7590-01-P

NUCLEAR REGULATORY COMMISSION

NRC-2009-0564

NOTICE

APPLICATIONS AND AMENDMENTS TO FACILITY OPERATING LICENSES INVOLVING PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATIONS AND CONTAINING SENSITIVE UNCLASSIFIED NON-SAFEGUARDS INFORMATION AND ORDER IMPOSING PROCEDURES FOR ACCESS TO SENSITIVE UNCLASSIFIED NON-SAFEGUARDS INFORMATION

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 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 notice includes notices of amendments containing sensitive unclassified non-safeguards information (SUNSI).

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 Title 10 of the *Code of Federal Regulations* (10 CFR), Section 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.

Written comments may be submitted by mail to the Chief, Rulemaking and Directives Branch (RDB), TWB-05-B01M, 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 faxed to the RDB at 301-492-3446. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland.

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. 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 Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, or at http://www.nrc.gov/reading-rm/doc-collections/cfr/part002/part002-0309.html. 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.html. 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 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, 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.

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 ten (10) days prior to the filing deadline, the participant should contact the Office of the Secretary by e-mail at *hearing.docket@nrc.gov*, or by telephone at (301) 415-1677, to request (1) a digital 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 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 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 EIE, users will be required to install a Web browser plug-in from the NRC 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 NRC guidance available on the NRC 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 e-mail notice confirming receipt of the document. The E-Filing system also distributes an e-mail notice that provides access to the document to the NRC 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 e-mail at *MSHD.Resource@nrc.gov*, or by a toll-free call at (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 NRC's electronic hearing docket which is available to the public at *http://ehd.nrc.gov/EHD_Proceeding/home.asp*, 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 **January 5, 2010.** Non-timely 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 amendment 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 electronically 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.resource@nrc.gov*.

<u>Carolina Power & Light Company, Docket No. 50-261, H. B. Robinson Steam Electric Plant (HBRSEP) Unit No. 2, Darlington County, South Carolina</u>

<u>Date of amendment request</u>: June 19, 2009, as supplemented by letter dated October 20, 2009.

<u>Description of amendment request</u>: **This amendment request contains sensitive unclassified non-safeguards information (SUNSI).** The proposed amendment would revise TS 3.3.1, "Reactor Protection System Instrumentation." The proposed change revises the requirements related to the reactor protection system interlock for the turbine trip input to the reactor protection system.

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

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

   The proposed change provides revised requirements for the reactor protection system interlock associated with the turbine trip protection function. The proposed change will allow the interlock for turbine trip function to be raised from the current interlock setting of nominally 10 percent reactor power to nominally 40 percent reactor power.

This change will allow the reactor to continue operating safely at power levels up to nominally 40 percent when the turbine is not operating. The applicable accident analyses, as described in the HBRSEP, Unit No. 2, Updated Final Safety Analysis Report (UFSAR) have been reviewed. The turbine trip input to reactor trip has been verified to be either not used in the accident analyses or that the change does not adversely affect the analyses results and conclusion. Therefore, it is concluded that the consequences as described in the UFSAR accident analyses are unaffected by the proposed change.

An analysis of plant response to a turbine trip at nominally 40 percent power provided with the amendment request shows that the applicable acceptance criteria are met. Specifically, analysis has shown that a turbine trip without a reactor trip below 40 percent power does not challenge the pressurizer PORVs [power operated relief valves] or the steam generator safety valves; thereby, not adversely affecting the probability of a small break LOCA [loss of coolant accident] due to a stuck open PORV, or an excessive cooldown event due to a stuck open steam generator safety valve. As a result, the probability of any accident previously evaluated is not significantly increased by the proposed changes.

Therefore, operation of the facility in accordance with the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

As described above, the proposed change provides revised requirements for the reactor protection system interlock associated with the turbine trip protection function. The proposed change will allow the interlock for turbine trip function to be raised from the current interlock setting of nominally 10 percent reactor power to nominally 40 percent reactor power.

No new accident initiators or precursors are introduced by the proposed change. Changing the interlock for the reactor trip on turbine trip from P-7 to P-8 changes the power level associated with enabling and disabling the reactor trip on turbine trip function. The turbine pressure input to the reactor protection system permissive is not an accident initiator. The change does not affect how the associated trip functional units operate or function. The changes do not create the possibility of a new or different kind of accident from any previously evaluated because these interlock changes do not affect the way that the associated trip functional units operate or function.

Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated.

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

As described above, the proposed change provides revised requirements for the reactor protection system interlock associated with the turbine trip protection function. The proposed change will allow the interlock for the turbine trip function to be raised from the current interlock setting of nominally 10 percent reactor power to nominally 40 percent reactor power.

Also, as previously described, this change will allow the reactor to continue operating safely at power levels up to nominally 40 percent when the turbine is not operating. The applicable UFSAR accident analyses have been reviewed and it is concluded that the accident analyses are unaffected by the proposed change. An analysis of plant response to a turbine trip at nominally 40 percent power shows that the applicable acceptance criteria are met. Based on these evaluations, the margins of safety that could potentially have been impacted by the proposed change associated with the reactor, which include departure from nucleate boiling (DNB) and fuel temperature margins, and the margin of safety associated with reactor coolant system integrity, are not affected.

Therefore, operation of the facility in accordance with the proposed amendment would 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: 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: Thomas H. Boyce.

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50-271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: October 27, 2009.

Description of amendment request: **This amendment request contains sensitive unclassified non-safeguards information (SUNSI).** This amendment request would change the Technical Specifications to provide revised values for the Safety Limit Minimum Critical Power Ratio (SLMCPR) for both single and dual recirculation loop operation.

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

1. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The basis of the Safety Limit MCPR (SLMCPR) is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPR values preserve the existing margin to transition boiling and probability of fuel damage is not increased. The derivation of the revised SLMCPR for Vermont Yankee for incorporation into the Technical Specifications and its use to determine plant and cycle-specific thermal limits has been performed using NRC approved methods. These plant-specific calculations are performed each operating cycle and if necessary, will require future changes to these values based upon revised core designs. The revised SLMCPR values do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

   Based on the above, Vermont Yankee has concluded that the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes result only from a specific analysis for the Vermont Yankee core reload design. These changes do not involve any new or different methods for operating the facility. No new initiating events or transients result from these changes.

   Based on the above, Vermont Yankee has concluded that the proposed change will not create the possibility of a new or different kind of accident from those previously evaluated.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

   The new SLMCPR is calculated using NRC approved methods with plant and cycle specific parameters for the current core design. The SLMCPR value remains conservative enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. The operating MCPR limit is set appropriately above the safety limit value to ensure adequate margin when the cycle specific transients are evaluated. Accordingly, the margin of safety is maintained with the revised values.

   As a result, Vermont Yankee has determined that 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: Mr. William C. Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 400 Hamilton Avenue, White Plains, NY 10601.

NRC Branch Chief: Nancy L. Salgado.

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 amendment request: October 27, 2009.

Description of amendment request: **This amendment request contains sensitive unclassified non-safeguards information (SUNSI).** The proposed amendment revises the Technical Specifications to increase the two recirculation loop minimum critical power ratio

(MCPR) safety limit from 1.08 to 1.09 and the single recirculation loop MCPR safety limit from 1.10 to 1.12.

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

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

   Response: No.

   The Minimum Critical Power Ratio (MCPR) limit is defined in the Bases to Technical Specification 2.1.1.2 as that limit, "that, in the event of an AOO [(Anticipated Operational Occurrence)] from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition." The MCPR safety limit satisfies the requirements of General Design Criterion 10 of Appendix A to 10CFR50 regarding acceptable fuel design limits. The MCPR safety limit is reevaluated for each reload using NRC-approved methodologies. The analyses for GGNS [Grand Gulf Nuclear Station] Cycle 18 have concluded that a two-loop MCPR safety limit of 1.09, based on the application of Global Nuclear Fuels' NRC approved MCPR safety limit methodology, will ensure that this acceptance criterion is met. For single-loop operation, a MCPR safety limit of 1.12, also ensures that this acceptance criterion is met. The MCPR operating limits are presented and controlled in accordance with the GGNS Core Operating Limits Report (COLR).

   The requested Technical Specification changes do not involve any plant modifications or operational changes that could affect system reliability or performance or that could affect the probability of operator error. The requested changes do not affect any postulated accident precursors, do not affect any accident mitigating systems, and do not introduce any new accident initiation mechanisms.

