(Closed—Ex. 1)'' be held on August 19, and on less than one week's notice to the public.

The NRC Commission Meeting Schedule can be found on the Internet at: *www.nrc.gov/what-we-do/policy-making/schedule.html.*

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving the Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov.*

Dated: August 15, 2002.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 02–21254 Filed 8–16–02; 10:05 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued, from July 26, 2002 through August 8, 2002. The last biweekly notice was published on August 6, 2002 (67 FR 50947).

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; (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 the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By September 19, 2002, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714,[1] which is available at the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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[1] The most recent version of Title 10 of the CODE OF FEDERAL REGULATIONS, published January 1, 2002, inadvertently omitted the last sentence of 10 CFR 2.714(d) and subparagraphs (d)(1) and (2), regarding petitions to intervene and contentions. Those provisions are extant and still applicable to petitions to intervene. Those provisions are as follows: ''In all other circumstances, such ruling body or officer shall, in ruling on—

(1) A petition for leave to intervene or a request for hearing, consider the following factors, among other things:

(i) The nature of the petitioner's right under the Act to be made a party to the proceeding.

(ii) The nature and extent of the petitioner's property, financial, or other interest in the proceeding.

(iii) The possible effect of any order that may be entered in the proceeding on the petitioner's interest.

(2) The admissibility of a contention, refuse to admit a contention if:

(i) The contention and supporting material fail to satisfy the requirements of paragraph (b)(2) of this section; or

(ii) The contention, if proven, would be of no consequence in the proceeding because it would not entitle petitioner to relief.''

As required by 10 CFR 2.714, 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 factors: (1) The nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies 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, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff, or may be delivered to the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and because of continuing disruptions in delivery of mail to United States Government offices, it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 304–415–4737 or by e-mail to *pdr@nrc.gov.*

Carolina Power & Light Company, et al., Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendments request: July 24, 2002.

Description of amendments request: The proposed amendments would revise Technical Specifications (TS) Section 3.1.7, ''Standby Liquid Control (SLC) System,'' to reflect modifications being made to the system as a result of transition to the GE14 fuel design. To support this transition, the required in-vessel boron concentration, supplied by the SLC system, would be increased from 660 ppm natural boron to a concentration equivalent to 720 ppm natural boron. This would be accomplished by use of sodium pentaborate solution enriched with the Boron-10 isotope. As a result, (1) a new Surveillance Requirement (SR) 3.1.7.8 would be added to verify sodium pentaborate enrichment, (2) the minimum sodium pentaborate concentration value would be lowered in TS Figure 3.1.7–1, ''Sodium Pentaborate Solution Volume Versus Concentration Requirements,'' and (3) the temperature versus concentration requirements of TS Figure 3.1.7–2, ''Sodium Pentaborate Solution Temperature Versus Concentration Requirements,'' would be revised. In a related change, SR 3.1.7.3 would also be revised. Currently, the SR verifies temperature of the SLC pump suction piping. The SR would be revised to verify temperature of the suction and discharge piping up to the SLC injection valves.

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 amendments do not alter the design or operation of the Standby Liquid Control (SLC) system, but rather revise Technical Specification (TS) Section 3.1.7 requirements to ensure acceptable SLC boron solution volume and concentration values to produce a minimum in-vessel boron concentration which is sufficient to bring the reactor to a subcritical condition without taking credit for control rod movement. The existing design of the SLC system is sufficient to handle enriched sodium pentaborate solution, which is chemically and physically similar to the current solution. The SLC system is not considered to be an initiator of any analyzed event. Therefore, the proposed amendments do not increase the probability of a previously evaluated accident.

The current TS Section 3.1.7 requirements ensure acceptable SLC boron solution volume and concentration values to produce a minimum in-vessel natural boron concentration of 660 ppm. The proposed change revises the boron solution requirements of TS Figures 3.1.7–1 and 3.1.7–2, to ensure a minimum in-vessel concentration equivalent to 720 ppm natural boron. A minimum concentration equivalent to 720 ppm natural boron in the reactor is sufficient to bring the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. This concentration was determined by General Electric using the approved methods described in Revision 14 of General Electric Standard Application for Reactor Fuel (GESTAR II), NEDE 24011–P–A. The analysis assumes Brunswick Steam Electric Plant (BSEP) operation with an equilibrium core of GE14 fuel, operating at 2923 megawatts thermal (MWt) with 24 month operating cycles.

As stated above, the in-vessel boron concentration is being raised from 660 ppm natural to 720 ppm equivalent. This will be accomplished by use of sodium pentaborate solution enriched with the Boron-10 isotope. As a result, a new Surveillance Requirement (SR) 3.1.7.8 is added. This SR verifies sodium pentaborate enrichment is greater than or equal to 47 atom percent Boron-10 prior to addition to the SLC tank, thereby ensuring a minimum concentration equivalent to 720 ppm natural boron in the reactor will be achieved.

Use of sodium pentaborate enriched to 47 atom percent Boron-10 allows the volume versus concentration requirements of TS Figure 3.1.7–1 to be lowered. This, in turn, lowers the solution's saturation temperature. Accordingly, the temperature versus concentration requirements of TS Figure 3.1.7–2 are revised. The existing 5° F margin to the saturation temperature specified in the bases is maintained in the revised TS Figure 3.1.7–2.

The concentration requirements of the SLC system boron solution will ensure that the SLC system continues to comply with the requirements of 10 CFR 50.62(c)(4).

The SLC system is also used to maintain suppression pool pH level above 7 following a loss-of-coolant-accident (LOCA) involving significant fission product releases. This ensures that iodine will be retained in the suppression pool water post-LOCA. The revised sodium pentaborate solution requirements were evaluated using the methodology provided in NUREG–1465, ''Accident Source Terms for Light-Water Nuclear Power Plants, Final Report,'' dated February 1, 1995 and NUREG/CR–5950, ''Iodine Evolution and pH Control,'' dated December 1992. This evaluation demonstrated that the SLC system continues to meet its post-LOCA suppression pool pH control design function.

The change to SR 3.1.7.3 is conservative in nature and is consistent with both the current Bases for SR 3.1.7.3 and plant operating practice. Required verification of the discharge as well as suction piping temperature provides additional assurance of system operability.

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

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

Response: No.

The proposed amendments do not alter the design or operation of the SLC system, but rather revise TS Section 3.1.7 requirements to ensure acceptable SLC boron solution volume and concentration values to produce a minimum in-vessel boron concentration which is sufficient to bring the reactor to a subcritical condition without taking credit for control rod movement. The existing design of the SLC system is sufficient to handle enriched sodium pentaborate solution, which is chemically and physically similar to the current solution. Using the enriched solution does not change any of the key SLC system process parameters (*i.e.,* flow rates, discharge pressure, required net positive suction head, etc.). Correct enrichment is ensured by the addition of a new SR to verify sodium pentaborate enrichment prior to addition to the SLC tank. The existing 5° F margin to the saturation temperature specified in the bases is maintained. The change to SR 3.1.7.3 is conservative in nature and is consistent with both the current Bases for SR 3.1.7.3 and plant operating practice. Required verification of the discharge as well as suction piping temperature provides additional assurance of system operability. Therefore, the proposed amendments cannot create a new or different kind of accident from any accident previously evaluated.

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

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

Response: No.

