS Surveillance Requirement 4.19.2, TS 6.9.6 ''Steam Generator Tube Inspection Report,'' and TS 6.19 ''Steam Generator (SG) Program,'' changing certain inspection periods and making other administrative changes and clarifications. These proposed changes are consistent with Technical Specification Task Force (TSTF) Traveler, TSTF–510, Revision 2, ''Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection.''

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

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

Response: No

The proposed change revises the Steam Generator (SG) Program to modify the frequency of verification of SG tube integrity and SG tube sample selection. A steam generator tube rupture (SGTR) event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. The proposed SG tube inspection frequency and sample selection criteria will continue to ensure that the SG tubes are inspected such that the probability of a SGTR is not increased. The consequences of a SGTR are bounded by the conservative assumptions in the design basis accident analysis. The proposed change will not cause the consequences of a SGTR to exceed those assumptions. The proposed change to reporting requirements and clarifications of the existing requirements have no affect on the probability or consequences of SGTR.

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

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

Response: No.

The proposed changes to the Steam Generator Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The proposed change does not affect the design of the SGs or their method of operation. In addition, the proposed change does not impact any other plant system or component.

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

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

Response: No.

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes. Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change will continue to require monitoring of the physical condition of the SG tubes such that there will not be a reduction in the margin of safety compared to the current requirements.

Therefore, it is concluded that 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: J. Bradley Fewell, Esquire, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Meena Khanna.

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

Date of amendment request: October 24, 2011.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.3.5, ''Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation,'' to correct the nonconservative first level undervoltage relays (FLUR) limits contained in Surveillance Requirement (SR) 3.3.5.3; revise the Final Safety Analysis Report Update (FSARU) Appendix 6.2D and Sections 6.3, 15.3, and 15.4; revise the loss-of-coolant accident (LOCA) control room operator and offsite dose analysis of record described in the FSARU; and provide a new process for revising input values to this analysis.

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.

The diesel generators (DGs) provide a source of emergency power when offsite power is either unavailable, or is degraded below a point that would allow safe unit operation. Undervoltage protection will generate a loss of power (LOP) DG start if a loss of voltage or degraded voltage condition occurs on the 4.16 kV [kilovolt] vital bus. The proposed technical specification (TS) change affects the voltage at which an emergency bus that is experiencing sustained degraded voltage will disconnect from offsite power and transfer to the DGs. While the TS limits are revised, the function remains the same and will continue to be performed. The first level undervoltage relays (FLUR) and second level undervoltage relays (SLUR) will continue to meet their required function to transfer 4.16 kV buses to the DGs in the event of insufficient offsite power voltage. This transfer will ensure that the class 1E equipment is capable of performing its function to meet the requirements of the accident analysis. The revised TS surveillance requirement (SR) 3.3.5.3 setpoints will not cause unnecessary separation of engineered safety [feature] (ESF) loads from the 230 kV System. The proposed change does not affect any accident initiators or precursors.

The ESF function delay times are bounding input parameters that represent actual plant performance for when these ESF functions can be credited to begin operating after an accident has already occurred. The increased ESF delay times are not physically related to the cause of any accident. Therefore, the increase in ESF delay times do not introduce the possibility of a change in the frequency of an accident previously evaluated. The revised LOCA control room operator and offsite dose analysis results remain within the applicable [General Design Criterion (GDC)] 19–1971 and 10 CFR 100 limits. Therefore, the proposed activity does not result in an increase in the consequence of an accident previously evaluated in the FSARU.

Therefore, the probability or consequences of any accident previously evaluated will not be significantly increased as a result of the proposed change.

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

Response: No.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The revised surveillance requirements will continue to assure equipment reliability such that plant safety is maintained or will be enhanced. An increased ESF delay time is not an initiator of any accident and does not create any new system interactions or failure modes of any structures, systems or components (SSC).

Equipment important to safety will continue to operate as designed. The changes do not result in adverse conditions or result in any increase in the challenges to safety systems. Therefore, operation of the Diablo Canyon Power Plant in accordance with the proposed amendment will not create the possibility of a new or different type of accident from any accident previously evaluated.

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

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

Response: No.

