[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[NRC-2011-0021]

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, NRC, 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) 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, Rules, Announcements and Directives Branch (RADB), 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 RADB 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, Room O1-F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852-2738.

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, Room O1-F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852-2738, 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, 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 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 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 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 the Electronic Information Exchange System, 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 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, 20852-2738, 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://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. 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, Room O1-F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852-2738. 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.*

Dominion Nuclear Connecticut Inc., et al., Docket Nos. 50-336 and 50-423, Millstone Power Station, Units 2 and 3, New London County, Connecticut

Date of amendment request: July 12, 2010, as supplemented by letter dated August 5, 2010.

Description of amendment request: **This amendment request contains sensitive unclassified non-safeguards information (SUNSI).** The licensee proposed an amendment to the Facility Operating Licenses for Millstone Power Station, Units 2 and 3 (MPS2 and MPS3, respectively). This amendment request pertains to the MPS2 and MPS3 Cyber Security Plans. In the same amendment request letter, sent under Dominion Resources Services, Inc. (DRC) letterhead, Kewaunee Power Station, Surry Power Station Units 1 and 2, and North Anna Power Station Units 1 and 2, submitted amendment requests pertaining to their Cyber Security Plans. This notice only addresses the application as it pertains to MPS2 and MPS3. The licensee requested NRC approval of the MPS2 and MPS3 Cyber Security Plan, provided a proposed implementation schedule, and proposed to add a sentence to License Condition 2.C.4, “Physical Protection,” of MPS2, Facility Operating License (FOL) DPR-65 and to License Condition 2.E, of MPS3, FOL NPF-49, that would affirm when the licensee would fully implement and maintain in effect all provisions of the Cyber Security Plan.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (NSHC). The NRC staff reviewed the licensee's NSHC analysis against the standards of 10 CFR 50.92(c). The NRC staff's review 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 Plan establishes the licensing basis for the Cyber Security Program for the sites. The Plan establishes how to achieve high assurance that specified nuclear power plant digital computer and communication systems, networks and functions are adequately protected against cyber attacks up to and including the design basis threat.

   Part one of the proposed change is designed to achieve high assurance that the systems are protected from cyber attacks. The Plan describes how plant modifications that involve digital computer systems are reviewed to provide high assurance of adequate protection against cyber attacks, up to and including the design basis threat. The proposed change does not alter accident analysis assumptions, add any initiators, or affect the function of plant systems or the manner in which systems are operated, maintained, modified, tested, or inspected. The first part of the proposed change is designed to achieve high assurance that the systems within the scope of the requirement are protected from cyber attacks and has no impact on the probability or consequences of an accident previously evaluated. The proposed change implements a Cyber Security Plan as a requirement not formally addressed previously. As such, the proposed Plan provides a significant enhancement to cyber security where no requirement existed before.

   The second part of the proposed change adds a sentence to the existing facility license conditions for Physical Protection. These changes are administrative and have no impact on the probability or consequences of an accident previously evaluated.

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

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

   Response: No.

This proposed amendment provides assurance that safety-related structures, systems and components (SSCs) are protected from cyber attacks. Implementation of 10 CFR 73.54 and the inclusion of a plan in the FOL do not result in the need of any new or different design-basis accident analysis. It does not introduce new equipment that could create a new or different kind of accident, and no new equipment failure modes are created. As a result, no new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of this proposed amendment.

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

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

Response: No.

The margin of safety is associated with the confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant pressure boundary, and containment structure) to limit the level of radiation to the public. The proposed amendment would not alter the way any safety-related SSC functions and would not alter the way the plant is operated. The amendment provides assurance that safety-related SSCs are protected from cyber attacks. The proposed amendment would not introduce any new uncertainties or change any existing uncertainties associated with any safety limit. The proposed amendment would have no impact on the structural integrity of the fuel cladding, reactor coolant pressure boundary, or containment structure. Based on the above considerations, the proposed amendment would not degrade the confidence in the ability of the fission product barriers to limit the level of radiation to the public.

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

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: Harold K. Chernoff.