The proposed change revises the boron solution requirements of TS Figures 3.1.7–1 and 3.1.7–2, to ensure a minimum in-vessel concentration equivalent to 720 ppm natural boron. A minimum concentration equivalent to 720 ppm natural boron in the reactor is sufficient to bring the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. This concentration was determined by General Electric using the approved methods described in GESTAR II. The existing design of the SLC system is sufficient to handle enriched sodium pentaborate solution, which is chemically and physically similar to the current solution. Correct enrichment is ensured by the addition of a new SR 3.1.7.8 to verify sodium pentaborate enrichment prior to addition to the SLC tank. The existing 5° F margin to the saturation temperature specified in the bases is maintained. The existing SLC system design requires that SLC inject a quantity of boron that includes an additional 25% above that needed for an in-vessel boron concentration of 660 ppm. This additional 25% is injected to compensate for imperfect mixing, leakage, and volume in other small piping connected to the reactor. This margin will be maintained such that an additional 25% above that needed for an in-vessel boron concentration equivalent to 720 ppm natural boron will also be injected. The minimum sodium pentaborate concentration of 8.5 weight percent, proposed by this amendment request, ensures that the SLC system continues to meet its post-LOCA suppression pool pH control design function. The change to SR 3.1.7.3 is conservative in nature and is consistent with both the current Bases for SR 3.1.7.3 and plant operating practice. Required verification of the discharge as well as suction piping temperature provides additional assurance of system operability.

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: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Kahtan Jabbour, Acting.

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

Date of amendment request: July 16, 2002.

Description of amendment request: Energy Northwest is requesting changes to the technical specifications (TS) to change the specified minimum emergency diesel generator (DG) steady state output voltage from 3740 volts to 3910 volts.

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

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

The proposed license amendment is administrative and does not involve any design changes or physical changes to plant equipment. The ability of the DGs to perform their safety functions to mitigate consequences is not affected and will continue to be demonstrated in the same manner. Therefore the proposed amendment will not affect the probability or consequences of an accident previously evaluated.

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

The proposed license amendment does not involve changes to plant equipment and the DGs will continue to perform to their required parameters in the same manner. Because the performance of the DGs will remain unchanged, this proposed amendment does not present the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendment is solely a request to revise the Technical Specifications regarding minimum steady state DG output voltage requirements. This change would not affect any operating parameter or equipment performance. Because this proposed amendment would not affect operation, the margin of safety maintained by Columbia would remain unchanged.

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

Attorney for licensee: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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: July 8, 2002.

Description of amendment request: The proposed amendments would change Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF–11 and NPF–18. Specifically, the proposed change would add two footnotes to TS Table 3.3.8.1–1, ''Loss of Power Instrumentation,'' Functions 1.e and 2.e, ''Degraded Voltage—Time Delay, LOCA,'' and makes an editorial change to the heading of TS Table 3.3.8.1–1. The Degraded Voltage—Time Delay, LOCA, function is currently required to be OPERABLE during plant configurations when the emergency core cooling system (ECCS) instrumentation that generates the loss-of-coolant accident (LOCA) signal is not required to be OPERABLE. The proposed changes correct this inconsistency by adding two new footnotes to TS Table 3.3.8.1–1 that modify the required OPERABILITY of the Degraded Voltage—Time Delay, LOCA, function.

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:

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

The TS Table 3.3.8.1–1 Function column heading change to add the reference to the Opposite Unit Division 2 is an editorial change. It was always the intent and practice of LaSalle County Station to apply TS requirements from this column to the Opposite Unit Division 2 4.16 kV emergency bus.

The operation of the Degraded Voltage—Time Delay, LOCA, function is not a precursor to any accident previously evaluated. Thus, the proposed changes to modify the OPERABILITY of the Degraded Voltage—Time Delay, LOCA, function to be consistent with the OPERABILITY of the ECCS instrumentation that generate the timer initiating LOCA signal do not have any affect on the probability of an accident previously evaluated.

Successful operation of the required safety functions of the ECCS is dependent upon the availability of adequate power sources for energizing the various components such as pump motors, motor operated valves, and the associated control components. Offsite power is the preferred source of power for the 4.16 kV emergency buses. The Degraded Voltage—Time Delay, LOCA, function does provide assurance that the ECCS will perform as designed by initiating the disconnect of the 4.16 kV emergency buses from the offsite power sources and connected to the onsite DG power sources, if it is determined that insufficient offsite voltage is available. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

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

The proposed changes to modify the OPERABILITY of the Degraded Voltage—Time Delay, LOCA, function to be consistent with the OPERABILITY of the ECCS instrumentation that generate the timer initiating LOCA signal, will not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed changes do not introduce any new equipment, modes of system operation or failure mechanisms.

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

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

The Degraded Voltage Time Delay circuitry is composed of two time delay components. Upon detection of a degraded voltage condition, the Degraded Voltage—Time Delay, No LOCA, function timer is initiated with a TS Allowable Value of ≥ 270.1 seconds and ≤ 329.9 seconds. If a coincident LOCA signal is detected, the Degraded Voltage—Time Delay, No LOCA, function timer is bypassed and the Degraded Voltage—Time Delay, LOCA, function timer is initiated. The Degraded Voltage—Time Delay, LOCA, function timer has a TS Allowable Value of ≥ 9.4 seconds and ≤ 10.9 seconds. The Time Delay Allowable Values are long enough to provide time for the offsite power supply to recover to normal voltages, but short enough to ensure that sufficient power is available to the required equipment. The shorter time delay associated with a coincident LOCA signal is required to ensure that the ECCS injection assumptions of the LOCA analyses are met. The proposed changes do not affect the Time Delay Allowable Values.

The Drywell Pressure—High instrumentation is required to be OPERABLE in MODES 1, 2 and 3. In MODES 4 and 5, the Drywell Pressure—High instrumentation is not required to be OPERABLE since there is insufficient energy in the reactor to pressurize the drywell to the Drywell Pressure—High setpoint.

The Reactor Vessel Water Level—Low Low Low, Level 1 and Reactor Vessel Water Level—Low Low, Level 2 ECCS instrumentation is required to be OPERABLE in MODES 1, 2, 3 and 4. In MODE 5, the ECCS instrumentation is required to be OPERABLE except with the spent fuel storage pool gates removed and the water level ≥ 22 feet over the top of the reactor pressure vessel flange. In this situation, the water level provides sufficient coolant inventory to allow operator action to terminate the inventory loss prior to fuel uncovery in case of an inadvertent draindown.

The Drywell Pressure—High, Reactor Vessel Water Level—Low Low Low, Level 1 and the Reactor Vessel Water Level—Low Low, Level 2 ECCS instrumentation are not required to be OPERABLE when not in MODES 1, 2, 3, 4, and 5 (i.e., no fuel in the vessel).

The proposed changes will modify the OPERABILITY of the Degraded Voltage—Time Delay, LOCA, function to be consistent with the OPERABILITY of the above described ECCS instrumentation that generate the timer initiating LOCA signal. Thus, the proposed changes are consistent with the ECCS injection assumptions of the LOCA analyses.

The Degraded Voltage—Time Delay, No LOCA, function provides adequate protection to ensure that other required systems powered from the diesel generators (DGs) function as designed in any non-LOCA accident in which a loss of power is assumed when the Degraded Voltage—Time Delay, LOCA, function is not required to be OPERABLE.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mr. Edward J. Cullen, Deputy General Counsel, Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Anthony J. Mendiola.

Florida Power and Light Company, et al. (FPL), Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: July 18, 2002.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) regarding Engineered Safety Feature Actuation System instrumentation. Specifically, they would limit the period of time that inoperable recirculation actuation signal (RAS), containment spray actuation signal (CSAS), and auxiliary feedwater actuation signal (AFAS) input channels could be in the bypassed and/or tripped condition. Generally, the proposed TS employ a 48-hour completion time to restore an inoperable channel, which, in most cases, is more restrictive than the existing TS, is comparable to the value used in the Standard TS for Combustion Engineering plants, and is a reasonable expected repair time based on plant maintenance history.

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

No, facility operation under the new Technical Specification (TS) restrictions would not increase the probability of occurrence of any accident previously evaluated. The proposed changes only affect the emergency safety features actuation system (ESFAS) functions of RAS, CSAS, and AFAS; generally limiting the time that any instrument channel may be inoperable in a bypassed or tripped condition. No physical plant changes are proposed in conjunction with these revisions. The proposed changes to RAS and AFAS channel operability greatly reduce the time that actuation systems are vulnerable to spurious, inadvertent actuation. The proposed changes do allow a new unlimited time for trip of one CSAS channel on Unit 1. Unit 2 already contains provision for the indefinite single channel trip of CSAS, and this change will also make the two units similar. Additionally, it is important to note that inadvertent actuation of any of these functions (RAS, CSAS, or AFAS) during plant operation is not an accident initiating event. Therefore, with no physical effects on the plant and no increase in probability that the subject ESFAS functions will initiate an accident, there is no increased probability that any previously evaluated accident will occur. The changes provided in this safety evaluation do not affect the assumptions or results of any accident evaluated in the UFSAR [Updated Final Safety Analysis Report].