The DGs provide emergency electrical power to the safeguard buses in support of equipment required to mitigate the consequences of design basis accidents and anticipated operational occurrences, including an assumed loss of all offsite power. SR 3.3.5.3 verifies that the LOP DG start instrumentation channels respond to measured parameters within the necessary range and accuracy. The proposed amendment corrects nonconservative values in the TS limits for the degraded voltage protection function. The proposed change to this SR assures that design requirements of the emergency electrical power system continue to be met.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences. The proposed increase in ESF delay times is considered an analysis input change. However, the safety analyses continue to meet all applicable acceptance criteria. The proposed amendment will not otherwise affect the plant protective boundaries, will not cause a release of fission products to the public, nor will it degrade the performance of any other SSCs important to safety.

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

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

Attorney for licensee: Jennifer Post, Esq., Pacific Gas and Electric Company, 77 Beale Street, Room 2496, Mail Code B30A, San Francisco, CA 94105.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: January 25, 2012.

Description of amendment request: The proposed amendment would revise the Diablo Canyon Power Plant, Units 1 and 2, Updated Final Safety Analysis Report Update (UFSAR) Section 4.3.2.2, ''Power Distribution,'' to allow use of the BEACON Power Distribution Monitoring System methodology described in Westinghouse Electric Company LLC (Westinghouse) WCAP–12472–P–A, Addendum 1–A, ''BEACON Core Monitoring and Operations Support System,'' dated January 2000.

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

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

Response: No.

The proposed change is to revise the Updated Final Safety Analysis Report to allow the use of the BEACON code methodology contained in WCAP–12472–P–A, Addendum 1–A. The BEACON code will be used to perform core flux mapping to support the performance of Technical Specification surveillances for power distribution limits and the use of the BEACON code will not cause an accident.

No physical changes are being made to the plant. With the change, Diablo Canyon Power Plant will continue to operate within the power distribution limits contained in the plant Technical Specifications.

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

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

Response: No.

The proposed change does not involve any physical changes to the plant. The BEACON code performs flux mapping of the core and is not used to control the performance of any plant equipment. Therefore, use of the BEACON code cannot cause an accident. If it is determined that the plant is not operating within the power distribution limits during the performance of a Technical Specification Surveillance using BEACON, then the applicable Technical Specification Condition and Required Action(s) will be entered.

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

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

Response: No.

With the use of the BEACON code methodology contained in WCAP–12472–P–A, Addendum 1–A, the plant will continue to operate within the power distribution limits contained in the plant Technical Specifications. The use of the BEACON code does not involve any changes to the fuel, reactor vessel, or containment fission product barriers. The use of the BEACON code methodology includes requirements for control of uncertainties associated with use of the methodology and therefore there will be no impact on the accident analyses that are contained in the Updated Final Safety Analysis Report.

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

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

Attorney for licensee: Jennifer Post, Esq., Pacific Gas and Electric Company, 77 Beale Street, Room 2496, Mail Code B30A, San Francisco, CA 94105.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: March 8, 2012.

Description of amendment request: The proposed amendment would revise (1) Technical Specification (TS) 3.3.7, ''Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation,'' by changing the Allowable Value for the main control room air intake radiation monitoring instrumentation in Table 3.3.7–1 from ≤ 9.45E–05 micro-Curie/cubic centimeter (μCi/cc) (3,308 counts per minute (cpm)) to ≤ 1.647E–04 μCi/cc (3,308 cpm); and (2) TS 3.4.16, ''RCS [reactor coolant system] Specific Activity,'' by lowering the DOSE EQUIVALENT 1–131 spike limit from 21 micro-Curie/gram (μCi/gm) to 14 μCi/gm in Required Action A.1 and Condition C.

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 consequence of an accident previously evaluated?

Response: No.

The proposed TS changes do not adversely affect any fission product barrier nor do they alter the safety function of safety systems, structures, or components, or their roles in accident prevention or mitigation. They do not change any credited operator actions nor do they physically change any plant system, structure, or component. The amount of iodine in the primary coolant and the Allowable Value for the main control room radiation monitors do not affect the initiation of any accident or transient. Therefore, the proposed amendment does not result in a significant increase in the probability of an accident previously evaluated. The changes do not adversely affect the protective and mitigative capabilities of the plant. The SSCs [structures, systems, and components] will continue to perform their intended safety functions. The proposed reduction in the amount of DOSE EQUIVALENT 1–131 (DEI–131) in the reactor coolant following a load transient has no impact on any plant configuration or system performance relied upon to mitigate the consequences of an accident.