Exelon Generation Company, LLC,

Docket Nos. STN 50-456 and 50-457, Braidwood Station, Units 1 and 2, Will County, Illinois

Docket Nos. STN 50-454 and 50-455, Byron Station, Units 1 and 2, Ogle County, Illinois

Date of amendment request: December 14, 2010.

Description of amendment request: **This amendment request contains sensitive unclassified non-safeguards information (SUNSI).** The amendment would revise Technical Specification (TS) 5.5.9, “Steam Generator (SG) Program,” to exclude portions of the tubes within the tubesheet from periodic SG inspections and plugging or repair. In addition, this amendment request proposes to revise TS 5.6.9, "Steam Generator (SG) Tube Inspection Report," to remove reference to previous interim alternate repair criteria and provide reporting requirements specific to the temporary alternate criteria.

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 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 various accidents previously evaluated, the proposed changes only affect the steam generator tube rupture (SGTR), postulated steam line break (SLB), feedwater line break (FLB), locked rotor and control rod ejection accident evaluations. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a safe shutdown earthquake (SSE); however, the seismic analysis of Model D5 SGs has shown that axial loading of the tubes is negligible during an SSE.

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 draft Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," and TS 5.5.9, 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 TS 5.5.9. 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-tosecondary 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.

Primary-to-secondary leakage from tube degradation in the tubesheet area during operating and accident conditions is restricted due to contact of the tube with the tubesheet. The leakage is modeled as flow through a porous medium through the use of the Darcy equation. The leakage model is used to develop a relationship between operational leakage and leakage at accident conditions that is based on differential pressure across the tubesheet and the viscosity of the fluid. A leak rate ratio was developed to relate the leakage at operating conditions to leakage at accident conditions. Since the fluid viscosity is based on fluid temperature and it is shown that for the most limiting accident, the fluid temperature does not exceed the normal operating temperature and therefore the viscosity ratio is assumed to be 1.0. Therefore, the leak rate ratio is a function of the ratio of the accident differential pressure and the normal operating differential pressure.

The leakage factor of 1.93 for Braidwood Station Unit 2 and Byron Station Unit 2, for a postulated SLB/FLB, has been calculated as shown in Table 9-7 of WCAP-17072-P. However, EGC Braidwood Station Unit 2 and Byron Station Unit 2 will apply a factor of 3.11 as determined by Westinghouse evaluation LTR-SGMP-09-100 P-Attachment, Revision 1, to the normal operating leakage associated with the tubesheet expansion region in the condition monitoring (CM) and operational assessment (OA). The leakage factor of 3.11 applies specifically to Byron Unit 2 and Braidwood Unit 2, both hot and

cold legs, in Table RAI24-2 of LTRSGMP-09-100 P-Attachment, Revision 1. 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 assumed accident induced leak rate limit is 0.5 gallons per minute at room temperature (gpmRT) for the faulted SG and 0.218 gpmRT for the unfaulted SGs for accidents that assume a faulted SG. These accidents are the SLB and the locked rotor with a stuck open PORV. The assumed accident induced leak rate limit for accidents that do not assume a faulted SG is 1.0 gpmRT for all SGs. These accidents are the locked rotor and control rod ejection.

No leakage factor will be applied to the locked rotor or control rod ejection transients due to their short duration, since the calculated leak rate ratio is less than 1.0.

The TS 3.4.13 operational leak rate limit is 150 gallons per day (gpd) (0.104 gpmRT) through any one SG. Consequently, there is sufficient margin between accident leakage and allowable operational leakage. The maximum accident leak rate ratio for the Model D5 design SGs is 1.93 as indicated in WCAP-1 7072-P, Table 9-7. However, EGC will use the more conservative value of 3.11 accident leak rate ratio for the most limiting SG model design identified in Table RA124-2 of LTR-SGMP-09-100 P-Attachment Revision 1. This results in significant margin between the conservatively estimated accident leakage and the allowable accident leakage (0.5 gpmRT).

For the CM assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 3.11 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the 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 3.11 and compared to the observed operational leakage.