Likewise, the consequences of any accident previously evaluated have not been increased. The proposed changes, by limiting the time that ESFAS functions are inoperable, will increase the reliability of the associated ESFAS functions to respond to accidents. In particular, the revision to the RAS TS will limit the time that the RAS will be vulnerable to single failure and will therefore improve the system reliability during an accident. As these proposed changes constitute no physical change to the facility and only serve to increase ESFAS function reliability, FPL concludes that the consequences of previously evaluated accidents are not increased. The ability of the ESFAS to respond to accident conditions as assumed in any accident analysis has not been affected.

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

No, the proposed activity does not create the possibility of an accident of a different type than any previously evaluated. The proposed changes only affect the ESFAS functions of RAS, CSAS, and AFAS; generally limiting the time that any instrument channel may be inoperable in a bypassed or tripped condition. No physical plant changes are proposed in conjunction with these revisions. Thereby, the proposed changes do not create any new equipment interfaces, equipment response characteristics, or operating configurations. Without creation of a new interaction of materials, operating configuration, or operating interface, there is no possibility that the proposed changes can introduce a new or different kind of accident.

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

The margin of safety as defined in the basis for any Technical Specification or in any licensing document has not been reduced. Except for the change in end state specified for the AFAS automatic actuation logic LCO [Limiting Condition for Operation], the TS Bases for the associated ESFAS LCO do not explicitly discuss a related margin of safety. Changing the AFAS automatic actuation logic LCO end state from Mode 5 to Mode 4 is not a reduction in a margin of safety. That proposed change is consistent with the TS applicability for the AFAS and auxiliary feedwater systems as well as the bases for TS LCOs. Additionally, by virtue of the increased ESFAS reliability provided by the proposed amendments, it is evident that the margin of safety will not be reduced in any manner.

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

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Kahtan N. Jabbour, Acting.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of amendment request: June 24, 2002.

Description of amendment request: The proposed amendment would revise Surveillance Requirement (SR) 3.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period would be extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less* * *'' to ''* * *up to 24 hours or up to the limit of the specified Frequency, whichever is greater* * *.'' In addition, the following requirement would be added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

The NRC staff issued a notice of opportunity for comment in the **Federal**

Register on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 28, 2001 (66 FR 49714). The licensee affirmed the applicability of the following NSHC determination in its application dated June 24, 2002. The NSHC determination is restated below.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC is presented below:

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

The proposed change relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, 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 Previously Evaluated

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. A missed surveillance will not, in and of itself, introduce new failure modes or effects and any increased chance that a standby system might fail to perform its safety function due to a missed surveillance would not, in the absence of other unrelated failures, lead to an accident beyond those previously evaluated. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The extended time allowed to perform a missed surveillance does not result in a significant reduction in [a] margin of safety. As supported by the historical data, the likely outcome of any surveillance is verification that the LCO [Limiting Condition for Operation] is met. Failure to perform a surveillance within the prescribed frequency does not cause equipment to become inoperable. The only effect of the additional time allowed to perform a missed surveillance on [a] margin of safety is the extension of the time until inoperable equipment is discovered to be inoperable by the missed surveillance. However, given the rare occurrence of inoperable equipment, and the rare occurrence of a missed surveillance, a missed surveillance on inoperable equipment would be very unlikely. This must be balanced against the real risk of manipulating the plant equipment or condition to perform the missed surveillance. In addition, parallel trains and alternate equipment are typically available to perform the safety function of the equipment not tested. Thus, there is confidence that the equipment can perform its assumed safety function.

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

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: July 17, 2002.

Description of amendment request: The proposed amendment would change the Unit 2 Technical Specifications (TSs) by including the Unit 2 Cycle 12 (U2C12) Minimum Critical Power Ratio (MCPR) Safety Limits in Section 2.1.1.2, changing the references listed in Section 5.6.5.b, and changing the design features in Section 4.2.1.

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?

No: The proposed change to the MCPR Safety Limits does not directly or indirectly affect any plant system, equipment, component, or change the processes used to operate the plant. Further, the U2C12 MCPR Safety Limits are generated using NRC approved methodology and meet the applicable acceptance criteria. Thus, this proposed amendment does not involve a significant increase in the probability of occurrence of an accident previously evaluated.

Prior to the startup of U2C12, licensing analyses are performed (using NRC approved methodology referenced in Technical Specification Section 5.6.5.b) to determine changes in the critical power ratio as a result of anticipated operational occurrences. These results are added to the MCPR Safety Limit values proposed herein to generate the MCPR operating limits in the U2C12 [Core Operating Limits Report] COLR. These limits could be different from those specified for the U2C11 COLR. The COLR operating limits thus assure that the MCPR Safety Limit will not be exceeded during normal operation or anticipated operational occurrences, thus providing the required level of protection for the fuel rod cladding. Postulated accidents are also analyzed prior to the startup of U2C12 and the results shown to be within the NRC approved criteria. The proposed change to the MCPR Safety Limit will have a negligible impact on the results of these accident analyses.

The U2C12 reload fuel bundles will utilize a small amount of depleted uranium (''tails'') in certain fuel rods, in addition to natural and slightly enriched uranium. There is no change to the composition of the fuel pellets containing tails material, (*i.e.,* UO2) except a slight decrease in the amount of [uranium-235] U_{235}. Therefore, the use of depleted uranium (''tails'') in the fuel rods does not affect the mechanical performance of the fuel rods. The impact of the use of tails on core performance is included in the reload licensing analysis.

The changes to the references in Section 5.6.5.b were made to properly reflect the NRC approved methodology used to generate the U2C12 core operating limits. The use of this approved methodology does not increase the probability of occurrence or consequences of an accident previously evaluated.

Therefore, this proposed amendment 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 analyzed?

No: The change to the MCPR Safety Limits and the U2C12 core loading which it supports does not directly or indirectly affect any plant system, equipment, or component (other than the core itself) and therefore does not affect the failure modes of any of these. Thus, the proposed [change does] not create the possibility of a previously unevaluated operator error or a new single failure.

The use of depleted uranium (''tails'') in the fuel rods does not affect the mechanical performance of the fuel rods.

The changes to the references in Section 5.6.5.b were made to properly reflect the NRC approved methodology used to generate the U2C12 core operating limits. The use of this approved methodology does not create the possibility of a new or different kind of accident.

Therefore, this proposed amendment does not involve a possibility of a new or different kind of accident from any accident previously analyzed.

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

No: Since the proposed [change does] not alter any plant system, equipment, component, or the processes used to operate the plant, the proposed change will not jeopardize or degrade the function or operation of any plant system or component governed by Technical Specifications. The proposed MCPR Safety Limits do not involve a significant reduction in the margin of safety as currently defined in the Bases of the applicable Technical Specification sections, because the MCPR Safety Limits calculated for U2C12 preserve the required margin of safety.

The use of depleted uranium (''tails'') in the fuel rods does not affect the mechanical performance of the fuel rods.

The changes to the references in Section 5.6.5.b were made to properly reflect the NRC approved methodology used to generate the U2C12 core operating limits. This approved methodology is used to demonstrate that all applicable criteria are met, thus, demonstrating that there is no reduction in the margin of safety.

Therefore, these changes do not involve a significant reduction in [the] margin of safety.

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

Attorney for licensee: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: July 18, 2002.