   Therefore, the changes to the Minimum Critical Power Ratio safety limit do not involve a significant increase in the probability or consequences of any 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 GNF2 fuel to be used in Cycle 18 is of a design compatible with the co-resident GE14 and ATRIUM-10 fuel. Therefore, the introduction of

GNF2 fuel into the Cycle 18 core will not create the possibility of a new or different kind of accident. The proposed changes do not involve any new modes of operation, any changes to setpoints, or any plant modifications. The proposed revised MCPR safety limits have accounted for the mixed fuel core and have been shown to be acceptable for Cycle 18 operation. Compliance with the criterion for incipient boiling transition continues to be ensured. The core operating limits will continue to be developed using NRC approved methods which also account for the mixed fuel core design. The proposed MCPR safety limits or methods for establishing the core operating limits do not result in the creation of any new precursors to an accident.

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

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

Response: No.

The MCPR safety limits have been evaluated in accordance with Global Nuclear Fuels NRC-approved cycle-specific safety limit methodology to ensure that during normal operation and during AOO's at least 99.9% of the fuel rods in the core are not expected to experience transition boiling. The proposed revised MCPR safety limits have accounted for the mixed fuel core and have been shown to be acceptable for Cycle 18 operation. Compliance with the criterion for incipient boiling transition continues to be ensured. On this basis, the implementation of the change to the MCPR safety limits does not involve 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: Terence A. Burke, Associate General Counsel - Nuclear Entergy Services, Inc., 1340 Echelon Parkway, Jackson, Mississippi 39213.

NRC Branch Chief: Michael T. Markley.

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 amendment request: November 3, 2009.

Description of amendment request: **This amendment request contains sensitive unclassified non-safeguards information (SUNSI).** The proposed amendment revises the Technical Specifications (TSs) to reflect the installation of the digital General Electric - Hitachi Nuclear Measurement Analysis and Control (NUMAC) Power Range Neutron Monitoring (PRNM) System.

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

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

   Response: No.

   The probability (frequency of occurrence) of design basis accidents (DBAs) occurring is not affected by the NUMAC PRNM System, since the system does not interact with equipment whose failure could cause an accident. Compliance with the regulatory criteria established for plant equipment are maintained with the installation of the upgraded NUMAC PRNM System. Scram setpoints in the NUMAC PRNM System are established such that the analytical limits are met.

   The unavailability of the new NUMAC PRNM System is equal to or less than the existing system and, as a result, the scram reliability is equal to or better than the existing analog power system. No new challenges to safety-related equipment result from the NUMAC PRNM System modification. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

   The proposed change replaces the current Option E-I-A stability solution with an NRC-approved Option III long-term stability solution. The NUMAC PRNM hardware incorporates the Oscillation Power Range Monitor (OPRM) Option III detect-and-suppress solution, which has been previously reviewed and approved by the NRC. The OPRM meets

[10 CFR Part 50, Appendix A] General Design Criterion (GDC) 10, *Reactor Design*, and GDC 12, *Suppression of Reactor Power Oscillations*, requirements by automatically detecting and suppressing design basis thermal-hydraulic oscillations prior to exceeding the fuel Minimum Critical Power Ratio (MCPR) Safety Limit.

Based on the above, installation of the new NUMAC PRNM System with the OPRM Option III stability solution integrated into the NUMAC PRNM equipment does not 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.

The components of the NUMAC PRNM System are equivalent or of better design and qualification criteria than those currently installed and utilized in the plant. No new operating mode, safety-related equipment lineup, accident scenario, or interaction mode not reviewed and approved as part of the design and licensing of the NUMAC PRNM System has been identified. Therefore, the NUMAC PRNM System retrofit does not adversely affect plant equipment.

The new NUMAC PRNM System uses digital equipment that has software-controlled digital processing compared to the existing power range system that uses mostly analog and discrete component processing. Specific failures of hardware and potential software common-cause failures are different from the existing system. The effects of potential software common-cause failure are mitigated by specific hardware design and system architecture as discussed in Section 6.0 of [GE Nuclear Energy Licensing Topical Report] NEDC-32410P-A. Failure(s) of the system have the same overall effect as the present design. No new or different kinds of accidents are introduced. Therefore, the NUMAC PRNM System does not adversely effect plant equipment.

The currently installed Average Power Range Monitoring (APRM) system is replaced with a NUMAC PRNM System that performs the existing power range monitoring functions and adds an OPRM to react automatically to potential reactor thermal-hydraulic instabilities.

Based on the above, 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 amendment involve a significant reduction in a margin of safety?

Response: No.