The calculated radiological doses remain within the limits prescribed in 10 CFR Part 100 and GDC–19 [General Design Criterion 19 of Appendix A to 10 CFR Part 50] and are consistent with the methodology and acceptance criteria of Section 15.6.3 of NUREG–0800 and Appendix A of Section 15.1.5 of NUREG–0800.

The change to the Allowable Value for the main control room radiation monitors continues to ensure that the monitors are capable of performing their intended design function of isolating the main control room subsequent to an accident.

Therefore, the proposed amendment does not involve a significant increase in the 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 proposed TS changes do not alter the configuration of the plant nor do they directly affect plant operation. The proposed TS changes do not result in the installation of any new equipment or system or the modification of any existing equipment or systems. No new operation procedures, conditions, or modes are created. As a result, the proposed TS changes do not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing basis. There will be no adverse effects or challenges imposed on any safety-related system as a result of these changes.

Therefore, the proposed amendment 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 calculated radiological doses remain within the limits prescribed in 10 CFR Part 100 and GDC–19, and are consistent with the methodology and acceptance criteria of

Section 15.6.3 of NUREG–0800 and Appendix A of Section 15.1.5 of NUREG–0800. The Allowable Value for the main control room radiation monitor continues to ensure that the monitors are capable of performing their intended design function of isolating the main control room subsequent to an accident.

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

The NRC staff has reviewed the licensee's analysis and, based on 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: Stephen J. Campbell.

Notice of Issuance of Amendments to Facility Operating Licenses and Combined 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.

A notice of consideration of issuance of amendment to facility operating license or combined license, as applicable, 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 51.22(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 NRC's Public Document Room (PDR), located at One White Flint North, Room O1–F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available documents created or received at the NRC are accessible electronically through the Agencywide Documents Access and Management System (ADAMS) in the NRC Library at *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's Reference staff at 1–800–397–4209, 301–415–4737 or by email to *pdr.resource@nrc.gov.*

Duke Energy Carolinas, LLC, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of application of amendments: May 6, 2010, as supplemented by letters dated February 11, 2011, April 28, 2011, July 19, 2011, and September 16, 2011.

Brief description of amendments: The amendments revised the Technical Specifications related to supporting operation with 24-month fuel cycles. Specifically, the change would revise the frequency of certain TS Surveillance Requirements (SRs) from ''18 months'' to ''24 months,'' in accordance with the guidance of Generic Letter (GL) 91–04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24–Month Fuel Cycle.''

Date of Issuance: April 20, 2012.

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

Amendment Nos.: Unit 1—379, Unit 2—381, and Unit 3—380.

Renewed Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the licenses and the technical specifications.

*Date of initial notice in **Federal Register**:* September 7, 2010 (75 FR 54394). The supplements dated February 11, 2011, April 28, 2011, July 19, 2011, and September 16, 2011, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 20, 2012.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 11, 2011.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.7.4, ''RCS Leakage Detection Instrumentation,'' to define a new time limit for restoring inoperable Reactor Coolant System (RCS) leakage detection instrumentation to operable status; to establish alternate methods of monitoring RCS leakage when one or more required monitors are inoperable; and to make TS Bases changes which reflect the proposed changes and more accurately reflect the contents of the facility design basis related to operability of the RCS leakage detection instrumentation. These changes are consistent with the guidance contained in NRC-approved Technical Specifications Task Force (TSTF) change traveler TSTF–514, Revision 3, ''Revise BWR [Boiling-Water Reactor] Operability Requirements and Actions for RCS Leakage Instrumentation.'' The NRC announced the availability of this TS improvement in the **Federal Register** on December 17, 2010 (75 FR 79048), as part of the consolidated line item improvement process.

Date of issuance: April 23, 2012.

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

Amendment No.: 224.

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

*Date of initial notice in **Federal Register**:* May 31, 2011 (76 FR 31373).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 23, 2012.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 16, 2011.

Brief description of amendment: The amendment revises the Inservice Testing Program, Technical Specification (TS) 5.5.6 for IP2 and TS 5.5.7 for IP3.

Date of issuance: May 2, 2012.