Description of amendment request: The licensee proposes to change Salem Technical Specifications (TSs) requirements associated with its containment spray nozzles. The frequency of TS Surveillance Requirement (SR) 4.6.2.1.d for verifying that the containment spray nozzles are unobstructed would be changed from a fixed 10-year frequency to after activities that could result in nozzle blockage. PSEG proposes to either evaluate the work performed to determine the impact to the containment spray system, or perform an air or smoke flow test.

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

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

Response: No.

The proposed change revises the testing requirements for the containment spray nozzles to only require verification that each spray nozzle is unobstructed following activities that could result in nozzle blockage. The proposed change does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. No active or passive failure mechanisms that could lead to an accident are affected. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. The containment spray system is not an accident initiator but is used for mitigation of design basis accidents. As a result, the probability of any accident previously evaluated, is not significantly increased.

The consequences of a previously evaluated accident are not significantly increased. The proposed change revises the current Surveillance Frequency from 10 years to following activities that could result in spray nozzle blockage. Since activities that could introduce foreign material into the system (such as inadvertent actuation of the containment spray system or loss of foreign material control) are the most likely cause for obstruction, testing or inspection following such activities would verify that the nozzle(s) are unobstructed, and the system is capable of performing its safety function. No other evolutions require the system boundary to be breached, so introduction of debris during times when maintenance activities are not in progress are precluded. Introduction of foreign materials into the system from the exterior is highly unlikely due to the location of the spray headers, the passive nature of the nozzles, and the fact that the stainless steel containment spray headers are maintained dry which does not lend itself to active degradation mechanisms such as corrosion. The proposed testing requirements are considered sufficient to provide a high degree of confidence that containment spray will function when required.

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

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

Response: No.

The proposed change to the test frequency for the containment spray system nozzles does not involve the use or installation of new equipment. Installed equipment is not operated in a new or different manner. No new or different system interactions are created, and no new processes are introduced. The current foreign material exclusion practices have been reviewed and judged sufficient to provide high confidence that debris will not be introduced during times when the system boundary is breached.

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

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

Response: No.

The revision to the containment spray nozzle testing frequency does not introduce any new setpoints at which protective or mitigative actions are initiated. No current setpoints are altered by this change. The design and functioning of the containment spray system is unchanged. Since the system is not susceptible to corrosion induced obstruction nor is the introduction of foreign material from the exterior likely, the proposed testing frequency is sufficient to provide high confidence that the containment spray system will be available to provide the flow necessary to mitigate the consequences of a design basis accident. Therefore, the capability of the system will remain unchanged. As a result, this change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: Jacob I. Zimmerman, Acting.

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

Date of amendment request: July 29, 2002.

Description of amendment request: The proposed change to the Technical Specifications (TSs) would revise the requirements for containment closure associated with the equipment hatch and personnel airlocks during Core Alterations and movement of irradiated fuel within the containment. This proposed change would allow the equipment hatch and the personnel airlocks to remain open during fuel movement in the containment provided administrative controls are developed and implemented, ensuring the closure of the equipment hatch and personnel airlock following a fuel handling accident within the containment building. In addition, the associated TS Bases are revised.

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.

An alternate source term calculation has been performed for Salem Nuclear Station that demonstrates that offsite and control room dose consequences of a postulated fuel handling accident remain within the limits provided sufficient decay has occurred prior to the movement of irradiated fuel without taking credit for certain mitigation features such as ventilation filter systems and containment closure. Fuel movement is allowed provided that irradiated fuel has undergone the required decay time.

The proposed amendment would allow movement of sufficiently decayed irradiated fuel within the containment building with the equipment hatch and personnel air locks open provided that administrative controls are implemented to promptly (within 1 hour) close the containment penetrations.

Either the Containment Purge system or the Auxiliary Building Ventilation System with suction from the containment atmosphere, with associated radiation monitoring will be available whenever movement of irradiated fuel is in progress in the containment building and the equipment hatch is open. If for any reason, this ventilation requirement can not be met, movement of fuel assemblies within the containment building shall be discontinued until the flow path(s) can be reestablished or close the equipment hatch and personnel airlocks. The amendment also would allow movement of irradiated fuel assemblies within the Fuel Handling Building with the Fuel Handling Area Ventilation System (FHAVS) in operation but no credit taken for filtration.

This amendment does not alter the methodology of the FHA [Fuel-Handling Accident] or equipment used directly in fuel handling operations. Neither ventilation filter systems, the CPES [Containment Purge Exhaust System] nor the FHAVS, is used to actually handle fuel. Therefore neither of these systems is an ''accident initiator''. Similarly, neither the equipment hatch, the personnel air locks, nor any other containment penetration, nor any component thereof is an accident initiator.

In the postulated Fuel Handling Accident, the revised dose calculations, performed using 10 CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term, do not take credit for automatic containment purge isolation thus allowing for continuous monitoring of containment activity until containment closure is achieved. If required, containment purge isolation can be initiated manually from the control room.

Actual fuel handling operations are not affected by the proposed changes. Therefore, the probability of a Fuel Handling Accident is not affected with the proposed amendment. No other accident initiator is affected by the proposed changes.

The FHA in the Fuel Handling Building has been analyzed without credit for filtration by the FHAVS. The analyses of these design basis events were conducted with the Alternative Source Term Methodology in accordance with 10 CFR 50.67 and Regulatory Guide 1.183. These analyses show that the resultant radiation doses are within the limits specified in these documents.

The TEDE [Total Effective Dose Equivalent] radiation doses from the analyses supporting this LCR [License Change Request] have been compared to equivalent TEDE radiation doses estimated with the guidelines of R.G. [Regulatory Guide] 1.183. The new values are shown to be within the regulatory guidelines.

The revision to the definition of Core Alterations simply reflects the definition in the Standard Technical Specifications, NUREG 1431 for Westinghouse Plants and is supported by the bounding effects of the Fuel Handling Accident analysis.

The deletion of Core Alterations from the APPLICABILITY section of the affected LCO's [Limiting Conditions for Operation] is based on the fact that, during Core Alterations only, the FHA results in cladding damage and potential radiological release. Consequently, the deletion of Core Alterations is consistent with industry approved practice and guidance documents (ex: TSTF [Technical Specification Task Force]-51, revision 2).

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

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

Response: No.

The proposed change does not involve addition or modification to any plant system, structure, or component. The proposed amendment would permit the equipment hatch and personnel air locks to be open during movement of irradiated fuel. The proposed amendment does not involve any change in the operation of these containment penetrations. Having these penetrations open does not create the possibility of a new accident.

The proposed amendment also would remove the requirements for operability of the FHAVS Filtration System during movement of sufficiently decayed irradiated fuel. It does not alter the operation of these systems. Therefore, the system is not an accident initiator. Modification of the requirements of operability for the system from the plant Technical Specifications does not create the possibility of a new accident.

The revision to the definition of Core Alterations simply reflects the industry position supported by the definition in the Standard Technical Specifications, NUREG 1431 for Westinghouse Plants and is supported by the bounding effects of the Fuel Handling Accident analysis.

The deletion of Core Alterations from the APPLICABILITY section of the affected LCO's is based on the fact that, during Core Alterations only, the FHA results in cladding damage and potential radiological release. Consequently, the deletion of Core Alterations is consistent with industry approved practice and guidance documents (ex: TSTF–51, revision 2).

The proposed amendment does not create the possibility of a new or different kind of accident than any previously evaluated.

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

Response: No.

The assumptions and input used in the analysis are conservative as noted below. The design basis Fuel Handling Accidents have been defined to identify conservative conditions. The source term and radioactivity releases have been calculated pursuant to Regulatory Guide 1.183 and with conservative assumptions concerning prior reactor operation. The control room atmospheric dispersion factors have been calculated with conservative assumptions associated with the release. The conservative assumptions and input noted above ensure that the radiation doses cited in this License Change Request are the upper bound to radiological consequences of a Fuel Handling Accident either in Containment or the Fuel Handling Building. The analyses show that there is a significant margin between the TEDE radiation doses calculated for the postulated Fuel Handling Accident using the Alternative Source Term and the acceptance limits of 10 CFR 50.67 and Regulatory Guide 1.183.