> The proposed TS changes associated with the NUMAC PRNM System retrofit implement the constraints of the NUMAC PRNM System design and related stability analyses. The NUMAC PRNM System change does not impact reactor operating parameters or the functional requirements of the APRM system. The replacement equipment continues to provide information, enforce control rod blocks, and initiate reactor scrams under appropriate specified conditions. The proposed change does not reduce safety margins. The replacement APRM equipment has improved channel trip accuracy compared to the current analog system, and meets or exceeds system requirements previously assumed in setpoint analysis. Thus, the ability of the new equipment to enforce compliance with margins of safety equals or exceeds the ability of the equipment which it replaces.
>
> Therefore, the proposed changes do not involve a reduction in a margin of safety.

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

Attorney for licensee: Terence A. Burke, Associate General Counsel - Nuclear Entergy Services, Inc., 1340 Echelon Parkway, Jackson, Mississippi 39213.

NRC Branch Chief: Michael T. Markley.

Exelon Generation Company, LLC, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: October 5, 2009.

Description of amendment request: **This amendment request contains sensitive unclassified non-safeguards information (SUNSI).** The amendment(s) would revise Technical Specification (TS) 4.3.1, “Criticality,” to address a non-conservative TS. The proposed change addresses the Boraflex degradation issue in the LaSalle County Station (LSCS) Unit 2 spent fuel storage racks by revising TS Section 4.3.1 to allow the use of NETCO-

SNAP-IN® rack inserts in LSCS Unit 2 spent fuel storage rack cells as a replacement for the neutron absorbing properties of the existing Boraflex panels.

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 adds an additional requirement to TS Section 4.3.1 to install a NETCO-SNAP-IN® rack insert in spent fuel storage rack cells that cannot otherwise maintain the requirements of TS Section 4.3.1.1.a to ensure that the effective neutron multiplication factor, K_{eff}, is less than or equal to 0.95, if the spent fuel pool (SFP) is fully flooded with unborated water. The proposed change also includes a revision to TS Section 4.3.1 to specify the bounding reactivity fuel design allowed for storage in the Unit 1 and Unit 2 SFPs. Since the proposed change pertains only to the SFP, only those accidents that are related to movement and storage of fuel assemblies in the SFP could be potentially affected by the proposed change.

   The current licensing basis for the LSCS Unit 2 SFP credits the neutron absorbing properties of the Boraflex neutron poison material in the spent fuel storage racks. The current licensing basis demonstrates: (1) adequate margin to criticality for spent fuel storage rack cells that credit the neutron absorption capabilities of Boraflex, (2) adequate margin for fuel assemblies inadvertently placed into locations adjacent to the spent fuel storage racks, and (3) adequate margin for assemblies accidentally dropped onto the spent fuel storage racks. Therefore, the probability that a misplaced fuel assembly would result in an inadvertent criticality is unchanged since the process and procedural controls governing fuel movement in the SFP will not be changed. The dose consequences of the most limiting drop of a fuel assembly in the SFP is limited by the number of the fuel rods damaged and other engineered features unaffected by the proposed change, including the fuel design, fuel decay time, water level in the SFP, water temperature of the SFP, and the engineering features of the Reactor Building Ventilation System.

   The installation of NETCO-SNAP-IN® rack inserts does not result in a significant increase in the probability of an accident previously analyzed. The revised criticality analysis takes no credit for the Boraflex material. The use of a rack insert provides an alternative neutron absorber to take the place of the degraded Boraflex material, without removal of the existing Boraflex. The probability that a fuel assembly would be dropped is unchanged by the installation of the NETCO-SNAP-IN® rack inserts. These events involve failures of administrative controls, human performance, and equipment failures that are unaffected by the presence or absence of Boraflex and the rack inserts.

The installation of NETCO-SNAP-IN® rack inserts does not result in a significant increase in the consequence of an accident previously analyzed. A criticality analysis has been prepared to demonstrate adequate margin to criticality for spent fuel storage rack cells with rack inserts in the LSCS Unit 2 SFP, and adequate criticality margin for assemblies accidentally dropped onto the spent fuel storage racks.

The installation of NETCO-SNAP-IN@ rack inserts does not affect the consequences of a dropped fuel assembly. The consequences of dropping a fuel assembly onto any other fuel assembly or other structure are unaffected by the change. The consequences of dropping a fuel assembly onto a spent fuel storage rack cell with a rack insert are bounded by the event of dropping an assembly onto another assembly, both for criticality and for radiological consequences. For criticality, the effects on K_{eff} of dropping a fuel assembly have been evaluated and are acceptable. For radiological consequences, the number of rods damaged when a fuel assembly is accidentally dropped onto a spent fuel storage rack cell with or without a rack insert is bounded by the number of rods damaged by an assembly dropped onto another assembly. The change does not affect the effectiveness of the other engineered design features to limit the offsite dose consequences of the limiting fuel assembly drop accident.