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

Amendment No.: 267 and 245.

Facility Operating License Nos. DPR–26 and DPR–64: The amendment revised the License and the TSs.

Date of initial notice in ***Federal Register****:* December 27, 2011 (76 FR 80976).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 2, 2012.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–255, Palisades Nuclear Plant, Van Buren County, Michigan

Date of application for amendment: April 6, 2011, supplemented by letter dated October 28, 2011.

Brief description of amendment: The amendment revised Technical Specification 5.5.14, ''Containment Leak Rate Testing Program,'' by replacing the reference to RG 1.163, ''Performance-Based Containment Leak-Test Program,'' with a reference to Topical Report NEI 94–01, Revision 2–A, ''Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,'' as the implementation document for the 10 CFR Part 50, Appendix J, Option B, performance-based containment leak rate testing program at the Palisades Nuclear Plant (PNP). This amendment allows PNP to extend its performance-based containment integrated leakage rate test (ILRT, or Type A test) interval up to 15 years. Accordingly, the licensee has also requested to extend its current Type A test interval from the current one-time approved 11.25 years to 15 years so that the next Type A test can be conducted by May 3, 2016, instead of the current due date of August 3, 2012.

Date of issuance: April 23, 2012.

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

Amendment No.: 247.

Facility Operating License No. DPR–20: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register****:* June 14, 2011, (76 FR 34766).

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 April 23, 2012.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: October 28, 2011, as supplemented by letter dated January 26, 2012.

Brief description of amendment: The amendment increased the numeric values of the Safety Limit Minimum Critical Power Ratio in Technical Specification Section 2.1.1.2 from 1.09 to 1.11 for two recirculation loop operation (TLO) and from 1.12 to 1.14 for single recirculation loop operation (SLO). The Minimum Critical Power Ratio Safety Limit values for both TLO and SLO are determined in accordance with the requirements set forth in NRC-approved General Electric Company (GE) licensing topical report NEDC–33173P, ''Applicability of GE Methods to Expanded Operating Domains,'' Revision 0, February 2006.

Date of issuance: April 20, 2012.

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

Amendment No.: 189.

Facility Operating License No. NPF–29: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in ***Federal Register****:* February 14, 2012 (77 FR 8291).

The supplemental letter dated January 26, 2012, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 20, 2012.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: April 13, 2011.

Brief description of amendment: The amendment modified the Technical Specifications (TSs) as a result of a revised Fuel Handling Accident analysis. The new analysis determined that the current TSs may not be conservative for all scenarios. The amendment provides new applicability and/or action language in the TSs that includes load movements over irradiated fuel assemblies. Specifically, the amendment modified the following TSs: TS 3.3.3.1 (Radiation Monitoring Instrumentation); TS 3.7.6.1 (Control Room Emergency Air Filtration System); TS 3.7.6.3 (Control Room Air Temperature—Operating); TS 3.7.6.4 (Control Room Air Temperature—Shutdown); TS 3.8.1.2 (A.C. [Alternating Current] Sources—Shutdown); TS 3.8.2.2 (D.C. [Direct Current] Sources—Shutdown); TS 3.8.3.2 (Onsite Power Distribution—Shutdown); TS 3.9.3 (Decay Time); TS 3.9.4 (Containment Building Penetrations); and TS 3.9.7 (Crane Travel—Fuel Handling Building).

Date of issuance: April 25, 2012.

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

Amendment No.: 235.

Facility Operating License No. NPF–38: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in ***Federal Register****:* August 23, 2011 (76 FR 52701).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 25, 2012.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 14, 2011, as supplemented by letters dated September 2, 2011, and November 18, 2011.

Brief description of amendment: The license amendment request changes the facility operating licenses and the Technical Specifications (TSs) 3.4.12–1, for the Braidwood Station, Units 1 and 2 and Byron Station, Units 1 and 2. The proposed change will reflect standard wording incorporated in NUREG–1431, Revision 3, ''Standard Technical Specifications—Westinghouse Plants,'' for plants with installed bypass test capability. The proposed change is needed to support utilization of bypass test capability that is planned to be installed, which will reduce the potential for unnecessary reactor trips or safeguards actuation due to a failure or transient in a redundant channel.

Date of issuance: March 30, 2012.