The revision to the definition of Core Alterations simply reflects the industry position supported by the definition in the Standard Technical Specifications, NUREG 1431 for Westinghouse Plants and is supported by the bounding effects of the Fuel Handling Accident analysis.

The deletion of Core Alterations from the APPLICABILITY section of the affected LCO's is based on the fact that, during Core Alterations only the FHA results in cladding damage and potential radiological release. Consequently, the deletion of Core Alterations is consistent with industry approved practice and guidance documents (ex: TSTF–51, revision 2).

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: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: Jacob Zimmerman, Acting.

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

Date of amendment request: March 20, 2002.

Description of amendment request: This proposed change revises the iodine dose conversion factors used in the determination of the dose equivalent I–131 reactor coolant specific activity and in the calculation of the offsite radiological consequences for those Final Safety Analysis Report (FSAR) Chapter 15 accidents that include iodine spiking effects.

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?

This proposed change revises the iodine dose conversion factors used in the determination of the dose equivalent I–131 reactor coolant specific activity and in the calculation of the offsite radiological consequences for those FSAR Chapter 15 accidents that include iodine spiking effects. The iodine dose conversion factors are changed from the values in TID [Technical Information Document] 14844 to the values in ICRP [International Commission on Radiological Protection] 30, consistent with NUREG–1431. The accidents affected by this change are the steam generator tube rupture, main steam line break and CVCS [Chemical and Volume Control System] line rupture. The proposed change also revises certain input assumptions (letdown demineralize iodine removal efficiency, primary coolant leakage and uncertainty in letdown flow) used in determining the accident initiated (concurrent) iodine spiking source terms input to the offsite radiological consequences calculations. The change in dose conversion factors and the input assumptions does not affect any normal operation or accident scenarios. There are no changes to any plant procedures or equipment that would relate to the probability of an accident. The change in the iodine spiking input assumptions identified in NSAL [Nuclear Safety Advisory Letter]-00–04 results in an increase in the calculated offsite dose consequences for the steam generator tube rupture, main steam line break and CVCS line rupture. Use of the ICRP 30 iodine dose conversion factors offsets this increase such that the resulting calculated offsite dose consequences are less severe than those previously presented in the FSAR for the steam generator tube rupture, main steam line break and CVCS line rupture. * * *

Thyroid doses for the other accidents described in the FSAR will continue to be reported using the conservative TID 14844 iodine dose conversion factors until a future update is required.

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

The proposed changes to Technical Specification [TS] 1.10, Definitions, Dose Equivalent I–131 and the use of the iodine spiking input assumptions listed in NSAL–00–04 [do] not introduce any new accident initiator mechanisms. The dose conversion factors are used in determining the reactor coolant dose equivalent I–131 specific activity and in the calculation of offsite dose consequences for certain design basis accidents which include the effects of iodine spiking as revised based on the iodine spiking input changes (letdown demineralizer iodine removal efficiency, primary coolant leakage and uncertainty in letdown flow) provided in NSAL–00–04. No existing accident scenarios are affected and no new scenarios are created. The proposed change does not introduce alterations to system operations, changes to equipment operability or technical specification operability requirements, nor to Engineered Safety Features Actuation System instrumentation or setpoints. The proposed change does not revise any of the actual equipment or instrumentation in the plant nor does it change the actual alarm setpoints or information available to the operators to monitor Technical Specification commitments. It does not introduce any new or different failure mechanisms or limiting single failures. A new or different kind of accident is thus not created.

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

The proposed change to Technical Specification 1.10, Definitions, Dose Equivalent I–131 preserves the conclusions of plant safety analyses presented in the FSAR. This proposed change revises the iodine dose conversion factors used in the calculation of the potential offsite radiological consequences following those Chapter 15 accidents that include iodine spiking effects as revised based on the iodine spiking input changes provided in NSAL–00–04. The dose conversion factors are changed from the values in TID–14844 to the values in ICRP 30, consistent with the criteria in NUREG–1431. This activity relates to TS section B3/4.4.8 and TS section 1.10 Dose Equivalent I–131. TS section B3/4.4.8 states that the limitation on the specific activity of the primary coolant ensures that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident. TS Section 1.10 defines the acceptable values for the iodine dose conversion factors. The change in the accident initiated iodine spiking calculation input parameters identified in NSAL–00–04 results in an increase in the calculated offsite dose consequences for the steam generator tube rupture, main steam line break and CVCS line rupture. Use of the ICRP 30 iodine dose conversion factors offsets this increase such that the resulting calculated offsite dose consequences are less severe than those previously presented in the FSAR. Therefore, the margin of safety is not reduced.

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

Attorney for licensee: Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: John A. Nakoski.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348, Joseph M. Farley Nuclear Plant, Unit 1, Houston County, Alabama

Date of amendment request: March 4, 2002, as supplemented by letter dated July 11, 2002.

Description of amendment request: The proposed amendment would revise Technical specifications (TS) 5.5.9.3.a, ''Steam Generator Tube Surveillance Program, Inspection Frequencies.'' Specifically, the proposed changes would revise the Farley Nuclear Plant, Unit 1 TS to allow a 40 month inspection interval after its first (post-replacement) inservice inspection, rather than after two consecutive inspections resulting in C–1 classification.

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?

The proposed one-time change revises the steam generator (SG) inspection interval requirements in Technical Specification (TS) 5.5.9.3, ''Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies,'' for the Farley Nuclear Plant, Unit 1, Spring 2003 refueling outage, to allow a 40 month inspection frequency after one inspection, rather than after two consecutive inspections with results that are within the C–1 category. C–1 category is defined as ''less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.''

The proposed one-time extension of the Unit 1 SG tube inservice inspection interval does not involve changing any structure, system, or component, or affect reactor operations. It is not an initiator of an accident and does not change any existing safety analysis previously analyzed in the Farley Nuclear Plants' Final Safety Analysis Report

(FSAR). As such, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

Since the proposed change does not alter the plant design, there is no direct increase in SG leakage. Industry experience indicates that the probability of increased SG tube degradation would not go undetected.

Additionally, steps described below will further minimize the risk associated with this extension. For example, the scope of inspections performed during the last Farley Nuclear Plant, Unit 1, refueling outage (*i.e.,* the first refueling outage following SG replacement) exceeded the TS requirements for the first two refueling outages after SG replacement. That is, more tubes were inspected than were required by TS. Currently, Farley Nuclear Plant, Unit 1, does not have an SG damage mechanism, and will meet the current industry examination guidelines without performing SG inspections during the next refueling outage. Additionally, as part of our SG Program, both a Condition Monitoring Assessment and an Operational Assessment are performed after each inspection and compared to the Nuclear Energy Institute (NEI) 97–06, ''Steam Generator Program Guidelines,'' performance criteria. The results of the Condition Monitoring Assessment demonstrated that all performance criteria were met during the Farley Nuclear Plant, Unit 1, Fall 2001 refueling outage, and the results of the Operational Assessment show that all performance criteria will be met over the proposed operating period. Considering these actions, along with the improved SG design and reliability of Westinghouse replacement SGs, extending the SG tube inspection frequency does not involve a significant increase in the consequences of an accident previously evaluated.

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?

The proposed change revises the SG inspection frequency requirements in TS 5.5.9.3.a, ''Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies,'' for the Farley Nuclear Plant, Unit 1, Spring 2003 refueling outage, to allow a 40 month inspection interval after one inspection, rather than after two consecutive inspections, with inspection results within the C–1 category.

The proposed change will not alter any plant design basis or postulated accident resulting from potential SG tube degradation. The scope of inspections performed during the last Farley Nuclear Plant, Unit 1, refueling outage (*i.e.,* the first refueling outage following SG replacement) significantly exceeded the TS requirements for the scope of the first two refueling outages after SG replacement.