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.

Onsite storage of spent fuel assemblies in the SFP is a normal activity for which LSCS has been designed and licensed. As part of assuring that this normal activity can be performed without endangering public health and safety, the ability to safely accommodate different possible accidents in the SFP, such as dropping a fuel assembly or misloading a fuel assembly, have been analyzed. The proposed spent fuel storage configuration does not change the methods of fuel movement or spent fuel storage. The proposed change allows for continued use of spent fuel storage rack cells that have been determined unusable based on the degradation of Boraflex within those spent fuel storage rack cells. The rack inserts are passive devices. These devices, when inside a spent fuel storage rack cell, perform the same function as the Boraflex in that cell without the potential for degradation. These devices do not add any limiting structural loads or affect the removal of decay heat from the assemblies. No change in total heat load in the SFP is being made. The devices are resistant to corrosion and will maintain their structural integrity over the life of the SFP. An accidental fuel assembly drop does not challenge their structural integrity. The existing fuel handling accident, which assumes the drop of a fuel assembly, bounds the drop of a rack insert and/or rack insert installation tool. This change does not create the possibility of a misloaded assembly into a spent fuel storage rack cell.

The misloading of a more reactive assembly targeted for placement in the LSCS Unit 1 SFP or the LSCS Unit 2 SFP Boraflex region in a rack insert region of the LSCS Unit 2 SFP has been prevented since the most reactive fuel assembly at LSCS is bounded by the rack insert criticality analysis, and the most reactive fuel assembly allowed for future insertion in either the Unit 1 or Unit 2 SFP is being limited to the reference bounding ATRIUM-10 fuel assembly.

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.

LSCS TS 4.3.1.1 requires the spent fuel storage racks to maintain the effective neutron multiplication factor, K_{eff}, less than or equal to 0.95 when fully flooded with unborated water, which includes an allowance for uncertainties. Therefore, for criticality, the required safety margin is 5% including a conservative margin to account for engineering and manufacturing uncertainties.

The proposed change provides an alternative method to ensure that K_{eff} continues to be less than or equal to 0.95, thus preserving the required safety margin of 5%. The criticality analysis demonstrates the required margin to criticality of 5%, including a conservative margin to account for engineering and manufacturing uncertainties, is maintained assuming an infinite array of fuel with all fuel at the peak reactivity. In addition, the margin of safety for radiological consequences of a dropped fuel assembly are unchanged because the event involving a dropped fuel assembly onto a spent fuel storage rack cell containing a fuel assembly with a rack insert is bounded by the consequences of a dropped fuel assembly without a rack insert. The proposed change also maintains the capacity of the Unit 2 SFP to be no more than 4078 fuel assemblies.

> 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. Bradley J. Fewell, Associate General Counsel, Exelon Nuclear, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Stephen J. Campbell.

PSEG Nuclear LLC, Docket No. 50-272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of amendment request: October 8, 2009.

Description of amendment request: **This amendment request contains sensitive unclassified non-safeguards information (SUNSI).** The proposed amendment would revise Technical Specification (TS) 6.8.4.i, "Steam Generator (SG) Program," by adding a one-time alternate repair criterion that excludes certain portions of the tube below the top of the SG tubesheet from periodic SG tube inspections. In addition, the proposed amendment would revise TS 6.9.10, "Steam Generator Tube Inspection Report," to provide reporting requirements specific to the alternate repair criteria. The proposed amendment is supported by Westinghouse Electric Company, LLC Topical Report WCAP-17071-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)." H* (pronounced "H star") is the length of hydraulically expanded SG tube that must remain intact within the tubesheet in order for the joint to resist pullout and leakage due to normal operating and accident conditions.

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

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

   The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that alters the steam generator (SG) inspection and reporting criteria does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident.

   Of the applicable accidents previously evaluated, the limiting transients with consideration to the proposed change to the SG tube inspection and repair criteria are the steam generator tube rupture (SGTR) event, the steam line break (SLB), and the feed line break (FLB) postulated accidents.