Based on the above, the performance criteria of NEI-97-06, Revision 2, and draft RG 1.121 continue to be met and the proposed change does not involve a significant increase in the probability or consequences of the applicable accidents 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 introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity is expected to be maintained for all plant conditions upon implementation of the permanent alternate repair criteria. The proposed change does not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

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

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

   Response: No

   The proposed change defines the safety significant portion of the SG tube that must be inspected and repaired. WCAP-17072-P as modified by WCAP-1 7330-P identifies the specific inspection depth below which any type tube degradation has no impact on the performance criteria in NEI 97-06, Revision 2, "Steam Generator Program Guidelines."

   The proposed change that alters the SG inspection and reporting criteria maintains the required structural margins of the SG tubes for both normal and accident conditions. NEI 97-06, and draft RG 1.121 are used as the bases in the development of the limited tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. Draft 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. Draft 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 draft 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, WCAP-1 7072-P as modified by WCAP-17330-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-17072-P as modified by LTRSGMP-09-100 P-Attachment shows that significant margin exists between an acceptable level of leakage during normal operating conditions that ensures meeting the SLB accident-induced leakage assumption and the TS leakage limit of 150 gpd.

   Based on the above, it is concluded that the proposed changes do not result in any 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: Robert D. Carlson.

Exelon Generation Company, LLC, Docket No. 50-461, Clinton Power Station (CPS), Unit 1, DeWitt County, Illinois

Date of amendment request: September 23, 2010.

Description of amendment request: **This amendment request contains sensitive unclassified non-safeguards information (SUNSI).** The proposed amendment would modify the CPS Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.7.6, "Main Turbine Bypass System," by allowing revision of the reactor operational limits, as specified in the CPS Core Operating Limits Report (COLR), to compensate for the inoperability of the Main Turbine Bypass System (MTBS). The revised TS will require that either the MTBS be OPERABLE or that the reactor power, Minimum Critical Power Ratio (MCPR), and Linear Heat Generation Rate (LHGR) limits for an inoperable MTBS be placed in effect as specified in the COLR. Additionally, the amendment proposes modifying TS 5.6.5, "Core Operating Limits Report (COLR)," to add a requirement to establish cycle dependent reactor thermal power limits for an inoperable MTBS.

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 MTBS functions to limit reactor pressure and power increases during certain transients postulated in the accident analysis. The MTBS is a mitigation function and not the initiator of any evaluated accident or transient. Operation with an inoperable MTBS while in compliance with the imposed reactor power limitation, and MCPR and LHGR limits will offset the impact of losing the MTBS function.

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

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

   Response: No.

   The proposed change will not create any new modes of plant or equipment operation. The proposed change allows the option to apply a reactor power penalty and an additional penalty factor to the MCPR and LHGR when the MTSS is inoperable. The imposed reactor power limitation and the revised set of MCPR and LHGR limits will offset the impact of losing the MTBS function, and maintain the margin to the MCPR safety limit and the thermal mechanical design limits.

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

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

   Response: No.

   By establishing more restrictive reactor power and MCPR and LHGR operating limits, there are no changes to the plant design and safety analysis. There are no changes to the reactor core design instrument setpoints. The margin of safety assumed in the safety analysis is not affected. Applicable regulatory requirements will continue to be met and adequate defense-in- depth will be maintained. Sufficient safety margins will be maintained.

   The analytical methods used to determine the reactor power limitation and the revised core operating limits were reviewed and approved by the NRC and are described in Technical Specification 5.6.5, "Core Operating Limits Report (COLR)."

   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: Robert D. Carlson.

Exelon Generation Company, LLC, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station (DNPS), Units 2 and 3, Grundy County, Illinois

Date of amendment request: October 4, 2010.

Description of amendment request: **This amendment request contains sensitive unclassified non-safeguards information (SUNSI).** The proposed amendment would revise Technical Specification (TS) Table 3.3.1.1 to eliminate Functions 5 and 10 from TS Table 3.3.1.1-1, delete footnote (c) from that table, and rename the footnote (d) to (c). These revisions would eliminate the requirement for a reactor scram, if vessel pressure is greater than or equal to 600 pounds per square inch gage (psig), with the reactor mode switch in startup and the main steam isolation valves closed or with a main turbine condenser vacuum low condition.