Primary-to-secondary leakage that may be experienced during all plant conditions is expected to remain within current accident analysis assumptions. The proposed change does not affect the design of the SGs, the method of SG operation, or reactor coolant chemistry controls. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. The proposed change involves a one-time extension to the SG tube inservice inspection frequency, and, therefore, will not give rise to new failure modes. In addition, the proposed change does not impact any other plant systems or components.

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?

The SG tubes are an integral part of the Reactor Coolant System (RCS) pressure boundary that are relied upon to maintain the RCS pressure and inventory. The SG tubes isolate the radioactive fission products in the reactor coolant from the secondary system. The safety function of the SGs is maintained by ensuring the integrity of the SG tubes. In addition, the SG tubes comprise the heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system.

SG tube integrity is a function of the design, environment, and current physical condition. Extending the SG tube inservice inspection frequency by one operating cycle will not alter the function or design of the SGs. SG inspections conducted during the first refueling outage following SG replacement demonstrated that the SGs do not have an active damage mechanism, and the scope of those inspections significantly exceeded those required by the TS. These inspection results were comparable to similar inspection results for second generation alloy 690 models of replacement SGs installed at other plants, and subsequent inspections at those plants yielded results that support this extension request. The improved design of the replacement SGs also provides reasonable assurance that significant tube degradation is not likely to occur over the proposed operating period.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: July 25, 2002.

Description of amendment request: The amendment would revise Surveillance Requirements 3.3.1.2 and 3.3.1.3 of Technical Specification 3.3.1, ''Reactor Trip System (RTS) Instrumentation.''

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

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

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The RTS instrumentation will be unaffected. Protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

The probability and consequences of accidents previously evaluated in the USAR [Updated Safety Analysis Report] are not adversely affected because the change to the NIS [Nuclear Instrumentation System] power range channel daily surveillance assures the conservative response of the channel even at part-power levels.

The proposed changes modify the NIS power range channel daily surveillance requirement to assure the NIS power range functions are tested in a manner consistent with the safety analysis and licensing basis.

The proposed changes will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to the normal plant operating parameters or accident mitigation performance.

The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the USAR.

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

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

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This amendment will not affect the normal method of plant operation or change any operating parameters. No performance requirements or response time limits will be affected; however, the proposed TS Bases changes impose explicit NIS power range high trip setpoint adjustment requirements prior to adjusting indicated power in a decreasing power direction. These requirements are consistent with assumptions made in the safety analysis and licensing basis.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment.

This amendment does not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, or Solid State Protection System used in the plant protection systems.

Therefore, the proposed changes do 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 a margin of safety.

The proposed changes require a revision to the criteria for implementation of NIS power range channel adjustments based on secondary power calorimetric calculations; however, the changes do not eliminate any RTS surveillances or alter the frequency of surveillances required by the Technical Specifications. The revision to the criteria for implementation of the daily surveillance will have a conservative effect on the performance of the NIS power range channels, particularly at part-power conditions. The nominal trip setpoints specified in the Technical Specification Bases and the safety analysis limits assumed in the transient and accident analyses are unchanged. None of the acceptance criteria for any accident analysis is changed.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the output power limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor ($F\Delta H$), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

The imposition of appropriate surveillance testing requirement will not reduce any margin of safety since the changes will assure that safety analysis assumptions on equipment operability are verified on a periodic frequency.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: July 25, 2002.

Description of amendment request: The amendment would revise Chapter 5.0, ''Administrative Controls,'' of the technical specifications (TSs) to allow the use of generic personnel titles in the TSs in place of plant-specific personnel titles.

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

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

The proposed changes do not affect accident initiators or assumptions. The radiological consequences of an accident previously evaluated remain unchanged. These changes involve administrative changes concerning the use of personnel titles and do not affect responsibilities or qualifications of plant personnel.

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

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

The proposed changes are administrative in nature. As such, there are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This amendment will not affect the normal method of plant operation or change any operating parameters. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this amendment. There will be no adverse effects or challenges imposed on any safety-related system as a result of this amendment.

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

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

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. The use of generic personnel titles will not reduce any margin of safety. (These changes involve administrative changes concerning the use of personnel titles and do not affect responsibilities or qualifications of plant personnel.)

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

Previously Published Notices of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the **Federal Register** on the day and page cited. This notice does not extend the notice period of the original notice.

South Carolina Electric & Gas, Docket No. 50–395, Virgil C. Summer Nuclear Station, Fairfield County, South Carolina

Date of amendment request: July 24, 2001, as supplemented April 4, 2002, May 7, 2002, June 17, 2002, July 2, 2002, July 15, 2002, and July 25, 2002.

Brief description of amendment request: This amendment would increase the spent fuel pool storage capacity by replacing all 11 existing rack modules with 12 new high density storage racks. The rerack will increase the storage capacity from 1,276 storage cells to 1,712 storage cells. The degrading Boraflex neutron-absorbing material in the existing racks will be replaced by Boral material that will be used in the new racks.

Date of publication of individual notice in ***Federal Register:*** June 25, 2002 (67 FR 42810).

Expiration date of individual notice: July 25, 2002.

Notice of Issuance of Amendments to Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397-4209, 301–415–4737 or by email to *pdr@nrc.gov.*

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

Date of application for amendments: December 13, 2001, as supplemented by letter dated May 1, 2002.

Brief description of amendments: The amendments add the following to the Technical Specifications: (1) The phrase, ''or if open, capable of being closed,'' to the Limiting Condition for Operation 3.9.3 for the equipment hatch, during core alterations or movement of irradiated fuel assemblies inside containment, and (2) the requirement to verify the capability to close the equipment hatch in a new Surveillance Requirement 3.9.3.3. The amendments allow the equipment hatch to be open in refueling outages during the conditions stated above.

Date of issuance: July 25, 2002.

Effective date: July 25, 2002, and shall be implemented within 90 days of the date of issuance, including the incorporation of the changes to the Technical Specification Bases as described in the licensee's letters dated December 13, 2001, and May 1, 2002.

Amendment Nos.: Unit 1–143, Unit 2–143, Unit 3–143.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 22, 2002 (67 FR 2919). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 25, 2002.

No significant hazards consideration comments received: No.

Calvert Cliffs Nuclear Power Plant, Inc., Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: January 31, 2002.

Brief description of amendments: These amendments correct administrative errors in Section 5.6.5, ''Core Operating Limits Report (COLR),'' of the Technical Specifications and Section 2.0, ''Environmental Protection Issues,'' of the Environmental Protection Program.

Date of issuance: August 6, 2002.

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

Amendment Nos.: 254/231.

Renewed Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 28, 2002 (67 FR 36927). The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated August 6, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 28, 2001.

Brief description of amendment: The amendment revised Technical Specification (TS) 3/4.5.1, ''Safety Injection Tanks (SITs)'' to delete surveillance requirement (SR) 4.5.1.f. This SR provided verification of the automatic opening features of the SIT outlet isolation valves.

Date of issuance: August 7, 2002.

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

Amendment No.: 268.

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

Date of initial notice in ***Federal Register:*** October 31, 2001 (66 FR 55010). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 7, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 26, 2001.

Brief description of amendment: The amendment modifies the Millstone Nuclear Power Station, Unit No. 3 (MP3) Technical Specifications (TSs) to relocate MP3 TSs related to the control rod position indication system requirements for shutdown to the licensee-controlled Technical Requirements Manual (TRM). The Index and Bases pages of the affected TSs will also be modified to address the proposed changes.

Date of issuance: July 30, 2002.

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

Amendment No.: 207.

Facility Operating License No. NPF–49: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 14, 2001 (66 FR 57120). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 30, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 16, 2002.

Brief description of amendments: The amendments revise Surveillance Requirement (SR) 3.0.3 to extend the delay period before entering a Limiting Condition for Operation following a missed surveillance. The delay period is extended from the current limit of '' * * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to '' * * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: August 1, 2002.

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

Amendment Nos.: 201 & 194.

Facility Operating License Nos. NPF–35 and NPF–52: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 14, 2002 (67 FR 34482). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 1, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 16, 2002.