   During the SGTR event, the required structural integrity margins of the SG tubes and the tube-to-tubesheet joint over the H* distance will be maintained. Tube rupture in tubes with cracks within the tubesheet is precluded by the constraint provided by the presence of the tubesheet and the tube-to-tubesheet joint. Tube burst cannot occur within the thickness of the tubesheet. The tube-to-tubesheet joint constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet, and from the differential pressure between the primary and secondary side, and tubesheet rotation. Based on this design, the structural margins against burst, as discussed in Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR [pressurized-water reactor] Steam Generator Tubes," and Technical Specification 6.8.4.i, are maintained for both normal and postulated accident conditions.

   The proposed change has no impact on the structural or leakage integrity of the portion of the tube outside of the tubesheet. The proposed change maintains structural and leakage integrity of the SG tubes consistent with the performance criteria of Technical Specification 6.8.4.i. Therefore, the proposed change results in no significant increase in the probability of the occurrence of a SGTR accident.

   At normal operating pressures, leakage from tube degradation below the proposed limited inspection depth is limited by the tube-to-tubesheet crevice. Consequently, negligible normal operating leakage is expected from degradation below the inspected depth within the tubesheet region.

The consequences of an SGTR event are not affected by the primary-to-secondary leakage flow during the event as primary-to-secondary leakage flow through a postulated tube that has been pulled out of the tubesheet is essentially equivalent to a severed tube. Therefore, the proposed change does not result in a significant increase in the consequences of a SGTR[.]

The probability of a SLB is unaffected by the potential failure of a steam generator tube as the failure of tube is not an initiator for a SLB event.

The leakage factor of 2.16 for Salem Unit 1, for a postulated SLB/FLB, has been calculated as shown in Table 9-7 of WCAP-17071-P as revised by the response to RAI [request for additional information] 24 (Attachment 7 [to the application dated October 8, 2009]). Through application of the limited tubesheet inspection scope, the existing operating leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur. The accident analysis calculations have an assumption of 0.6 [gallons per minute (gpm)] at room temperature (gpmRT) primary-to-secondary leakage in a single SG and 1 gpm at room temperature (gpmRT) total primary-to-secondary leakage for all SGs. This apportioned primary-to-secondary leakage is used in the Main Steam Line Break and Locked Rotor accidents. Primary-to-secondary leakage of 1 gpm at room temperature (gpmRT) in a single SG is used in the Control Rod Ejection (CRE) accident.

No leakage factor will be applied to the locked rotor or control rod ejection transients due to their short duration.

The TS operational leak rate limit is 150 gallons per day (gpd) (0.104 gpmRT). The maximum accident leak rate ratio for Salem Unit 1 is 2.16. Consequently, this results in significant margin between the conservatively estimated accident leakage and the allowable accident leakage.

For the condition monitoring (CM) assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 2.16 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 2.16 and compared to the observed operational leakage.

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

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

The proposed change that alters the steam generator inspection and reporting criteria does not introduce any new equipment, create new failure modes for existing equipment, or create any new limiting single failures. Plant operation will not be altered, and all safety functions will continue to perform as previously assumed in accident analyses. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed changes do not involve a significant reduction in the margin of safety.

The proposed change defines the safety significant portion of the tube that must be inspected and repaired (plugged). WCAP-17071-P identifies the specific inspection depth below which any type tube degradation shown to have no impact on the performance criteria in [Nuclear Energy Institute (NEI) document] NEI 97-06 [Revision] 2, "Steam Generator Program Guidelines."

The proposed change that alters the steam generator inspection and reporting criteria maintains the required structural margins of the SG tubes for both normal and accident conditions. Nuclear Energy Institute 97-06, "Steam Generator Program Guidelines," and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," are used as the bases in the development of the limited hot leg tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting General Design Criteria (GDC) 14, "Reactor Coolant Pressure Boundary," GDC 15, "Reactor Coolant System Design," GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," and GDC 32, "Inspection of Reactor Coolant Pressure Boundary," by reducing the probability and consequences of a SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation, the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, Westinghouse WCAP-17071-P defines a length of degradation-free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces, with applicable safety factors applied. Application of the limited hot and cold leg tubesheet inspection criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage as described in WCAP-1707[1]-P shows that significant margin exists between an acceptable level of leakage during normal operating

> conditions that ensures meeting the accident-induced leakage assumptions and the TS leakage limit of 150 gpd.