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 changes to the DNPS Units 2 and 3 TS revise the applicability of two protective functions and delete the associated TS Action statement. TS requirements that govern operability or routine testing of plant instruments are not assumed to be initiators of any analyzed event because these instruments are intended to prevent, detect, or mitigate accidents. Specifically, the reactor scram associated with the main steam isolation valve (MSIV) closure and low condenser vacuum (i.e., Functions 5 and 10 of TS 3.3.1.1) is in anticipation of the loss of the normal heat sink and subsequent overpressurization transient. The scram at high pressure in startup conditions when MSIVs close and/or main condenser vacuum is low does not impact the limiting accident

or transient analyses. An analysis by General Electric Hitachi Nuclear Energy (GEH) demonstrated that the Mode 2 scram function for MSIV closure and low condenser vacuum can be eliminated without affecting safe plant operation. Elimination of these required scrams will not involve an increase in the probability of an accident previously evaluated.

Additionally, these proposed changes will not increase the consequences of an accident previously evaluated because the proposed changes do not adversely impact structures, systems, or components. These changes will not alter the operation of equipment assumed to be available for the mitigation of accidents or transients by the plant safety analysis.

Function 5 is currently required in Mode 2 with reactor pressure greater than or equal to 600 psig to ensure that the reactor is shut down, thus helping to prevent an overpressurization transient due to closure of main steam isolation valves. Similarly, Function 10 is currently required in Mode 2 with reactor pressure greater than or equal to 600 psig to help prevent an overpressurization transient by anticipating the turbine stop valve closure scram on loss of condenser vacuum.

The existing scram logic is the result of experience gained during startup of an early vintage bailing water reactor in 1966 when operators had difficulty controlling reactor power above approximately 600 psig without pressure control. Experience on later plant startups indicates that the early experience may not be inherent to later boiling water reactor designs. As such, GEH subsequently recommended elimination of the Mode 2 scram requirement.

In Mode 2, the heat generation rate is low enough so that the other diverse Reactor Protection System (RPS) functions provide sufficient protection from an overpressurization transient. During normal power ascension in Mode 2 with the MSIVs open, reactor pressure vessel (RPV) pressure is controlled by the pressure regulator with increasing pressure setpoints. The maximum pressure regulator setpoint, which would translate to 1000 psig at rated power, would only allow a maximum dome pressure of approximately 900 psig in the Mode 2 power range. The potential scenario in Mode 2 whereby the MSIVs would close unexpectedly and cause the pressure to increase would lead to the Average Power Rate Monitors, Neutron Flux-High, Setdown scram (i.e., TS 3.3.1.1, Function 2.a), followed by the Reactor Vessel Steam Dome Pressure-High scram (i.e., TS 3.3.1.1, Function 3).

The consequences of a previously analyzed event are dependent on the initial conditions assumed in the analysis, the availability and successful functioning of equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The consequences of a previously evaluated accident are not significantly increased by the proposed change. The proposed change does not affect the performance of any equipment credited to mitigate the radiological consequences of an accident. Furthermore, there will be no change in the types or significant increase in the amounts of any effluents released offsite.

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

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

   Response: No.

   The proposed changes to the DNPS Units 2 and 3 TS revise the applicability of two protective functions and delete the associated TS Action statement. The RPS functions are not an initiator of any accident. Rather, the RPS is designed to initiate a reactor scram when one or more monitored parameters exceed their specified limits to preserve the integrity of the fuel cladding and the reactor coolant pressure boundary and minimize the energy that must be absorbed following an accident. The proposed changes do not alter the applicability for RPS functions during plant conditions in which an overpressurization transient is assumed to occur. Specifically, no changes are being made to the required number of channels per trip system, surveillance requirements, or allowable values for these functions during Mode 1 operation.

   The proposed change does not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed change does not change or introduce any new equipment, modes of system operation or failure mechanisms.

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

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

   Response: No.

   Margins of safety are established in the design of components, the configuration of components to meet certain performance parameters, and in the establishment of setpoints to initiate alarms and actions. The proposed changes revise the applicability for Functions 5 and 10 of TS 3.3.1.1 and delete an associated TS Action Statement. The proposed changes do not alter the applicability for RPS functions during plant conditions in which an overpressurization transient is assumed to occur.