Brief description of amendments: The amendments revise Surveillance Requirement (SR) 3.0.3 to extend the delay period before entering a Limiting Condition for Operation following a missed surveillance. The delay period is extended from the current limit of '' * * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to '' * * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: July 30, 2002.

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

Amendment Nos.: 205/186.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 14, 2002 (67 FR 34483). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 30, 2002.

No significant hazards consideration comments received: No.

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

Date of application of amendments: April 16, 2002.

Brief description of amendments: The amendments revise Surveillance Requirement (SR) 3.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period is extended from the current limit of '' * * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to '' * * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of Issuance: July 30, 2002.

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

Amendment Nos.: 327, 327, 328.

Renewed Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 14, 2002 (67 FR 34483). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 30, 2002.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: June 13, 2002.

Brief description of amendment: The amendment revised Technical Specifications Section 4.13.A, ''Inspection Requirements,'' to allow the use of the optimum eddy current probe size when performing steam generator tube inspections. The amendment also corrects grammatical and typographical errors.

Date of issuance: July 29, 2002.

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

Amendment No.: 230.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 25, 2002 (67 FR 42806). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 29, 2002.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: January 8, 2002.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.1.B, ''Heatup and Cooldown,'' to delete the requirements governing the reactor vessel surveillance program, including the reactor vessel specimen withdrawal schedule. In addition, the changes corrected errors in TS 4.2, ''Inservice Inspection and Testing;'' TS 5.2.C, ''Design Features—Containment;'' and TS 6.4, ''Administrative Controls—Training.'' TS Sections 6.1, ''Responsibility'' and 6.2, ''Organization'' were changed to reflect the organizational changes resulting from the transfer of the operating license to Entergy Nuclear Operations, Inc. on September 6, 2001.

Date of issuance: July 30, 2002.

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

Amendment No.: 231.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 5, 2002 (67 FR 10011). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 30, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 19, 2002.

Brief description of amendments: These amendments allow plant operation to continue if the temperature of the normal heat sink (NHS) exceeds the Technical Specification (TS) limit of 90 °F provided that the NHS temperature averaged over the previous

24-hour period is verified at least once per hour to be less than or equal to 90 °F, and the NHS temperature does not exceed a maximum value of 92 °F. The format for this change had been previously approved by the Nuclear Regulatory Commission for the Standard TSs as per TS Task Force (TSTF) change TSTF–330, Revision 3, ''Allowed Outage Time—Ultimate Heat Sink'', on October 13, 2000. In addition, an administrative change removes references to a temporary TS change which expired on May 31, 2000.

Date of issuance: July 29, 2002.

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

Amendments Nos.: 244/248.

Facility Operating License Nos. DPR–44 and DPR–56: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 14, 2002 (67 FR 34486). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 29, 2002.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–254, Quad Cities Nuclear Power Station, Unit 1, Rock Island County, Illinois

Date of application for amendment: April 8, 2002, as supplemented June 18 and July 3, 2002.

Brief description of amendment: The amendment revises the safety limit minimum critical power ratio for two-loop and single-loop operation.

Date of issuance: July 29, 2002.

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

Amendment No.: 207.

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

Date of initial notice in ***Federal Register:*** May 14, 2002 (67 FR 34487). The supplements dated June 18 and July 3, 2002, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 29, 2002.

No significant hazards consideration comments received: No.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of application for amendment: August 16, 2001, as supplemented by letter dated November 19, 2001.

Brief description of amendment: The amendment revises the license to incorporate a new License Condition 2.B.(9). The license condition terminates license jurisdiction for a portion of the Maine Yankee Atomic Power Station site (referred to as the Non-Impacted Backlands (west of Bailey Cove and west of Young's Brook and north of Old Ferry Road)), thereby releasing these lands from Facility Operating License No. DPR–36.

Date of issuance: July 30, 2002.

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

Amendment No.: 167.

Facility Operating License No. DPR–36: The amendment revised the license.

Date of initial notice in ***Federal Register:*** March 19, 2002 (67 FR 12604). The November 19, 2001, supplemental letter provided additional information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 30, 2002.

No significant hazards consideration comments received: No.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of application for amendment: November 26, 2001, as supplemented on May 20, 2002.

Brief description of amendment: The amendment deleted Section 3/4.2.6, ''Inservice Inspection and Testing,'' revised Section 4.2.7, ''Reactor Coolant System Isolation Valves,'' added a new Section 6.17, ''Inservice Testing Program,'' and deleted several reporting requirements in Section 6.9.3, ''Special Reports.''

Date of issuance: August 5, 2002.

Effective date: August 5, 2002.

Amendment No.: 173.

Facility Operating License No. DPR–63: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 26, 2001 (66 FR 66468). The May 20, 2002, supplemental letter provided clarifying information that was within the scope of the amendment request and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 5, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 8, 2001, as supplemented by letters dated February 4, April 8, May 7, June 6, and June 28, 2002.

Brief description of amendments: These amendments revised TS 3.3.5.1, ''Emergency Core Cooling System Instrumentation,'' by deleting Function 3e, thus preventing the automatic swap of the suction source for the high pressure coolant injection pump from the condensate storage tank to the suppression pool on high suppression pool level.

Date of issuance: August 5, 2002.

Effective date: As of the date of issuance and shall be implemented within 30 days of their associated plant modifications, and no later than December 31, 2002.

Amendment Nos.: 204/178.

Facility Operating License Nos. NPF–14 and NPF–22: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** October 3, 2001 (66 FR 50471). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 5, 2002.

No significant hazards consideration comments received: No.

STP Nuclear Operating Company, Docket No. 50–498, South Texas Project, Unit 1, Matagorda County, Texas

Date of amendment request: January 28, 2002, as supplemented by letters dated June 20 and July 3, and 30, 2002. The supplemental information provided clarification that did not change the scope or the initial no significant hazards consideration determination.

Brief description of amendment: The amendment revises Technical Specification (TS) 4.4.5.3a, ''Steam Generator Surveillance Requirements.'' Specifically, the changes would revise the South Texas Project Unit 1 TS to a 40-month inspection interval after its first (post-replacement) inservice inspection rather than two consecutive inspection resulting in C–1 classification.

Date of issuance: July 31, 2002.

Effective date: July 31, 2002.

Amendment No.: 140.

Facility Operating License No. NPF–76: The amendment revises the Technical Specification.

*Date of initial notice in **Federal Register:*** March 19, 2002 (67 FR 12607). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 31, 2002.

No significant hazards consideration comments received: No.

Vermont Yankee Nuclear Power Corporation, Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: October 5, 2001, as supplemented on November 7 and 8, 2001, and January 23 and April 30, 2002.

Brief description of amendment: The amendment revises the license and technical specifications to reflect changes related to the transfer of the license for the Vermont Yankee Nuclear Power Station, previously held by Vermont Yankee Nuclear Power Corporation, to Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc.

Date of Issuance: July 31, 2002.

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

Amendment No.: 208.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications and License.

*Date of initial notice in **Federal Register:*** December 7, 2001 (66 FR 63566). The letters dated January 23 and April 30, 2002, provided clarifying information and did not expand the application beyond the scope of the notice or affect the applicability of the Commission's generic no significant hazards consideration determination pursuant to 10 CFR 2.1315.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated July 31, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: February 21, 2002.

Brief description of amendment: The amendment revises several of the Required Actions in the technical specifications that require suspension of operations involving positive reactivity additions or suspension of operations involving reactor coolant system (RCS) boron concentration reductions. In addition, the proposed amendment revises several Limiting Condition for Operation (LCO) Notes that preclude reductions in RCS boron concentration. This amendment revises these Required Actions and LCO Notes to allow small, controlled, safe insertions of positive reactivity, but limits the introduction of positive reactivity such that compliance with the required shutdown margin or refueling boron concentration limits will still be satisfied. This amendment is based on an NRC-approved traveler, Technical Specification Task Force (TSTF)–286, Revision 2.

Date of issuance: July 29, 2002.