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: Vincent Zabielski, PSEG Nuclear LLC – N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Branch Chief: Harold K. Chernoff.

Order Imposing Procedures for Access to Sensitive Unclassified Non-Safeguards Information for Contention Preparation

Carolina Power & Light Company, Docket No. 50-261, H. B. Robinson Steam Electric Plant (HBRSEP) Unit No. 2, Darlington County, South Carolina

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50-271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

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

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

Exelon Generation Company, LLC, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

PSEG Nuclear LLC, Docket No. 50-272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

A. This Order contains instructions regarding how potential parties to this proceeding may request access to documents containing Sensitive Unclassified Non-Safeguards Information (SUNSI).

B. Within 10 days after publication of this notice of hearing and opportunity to petition for leave to intervene, any potential party who believes access to SUNSI is necessary to respond to this notice may request such access. A "potential party" is any person who intends to participate as a party by demonstrating standing and filing an admissible contention under 10 CFR 2.309. Requests for access to SUNSI submitted later than 10 days after publication will not be considered absent a showing of good cause for the late filing, addressing why the request could not have been filed earlier.

C. The requestor shall submit a letter requesting permission to access SUNSI to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemakings and Adjudications Staff, and provide a copy to the Associate General Counsel for Hearings, Enforcement and Administration, Office of the General Counsel, Washington, DC 20555-0001. The expedited delivery or courier mail address for both offices is: U.S. Nuclear Regulatory Commission, 11555 Rockville Pike, Rockville, Maryland 20852. The e-mail address for the Office of the Secretary and the Office of the General Counsel are

Hearing.Docket@nrc.gov and *OGCmailcenter@nrc.gov*, respectively.[1] The request must include the following information:

(1) A description of the licensing action with a citation to this *Federal Register* notice;

(2) The name and address of the potential party and a description of the potential party's particularized interest that could be harmed by the action identified in C.(1);

(3) The identity of the individual or entity requesting access to SUNSI and the requestor's basis for the need for the information in order to meaningfully participate in this adjudicatory proceeding. In particular, the request must explain why publicly-available versions of the information requested would not be sufficient to provide the basis and specificity for a proffered contention;

D. Based on an evaluation of the information submitted under paragraph C.(3) the NRC staff will determine within 10 days of receipt of the request whether:

(1) There is a reasonable basis to believe the petitioner is likely to establish standing to participate in this NRC proceeding; and

(2) The requestor has established a legitimate need for access to SUNSI.

E. If the NRC staff determines that the requestor satisfies both D.(1) and D.(2) above, the NRC staff will notify the requestor in writing that access to SUNSI has been granted. The written notification will contain instructions on how the requestor may obtain copies of the requested documents, and any other conditions that may apply to access to those documents. These conditions may include, but are not limited to, the signing of a Non-Disclosure Agreement

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[1] While a request for hearing or petition to intervene in this proceeding must comply with the filing requirements of the NRC's "E-Filing Rule," the initial request to access SUNSI under these procedures should be submitted as described in this paragraph.

or Affidavit, or Protective Order[2] setting forth terms and conditions to prevent the unauthorized or inadvertent disclosure of SUNSI by each individual who will be granted access to SUNSI.

F. Filing of Contentions. Any contentions in these proceedings that are based upon the information received as a result of the request made for SUNSI must be filed by the requestor no later than 25 days after the requestor is granted access to that information. However, if more than 25 days remain between the date the petitioner is granted access to the information and the deadline for filing all other contentions (as established in the notice of hearing or opportunity for hearing), the petitioner may file its SUNSI contentions by that later deadline.

G. Review of Denials of Access.

(1) If the request for access to SUNSI is denied by the NRC staff either after a determination on standing and need for access, or after a determination on trustworthiness and reliability, the NRC staff shall immediately notify the requestor in writing, briefly stating the reason or reasons for the denial.

(2) The requestor may challenge the NRC staff's adverse determination by filing a challenge within 5 days of receipt of that determination with: (a) the presiding officer designated in this proceeding; (b) if no presiding officer has been appointed, the Chief Administrative Judge, or if he or she is unavailable, another administrative judge, or an administrative law judge with jurisdiction pursuant to 10 CFR 2.318(a); or (c) if another officer has been designated to rule on information access issues, with that officer.

[2] Any motion for Protective Order or draft Non-Disclosure Affidavit or Agreement for SUNSI must be filed with the presiding officer or the Chief Administrative Judge if the presiding officer has not yet been designated, within 30 days of the deadline for the receipt of the written access request.