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

Amendment Nos.: Braidwood Unit 1—169; Braidwood Unit 2—169; Byron Unit 1—176 and Byron Unit 2—176.

Facility Operating License Nos. NPF–72, NPF–77, NPF–37, and NPF–66: The amendments revised the Technical Specifications and License.

Date of initial notice in ***Federal Register****:* August 16, 2011 (76 FR 50759).

The September 2, 2011, and November 18, 2011, supplements contained clarifying information and did not change the staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 30, 2012.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 6, 2011.

Brief description of amendments: The amendment modifies the actions to be taken when the atmospheric gaseous radioactivity monitor is the only operable reactor coolant leakage detection instrument. The modified actions require additional, more frequent monitoring of other indications of Reactor Coolant System (RCS) leakage and provide appropriate time to restore another leakage detection instrument to operable status. This change is consistent with the U.S. Nuclear Regulatory Commission (NRC) approved safety evaluation on Technical Specification Task Force (TSTF) Traveler, TSTF–514–A, Revision 3, ''Revised BWR [boiling-water reactor] Operability Requirements and Actions for RCS Leakage Instrumentation'' dated November 24, 2010.

Date of issuance: April 23, 2012.

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

Amendment Nos.: 283 and 286.

Renewed Facility Operating License Nos. DPR–44 and DPR–56: Amendments revised the License and Technical Specifications.

Date of initial notice in ***Federal Register:*** September 6, 2011, (76 FR 55128).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 23, 2012.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of application for amendment: March 8, 2012, as supplemented by letters dated March 23, March 29, and April 2, 2012.

Brief description of amendment: The amendment allows an extension of 24 hours to the Completion Time for Condition C in the Susquehanna Steam Electric Station (SSES) Unit 2 Technical Specification (TS) 3.8.7, ''Distribution Systems—Operating,'' to allow a Unit 1 4160 V subsystem to be de-energized and removed from service for 96 hours to perform modifications on the bus. It also allows an extension of 24 hours to the Completion Time for Condition A in SSES Unit 2 TS 3.7.1, ''Plant Systems—RHRSW [residual heat removal service water system] and UHS [ultimate heat sink],'' to allow the UHS spray array and spray array bypass valves associated with applicable division RHRSW, and in Condition B, the applicable division Unit 2 RHRSW subsystem, to be inoperable for 96 hours during the Unit 1 4160 V bus breaker control logic modifications.

Date of issuance: April 19, 2012.

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

Amendment No.: 258.

Facility Operating License No. NPF–22: This amendment revised the License and Technical Specifications.

Date of initial notice in ***Federal Register***: March 16, 2012 (77 FR 15814).

The supplements dated March 23, March 29, and April 2, 2012, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 19, 2012, which also contains its final no significant hazards consideration determination.

No significant hazards consideration comments received: No.

South Carolina Electric and Gas Company, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit 1, Jenkinsville, South Carolina

Date of application for amendment: October 12, 2011, as supplemented by letter dated April 5, 2012.

Brief description of amendment: This amendment revised the Virgil C. Summer Nuclear Station (VCSNS) Technical Specification to allow a one-time extension of the 10-year interval for the containment integrated leakage rate test such that the existing test interval would be extended from 120 months to 130 months.

Date of Issuance: May 1, 2012.

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

Amendment No: 189.

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

Date of initial notice in ***Federal Register***: December 13, 2011 (76 FR 77571).

The licensee's supplemental letter contained clarifying information, did not change the scope of the original license amendment request, did not change the NRC staff's initial proposed finding of 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 May 1, 2012.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–259, Browns Ferry Nuclear Plant, Unit 1, Limestone County, Alabama

Date of application for amendment: April 16, 2010, as supplemented by letters dated February 23, May 12, October 7, 2011, and April 18, 2012 (TS–473).

Brief description of amendment: The licensee proposes to transition Unit 1 to AREVA fuel. To support the transition to AREVA fuel, the proposed amendment adds the AREVA NP analysis methodologies to the list of approved methods to be used in determining the core operating limits in the core operating limits report. Additional technical specification changes are requested to reflect the AREVA NP specific methods for monitoring and enforcing of the thermal limits. The licensee's request is for non-extended power uprate conditions (i.e., 105 percent of Original Licensed Thermal Power level) only.

Date of issuance: April 27, 2012.