Effective date: July 29, 2002, and shall be implemented within 60 days of the date of issuance.

Amendment No.: 145.

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

*Date of initial notice in **Federal Register:*** April 16, 2002 (67 FR 18650). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated: July 29, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 7, 2001, as supplemented by letter dated February 20, 2002.

Brief description of amendment: The amendment revises Limiting Condition for Operation 3.9.4 to allow the equipment hatch to be open during core alterations or movement of irradiated fuel assemblies inside containment, and adds the requirement to verify the capability to install the equipment hatch in a new Surveillance Requirement 3.9.4.2. The existing SR 3.9.4.2 would be renumbered SR 3.9.4.3, but would otherwise not be changed.

Date of issuance: July 30, 2002.

Effective date: July 30, 2002, and shall be implemented within 6 months of the date of issuance, including the incorporation of changes to the Technical Specification Bases as described in licensee's application dated August 7, 2001, and supplemental letter dated February 20, 2002.

Amendment No.: 146.

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

Date of initial notice in **Federal Register:** September 5, 2001 (66 FR 46482). The supplemental letter dated February 20, 2002, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 30, 2002.

No significant hazards consideration comments received: No.

Notice of Issuance of Amendment to Facility Operating License and Final No Significant Hazards Consideration Determination

During the period since publication of the last biweekly notice, individual notices of issuance of amendments have been issued for the facilities as listed below. These notices were previously published as separate individual notices. They are repeated here because this biweekly notice lists all amendments that have been issued for which the Commission has made a final determination that an amendment involves no significant hazards consideration.

In this case, a prior Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing was issued, a hearing was requested, and the amendment was issued before any hearing because the Commission made a final determination that the amendment involves no significant hazards consideration.

Details are contained in the individual notice as cited.

Entergy Nuclear Indian Point 2, LLC, et al., Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: July 13, 2001, as supplemented November 30, 2001, March 13, April 3, May 30, and June 13, 2002.

Brief description of amendment: The amendment made a one-time only change to the Technical Specification Surveillance Requirement 4.4.A.3 to revise the frequency for the containment integrated leak rate test (ILRT, Type A test) from at least once per 10 years to once per 15 years. This change applies only to the interval following the last Type A test that was performed satisfactorily in June 1991 at IP2.

Date of issuance: August 5, 2002.

Amendment No.: 232.

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

Facility Operating License No. DPR–26: Amendment revise the technical specifications.

Date of individual notice in ***Federal Register:*** August 22, 2001 (66 FR 44165). The November 30, 2001, March 13, April 3, May 30, and June 13, 2002, letters provided clarifying information that did not expand the application beyond the scope of the initial notice or change the initial proposed no significant hazards consideration determination.

Dated at Rockville, Maryland, this 9th day of August 2002.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

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BILLING CODE 7590–01–P

COMMISSION ON OCEAN POLICY

Public Meeting

AGENCY: Commission on Ocean Policy.

ACTION: Notice.

SUMMARY: The U.S. Commission on Ocean Policy will hold its ninth and final regional meeting, the Commission's eleventh public meeting, to hear and discuss issues of concern to the Great Lakes region.

DATES: Public meetings will be held Tuesday, September 24, 2002 from 8:30 a.m. to 6 p.m. and Wednesday, September 25, 2002 from 8:30 a.m. to 5 p.m.

ADDRESSES: The meeting location is the Phelps Auditorium, John G. Shedd Aquarium, 1200 South Lake Shore Drive, Chicago, IL 60605. (Please use the Group Entrance located on the South side of the John G. Shedd Aquarium.)

FOR FURTHER INFORMATION CONTACT: Terry Schaff, U.S. Commission on Ocean Policy, 1120 20th Street, NW., Washington, DC, 20036, 202–418–3442, *schaff@oceancommission.gov.*

SUPPLEMENTARY INFORMATION: This meeting is being held pursuant to requirements under the Oceans Act of 2000 (Public Law 106–256, Section 3(e)(1)(E)). The agenda will include presentations by invited speakers representing local and regional government agencies and non-governmental organizations, comments from the public and any required administrative discussions and executive sessions. Invited speakers and members of the public are requested to submit their statements for the record electronically by Monday, September 16, 2002 to the meeting Point of Contact. A public comment period is scheduled for Wednesday, September 25, 2002. The meeting agenda, including the specific time for the public comment period, and guidelines for making public comments will be posted on the Commission's Web site at *http://www.oceancommission.gov* prior to the meeting.

Dated: August 13, 2002.

Thomas R. Kitsos,

Executive Director, U.S. Commission on Ocean Policy.

[FR Doc. 02–21049 Filed 8–19–02; 8:45 am]

BILLING CODE 6820–WM–P

SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

Upon written request, copies available from: Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549.

Extension

Rule 10a–1, SEC File No. 270–413, OMB Control No. 3235–0475
Rule 12d2–1, SEC File No. 270–98, OMB Control No. 3235–0081
Rule 12d2–2 and Form 25, SEC File No. 270–86, OMB Control No. 3235–0080
Rule 17Ab2–1 and Form CA–1, SEC File No. 270–203 OMB Control No. 3235–0195
Rule 17Ad–3(b), SEC File No. 270–424, OMB Control No. 3235–0473

Notice is hereby given that, pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*), the Securities and Exchange Commission (Commission) has submitted to the Office of Management and Budget requests for approval of extension on the following:

Rule 10a–1 (17 CFR 240.10a–1) under the Securities Exchange Act of 1934 (Exchange Act) is designed to limit short selling of a security in a declining market, by requiring, in effect, that each successive lower price be established by a long seller. The price at which short sales may be effected is established by reference to the last sale price reported in the consolidated system or on a particular marketplace. Rule 10a–1 requires each broker or dealer that effects any sell order for a security registered on, or admitted to unlisted trading privileges on, a national securities exchange to mark the relevant order ticket either ''long'' or ''short.''

There are approximately 7,258 brokers and dealers registered with the national securities exchanges. The Commission has considered each of these respondents for the purposes of calculating the reporting burden under Rule 10a–1. Each of these approximately 7,258 registered broker-dealers effects sell orders for securities registered on, or admitted to unlisted trading privileges on, a national securities exchange. In addition, each respondent makes an estimated 59,071 annual responses, for an aggregate total of 428,743,000 responses per year. Each response takes approximately .000139 hours (.5 seconds) to complete. Thus, the total compliance burden per year is 59,595 burden hours.

There is no retention period requirement under Rule 10a–1. This Rule does not involve the collection of confidential information.

Rule 12d2–1 (17 CFR 240.17d2–1) was adopted in 1935 pursuant to sections 12 and 23 of the Exchange Act. The Rule provides the procedures by which a national securities exchange may suspend from trading a security that is listed and registered on the exchange. Under Rule 12d2–1, an exchange is permitted to suspend from trading a listed security in accordance with its rules, and must promptly notify the Commission of any such suspension, along with the effective date and the reasons for the suspension.

Any such suspension may be continued until such time as the Commission may determine that the suspension is designed to evade the provisions of section 12(d) of the Exchange Act and Rule 12d2–2 thereunder.[1] During the continuance of such suspension under Rule 12d2–1, the exchange is required to notify the Commission promptly of any change in the reasons for the suspension. Upon the restoration to trading of any security suspended under the Rule, the exchange must notify the Commission promptly of the effective date of such restoration.

The trading suspension notices serve a number of purposes. First, they inform the Commission that an exchange has suspended from trading a listed security or reintroduced trading in a previously suspended security. They also provide the Commission with information necessary for it to determine that the suspension has been accomplished in accordance with the rules of the exchange, and to verify that the exchange has not evaded the requirements of section 12(d) of the Exchange Act and Rule 12d2–2 thereunder by improperly employing a trading suspension. Without the Rule, the Commission would be unable to fully implement these statutory responsibilities.

There are nine national securities exchanges that are subject to Rule 12d2–

[1] Rule 12d2–2 prescribes the circumstances under which a security may be delisted, and provides the procedures for taking such action.