H. Review of Grants of Access. A party other than the requestor may challenge an NRC staff determination granting access to SUNSI whose release would harm that party's interest independent of the proceeding. Such a challenge must be filed with the Chief Administrative Judge within 5 days of the notification by the NRC staff of its grant of access.

If challenges to the NRC staff determinations are filed, these procedures give way to the normal process for litigating disputes concerning access to information. The availability of interlocutory review by the Commission of orders ruling on such NRC staff determinations (whether granting or denying access) is governed by 10 CFR 2.311.[3]

I. The Commission expects that the NRC staff and presiding officers (and any other reviewing officers) will consider and resolve requests for access to SUNSI, and motions for protective orders, in a timely fashion in order to minimize any unnecessary delays in identifying those petitioners who have standing and who have propounded contentions meeting the

[3] Requestors should note that the filing requirements of the NRC's E-Filing Rule (72 FR 49139; August 28, 2007) apply to appeals of NRC staff determinations (because they must be served on a presiding officer or the Commission, as applicable), but not to the initial SUNSI request submitted to the NRC staff under these procedures.

specificity and basis requirements in 10 CFR Part 2. Attachment 1 to this Order summarizes the general target schedule for processing and resolving requests under these procedures.

IT IS SO ORDERED.

Dated at Rockville, Maryland, this 23rd day of December 2009.

For the Nuclear Regulatory Commission.

/RA/

Annette L. Vietti-Cook,
Secretary of the Commission.

ATTACHMENT 1--General Target Schedule for Processing and Resolving Requests for Access to Sensitive Unclassified Non-Safeguards Information in this Proceeding

Day	Event/Activity
0	Publication of *Federal Register* notice of hearing and opportunity to petition for leave to intervene, including order with instructions for access requests.
10	Deadline for submitting requests for access to Sensitive Unclassified Non-Safeguards Information (SUNSI) with information: supporting the standing of a potential party identified by name and address; describing the need for the information in order for the potential party to participate meaningfully in an adjudicatory proceeding.
60	Deadline for submitting petition for intervention containing: (i) Demonstration of standing; (ii) all contentions whose formulation does not require access to SUNSI (+25 Answers to petition for intervention; +7 requestor/petitioner reply).
20	Nuclear Regulatory Commission (NRC) staff informs the requestor of the staff's determination whether the request for access provides a reasonable basis to believe standing can be established and shows need for SUNSI. (NRC staff also informs any party to the proceeding whose interest independent of the proceeding would be harmed by the release of the information.) If NRC staff makes the finding of need for SUNSI and likelihood

Day	Event/Activity
	of standing, NRC staff begins document processing (preparation of redactions or review of redacted documents).
25	If NRC staff finds no “need” or no likelihood of standing, the deadline for requestor/petitioner to file a motion seeking a ruling to reverse the NRC staff’s denial of access; NRC staff files copy of access determination with the presiding officer (or Chief Administrative Judge or other designated officer, as appropriate). If NRC staff finds “need” for SUNSI, the deadline for any party to the proceeding whose interest independent of the proceeding would be harmed by the release of the information to file a motion seeking a ruling to reverse the NRC staff’s grant of access.
30	Deadline for NRC staff reply to motions to reverse NRC staff determination(s).
40	(Receipt +30) If NRC staff finds standing and need for SUNSI, deadline for NRC staff to complete information processing and file motion for Protective Order and draft Non-Disclosure Affidavit. Deadline for applicant/licensee to file Non-Disclosure Agreement for SUNSI.

A	If access granted: Issuance of presiding officer or other designated officer decision on motion for protective order for access to sensitive information (including schedule for providing access and submission of contentions) or decision reversing a final adverse determination by the NRC staff.
A + 3	Deadline for filing executed Non-Disclosure Affidavits. Access provided to SUNSI consistent with decision issuing the protective order.
A + 28	Deadline for submission of contentions whose development depends upon access to SUNSI. However, if more than 25 days remain between the petitioner's receipt of (or access to) the information and the deadline for filing all other contentions (as established in the notice of hearing or opportunity for hearing), the petitioner may file its SUNSI contentions by that later deadline.
A + 53	(Contention receipt +25) Answers to contentions whose development depends upon access to SUNSI.
A + 60	(Answer receipt +7) Petitioner/Intervenor reply to answers.
>A + 60	Decision on contention admission.