Effective date: This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

Amendment No.: 281.

Renewed Facility Operating License No. DPR–33: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: January 10, 2011 (76 FR 1467). The supplemental letters provided clarifying information that did not expand the scope of the original application or change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 27, 2012.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 3rd day of May 2012.

For the Nuclear Regulatory Commission.

Michele G. Evans,

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

[FR Doc. 2012–11599 Filed 5–14–12; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards (ACRS); Meeting of the ACRS Subcommittee on Fukushima; Notice of Meeting

The ACRS Subcommittee on Fukushima will hold a meeting on May 22–23, 2012, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Tuesday, May 22, 2012—1:00 p.m. Until 5:00 p.m.; Wednesday, May 23, 2012—8:30 a.m. Until 5:00 p.m.

The Subcommittee will review and discuss the staff's plans for implementation of the Near-Term Task Force Tier 3 Recommendations. The Subcommittee will hear presentations by and hold discussions with the NRC staff and other interested persons regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the Full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official (DFO), Antonio Dias (Telephone 301–415–6805 or Email: *Antonio.Dias@nrc.gov*) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Thirty-five hard copies of each presentation or handout should be provided to the DFO thirty minutes before the meeting. In addition, one electronic copy of each presentation should be emailed to the DFO one day before the meeting. If an electronic copy cannot be provided within this timeframe, presenters should provide the DFO with a CD containing each presentation at least thirty minutes before the meeting. Electronic recordings will be permitted only during those portions of the meeting that are open to the public. Detailed procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 17, 2011, (76 FR 64126–64127).

Detailed meeting agendas and meeting transcripts are available on the NRC Web site at *http://www.nrc.gov/reading-rm/doc-collections/acrs.* Information regarding topics to be discussed, changes to the agenda, whether the meeting has been canceled or rescheduled, and the time allotted to present oral statements can be obtained from the Web site cited above or by contacting the identified DFO. Moreover, in view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with these references if such rescheduling would result in a major inconvenience.

If attending this meeting, please enter through the One White Flint North building, 11555 Rockville Pike Rockville, MD. After registering with security, please contact Mr. Theron Brown (Telephone 240–888–9835) to be escorted to the meeting room.

Dated: May 8, 2012.

Cayetano Santos,

Chief, Technical Support Branch, Advisory Committee on Reactor Safeguards.

[FR Doc. 2012–11714 Filed 5–14–12; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee On Reactor Safeguards (ACRS) Meeting of the ACRS Subcommittee on Power Uprates; Notice of Meeting

The ACRS Subcommittee on Power Uprates will hold a meeting on May 24, 2012, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The meeting will be open to public attendance, with the exception of portions that may be closed to protect information that is proprietary pursuant to 5 U.S.C. 552b(c)(4).

The agenda for the subject meeting shall be as follows:

Thursday, May 24, 2012—8:30 a.m. Until 5 p.m.

The Subcommittee will review the Safety Evaluation Report (SER) associated with the Grand Gulf Nuclear Station Unit 1 extended power uprate application. The Subcommittee will hear presentations by and hold discussions with the NRC staff, the licensee (Entergy Operations, Inc.), and other interested persons regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the Full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official (DFO), John Lai (Telephone 301–415–5197 or Email: *John.Lai@nrc.gov*) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Thirty-five hard copies of each presentation or handout should be provided to the DFO thirty minutes before the meeting. In addition, one electronic copy of each presentation should be emailed to the DFO one day before the meeting. If an electronic copy cannot be provided within this timeframe, presenters should provide the DFO with a CD containing each presentation at least thirty minutes before the meeting. Electronic recordings will be permitted only during those portions of the meeting that are open to the public. Detailed procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 17, 2011, (76 FR 64126–64127).

Detailed meeting agendas and meeting transcripts are available on the NRC Web site at *http://www.nrc.gov/reading-rm/doc-collections/acrs.* Information regarding topics to be discussed, changes to the agenda, whether the meeting has been canceled or rescheduled, and the time allotted to present oral statements can be obtained from the Web site cited above or by contacting the identified DFO. Moreover, in view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with these references if such rescheduling would result in a major inconvenience.

If attending this meeting, please enter through the One White Flint North building, 11555 Rockville Pike, Rockville, MD. After registering with