   In addition, the proposed changes do not affect the probability of failure or availability of the affected instrumentation. Furthermore, the proposed changes will reduce the probability of test-induced plant transients and equipment failures.

   The proposed changes to the applicability for Functions 5 and 10 of TS 3.3.1.1 have no impact on equipment design or fundamental operation. There are no changes being made to safety limits or safety system allowable values that would adversely affect plant safety. The performance of the systems important to safety is not significantly affected by the proposed changes. The proposed change does not affect safety analysis assumptions or initial conditions and therefore, the margin of safety in the original safety analyses is maintained.

As documented above, 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: Robert. D. Carlson.

Exelon Generation Company, LLC, Docket No. 50-353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of amendment request: December 15, 2010.

Description of amendment request: **This amendment request contains sensitive unclassified non-safeguards information (SUNSI).** The proposed changes revise the Technical Specification (TS) relating to the Safety Limit Minimum Critical Power Ratios (SLMCPRs). The changes result from a cycle-specific analysis performed to support the operation of Limerick Generating Station, Unit 2, in the upcoming Cycle 12. Specifically, the proposed TS changes will revise the SLMCPRs contained in TS 2.1 for two recirculation loop operation and single recirculation loop operation to reflect the changes in the cycle-specific analysis. The new SLMCPRs are calculated using Nuclear Regulatory Commission (NRC)-approved methodology described in NEDE 24011-P-A, “General Electric Standard Application for Reactor Fuel,” Revision 17.

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 derivation of the cycle specific Safety Limit Minimum Critical Power Ratios (SLMCPRs) for incorporation into the Technical Specifications (TS), and their use to determine cycle specific thermal limits, has been performed using the methodology discussed in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Revision 17.

   The basis of the SLMCPR calculation is to ensure that during normal operation and during abnormal operational transients, at least 99.9% of all fuel rods in the core do not experience transition boiling if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling.

   The MCPR [minimum critical power ratio] safety limit is reevaluated for each reload using NRC-approved methodologies. The analyses for Limerick Generating Station (LGS), Unit 2, Cycle 12 have concluded that a two loop MCPR safety limit of ≥1.09, based on the application of Global Nuclear Fuel's 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 LGS, Unit 2 Core Operating Limits Report (COLR).

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

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

   Response: No.

   The SLMCPR is a TS numerical value, calculated to ensure that during normal operation and during abnormal operational transients, at least 99.9% of all fuel rods in the core do not experience transition boiling if the limit is not violated. The new SLMCPRs are calculated using NRC-approved methodology discussed in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Revision 17. The proposed changes do not involve any new modes of operation or any plant modifications. The proposed revised MCPR safety limits have been shown to be acceptable for Cycle 12 operation. The core operating limits will

continue to be developed using NRC-approved methods. 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, the proposed TS changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

There is no significant reduction in the margin of safety previously approved by the NRC as a result of the proposed change to the SLMCPRs. The new SLMCPRs are calculated using methodology discussed in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Revision 17. The SLMCPRs ensure that during normal operation and during abnormal operational transients, at least 99.9% of all fuel rods in the core do not experience transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity.

Therefore, the proposed TS changes do not involve a significant reduction in the margin of safety previously approved by the NRC.

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: Harold K. Chernoff.

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

Date of amendment request: July 16, 2010, as supplemented by letters dated September 28, and November 23, 2010.

Description of amendment request: **This amendment request contains sensitive unclassified non-safeguards information (SUNSI).** The proposed amendment to the Facility

Operating License (FOL) includes: (1) the proposed Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS) Cyber Security Plan (the Plan), (2) an implementation schedule, and (3) revise the existing FOL Physical Protection license condition to require the FirstEnergy Nuclear Operating Company (FENOC, the licensee) to fully implement and maintain in effect all provisions of the Commission approved Cyber Security Plan as required by Title 10 of the *Code of Federal Regulations* (10 CFR) 73.54.

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:

> Criterion 1**:** The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.
>
> The proposed change is required by 10 CFR 73.54 and includes three parts. The first part is the submittal of the Plan for NRC review and approval. The Plan provides a description of how the requirements of the rule will be implemented at the DBNPS. The Plan establishes the licensing basis for the FENOC cyber security program for the DBNPS. The Plan establishes how to achieve high assurance that nuclear power plant digital computer and communication systems and networks associated with the following are adequately protected against cyber attacks up to and including the design basis threat:
>
> 1. Safety-related and important-to-safety functions,
> 2. Security functions,
> 3. Emergency preparedness functions including offsite communications, and
> 4. Support systems and equipment which if compromised, would adversely impact safety, security, or emergency preparedness functions.
>
> Part one of the proposed change is designed to achieve high assurance that the systems are protected from cyber attacks. The Plan itself does not require any plant modifications. However, the Plan does describe how plant modifications which involve digital computer systems are reviewed to provide high assurance of adequate protection against cyber attacks, up to and including the design basis threat as defined in the rule.
>
> The proposed change does not alter the plant configuration, require new plant equipment to be installed, alter accident analysis assumptions, add any initiators, affect the function of plant systems, or affect the manner in which systems are operated. The first part of the proposed change is designed to achieve high assurance that the systems within the scope of the rule are protected from cyber attacks and has no impact on the probability or consequences of an accident previously evaluated.

The second part of the proposed change is an implementation schedule. The third part adds a sentence to the existing FOL license condition 2.D for Physical Protection. Both of these changes are administrative and have no impact on the probability or consequences of an accident previously evaluated.

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

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

The proposed change is required by 10 CFR 73.54 and includes three parts. The first part is the submittal of the Plan for NRC review and approval. The Plan provides a description of how the requirements of the rule will be implemented at the DBNPS. The Plan establishes the licensing basis for the FENOC cyber security program for the DBNPS. The Plan establishes how to achieve high assurance that nuclear power plant digital computer and communication systems and networks associated with the following are adequately protected against cyber attacks up to and including the design basis threat:

1. Safety-related and important-to-safety functions,
2. Security functions,
3. Emergency preparedness functions including offsite communications, and
4. Support systems and equipment which if compromised, would adversely impact safety, security, or emergency preparedness functions.

Part one of the proposed change is designed to achieve high assurance that the systems within the scope of the rule are protected from cyber attacks. The Plan itself does not require any plant modifications. However, the Plan does describe how plant modifications which involve digital computer systems are reviewed to provide high assurance of adequate protection against cyber attacks, up to and including the design basis threat defined in the rule.

The proposed change does not alter the plant configuration, require new plant equipment to be installed, alter accident analysis assumptions, add any initiators, affect the function of plant systems, or affect the manner in which systems are operated. The first part of the proposed change is designed to achieve high assurance that the systems within the scope of the rule are protected from cyber attacks and does not create the possibility of a new or different kind of accident from any previously evaluated.

The second part of the proposed change is an implementation schedule. The third part adds a sentence to the existing FOL license condition 2.D for Physical Protection. Both of these changes are administrative and do not create the possibility of a new or different kind of accident from any previously evaluated.

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

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

The proposed change is required by 10 CFR 73.54 and includes three parts. The first part is the submittal of the Plan for NRC review and approval. The Plan provides a description of how the requirements of the rule will be implemented at the DBNPS. The Plan establishes the licensing basis for the FENOC cyber security program for the DBNPS. The Plan establishes how to achieve high assurance that nuclear power plant digital computer and communication systems and networks associated with the following are adequately protected against cyber attacks up to and including the design basis threat:

1. Safety-related and important-to-safety functions,
2. Security functions,
3. Emergency preparedness functions including offsite communications, and
4. Support systems and equipment which if compromised, would adversely impact safety, security, or emergency preparedness functions.

Part one of the proposed change is designed to achieve high assurance that the systems within the scope of the rule are protected from cyber attacks. Plant safety margins are established through Limiting Conditions for Operation, Limiting Safety System Settings and Safety limits specified in the Technical Specifications, methods of evaluation that establish design basis or change Updated Final Safety Analysis. Because there is no change to these established safety margins, the proposed change does not involve a significant reduction in a margin of safety.

The second part of the proposed change is an implementation schedule. The third part adds a sentence to the existing FOL license condition 2.D for Physical Protection. Both of these changes are administrative and do 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: David W. Jenkins, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Branch Chief: Robert. D. Carlson.

Luminant Generation Company LLC, Docket Nos. 50-445 and 50-446, Comanche Peak Nuclear Power Plant, Units 1 and 2, Somervell County, Texas

Date of amendment request: December 1, 2010.

Brief description of amendments: **This amendment request contains sensitive unclassified non-safeguards information (SUNSI).** The proposed amendment would revise Technical Specification (TS) 5.5.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program," to exclude portions of the Unit 2 Model D5 steam generator (SG) tubes below the top of the SG tubesheet from periodic SG tube inspections during Comanche Peak Nuclear Power Plant (CPNPP), Unit 2 Refueling Outage 12 and the subsequent operating cycle. In addition, the proposed amendment would revise TS 5.6.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report," to provide reporting requirements specific to CPNPP, Unit 2 for the temporary alternate repair criteria.

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

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

   Response: No.

   Of the 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.

   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,

differential pressure between the primary and secondary side, and tubesheet rotation. Based on this design, the structural margins against burst, as discussed in [NRC] Regulatory Guide (RG) 1.121, “Bases for Plugging Degraded PWR [Pressurized-Water Reactor] Steam Generator Tubes,” and TS 5.5.9 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 in TS 5.5.9. Therefore, the proposed change results in no significant increase in the probability of the occurrence of a[n] 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[n] SGTR.

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

The leakage factor of 3.16 for CPNPP Unit 2, for a postulated SLB/FLB, has been calculated as described in Reference 8.29 [Westinghouse Letter LTR-SGMP-09-100P-Attachment, Revision 1, dated September 7, 2010] and is shown in Revised Table 9- 7 of this same reference. Specifically, 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 3.16 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 3.16 and compared to the observed operational leakage. The accident-induced leak rate limit for CPNPP Unit 2 is 1.0 gpm [gallons per minute]. The TS operational leak rate limit through any one steam generator is 150 gpd [gallons per day] (0.1 gpm). Consequently, there is significant margin between accident leakage and allowable operational leakage. The SLB/FLB overall leakage factor is 3.16 resulting in significant margin between the conservatively estimated accident induced leakage and the allowable accident leakage.

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

The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that alters the 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.

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

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

   Response: No.

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

   Response: No.

   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, Rev.2, “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 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[n] SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation, the

> probability and consequences of a[n] SGTR are reduced. RG 1.121 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) [Boiler and Pressure Vessel] Code.
>
> For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, the H* Analysis documented in Section 4.1 [Attachment 1 to letter dated December 1, 2010] 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 provides for large margins between calculated and actual leakage values in the proposed limited tubesheet inspection depth criteria.
>
> Therefore, the proposed change does not involve a significant reduction in any 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: Timothy P. Matthews, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: November 30, 2010.

Description of amendment request: **This amendment request contains sensitive unclassified non-safeguards information (SUNSI).** The proposed amendment would revise the Wolf Creek Generating Station's (WCGS's) Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," to exclude portions of the tube below the top of the steam generator

tubesheet from periodic steam generator tube inspections during Refueling Outage 18 and the subsequent operating cycle. In addition, the proposed amendment would revise TS 5.6.10, ”Steam Generator Tube Inspection Report," to remove references to previous interim alternate repair criteria and provide reporting requirements specific to the temporary alternate repair criteria.

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 previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that alters the steam generator inspection 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 steam generator tube inspection and repair criteria are the steam generator tube rupture (SGTR) event and the feedline break (FLB) postulated accidents.

   During the SGTR event, the required structural integrity margins of the steam generator 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 presence of the tubesheet and constraint provided by 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, from the differential pressure between the primary and secondary side, and tubesheet deflection. 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 TS 5.5.9 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 steam generator tubes consistent with the performance criteria in TS 5.5.9. Therefore, the proposed change results in no significant increase in the probability of the occurrence of a[n] SGTR accident.

At normal operating pressures, leakage from tube degradation below the proposed limited inspection depth is limited by the tube-to-tubesheet joint. 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 changes do not result in a significant increase in the consequences of a[n] SGTR.

The consequences of a steam line break (SLB) are also not significantly affected by the proposed changes. During a[n] SLB accident, the reduction in pressure above the tubesheet on the shell side of the steam generator creates an axially uniformly distributed load on the tubesheet due to the reactor coolant system pressure on the underside of the tubesheet. The resulting bending action constrains the tubes in the tubesheet thereby restricting primary-to-secondary leakage below the midplane.

Primary-to-secondary leakage from tube degradation in the tubesheet area during the limiting accident (i.e., an SLB) is limited by flow restrictions. These restrictions result from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications.

The leakage factor of 2.50 for WCGS, for a postulated SLB/FLB, has been calculated as shown in Revised Table 9-7 of Reference 15 [Westinghouse Letter LTR-SGMP-09-100, dated August 12, 2009]. Specifically, 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 2.50 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 2.50 and compared to the observed operational leakage.

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

event. SLB leakage is limited by leakage flow restrictions resulting from the leakage path above potential cracks through the tube-to-tubesheet crevice. The leak rate during postulated accident conditions (including locked rotor) has been shown to remain within the accident analysis assumptions for all axial and or circumferentially orientated cracks occurring 15.2 inches below the top of the tubesheet. The accident induced leak rate limit for WCGS is 1.0 gpm [gallon per minute]. The TS 3.4.13, "RCS [Reactor Coolant System] Operational LEAKAGE," operational leak rate limit is 150 gpd [gallons per day] (0.1 gpm) through anyone steam generator. Consequently, accident leakage is approximately 10 times the allowable leakage, if only one steam generator is leaking. Using an SLB/FLB overall leakage factor of 2.50, accident induced leakage is approximately 0.5 gpm, if all 4 steam generators are leaking at 150 gpd at the beginning of the accident. Therefore, significant margin exists between the conservatively estimated accident induced leakage and the allowable accident leakage (1.0 gpm).

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

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

Response: No.

The proposed change alters the steam generator inspection and reporting criteria. It 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 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 accident previously evaluated.

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

Response: No.

The proposed change alters the steam generator inspection and reporting criteria. It maintains the required structural margins of the steam generator tubes for both normal and accident conditions. NEI [Nuclear Energy Institute] 97-06, Revision 2, and RG 1.121, are used as the bases in the development of the limited tubesheet inspection depth methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting GDC [General Design Criterion] 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[n] SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation, the probability and consequences of a[n] 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) [Boiler and Pressure Vessel] Code. For axially-oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially-oriented cracking, the H* Analysis documented in Section 3 [of letter dated November 30, 2010], 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 provides for large margins between calculated and actual leakage values in the proposed limited tubesheet inspection depth criteria.
>
> Therefore, the proposed change does not involve a significant reduction in any margin of safety.

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

Attorney for licensee: Jay Silberg, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, N.W., Washington, DC 20037.

NRC Branch Chief: Michael T. Markley.

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

Dominion Nuclear Connecticut Inc., et al., Docket Nos. 50-336 and 50-423, Millstone Power Station, Unit 2 and 3, New London County, Connecticut

Exelon Generation Company, LLC, Docket No. 50-461, Clinton Power Station (CPS), Unit 1, DeWitt County, Illinois

Exelon Generation Company, LLC, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station (DNPS), Units 2 and 3, Grundy County, Illinois

Exelon Generation Company, LLC, Docket No. 50-353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Exelon Generation Company, LLC

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

Luminant Generation Company LLC, Docket Nos. 50-445 and 50-446, Comanche Peak Nuclear Power Plant, Units 1 and 2, Somervell County, Texas

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

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;

[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.

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 those documents. These conditions may include, but are not limited to, the signing of a Non-Disclosure Agreement 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.

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[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.

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.

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]

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[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.

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 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 25th day of January 2011.

For the 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 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.

Day	Event/Activity
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.