MATTERS TO BE CONSIDERED:

Week of June 1

Tuesday, June 2

8:00 a.m.—Briefing on Remaining Issues Related to Proposed Restart of Millstone Unit 3. (Public Meeting) (Contact: Bill Travers, 301–415–1200).
1:00 p.m.—(Continuation of morning meeting on Millstone.)

Wednesday, June 3

2:00 p.m.—Briefing by the Executive Branch (Closed—Ex. 1).

Thursday, June 4

2:00 p.m.—Briefing by NEI and NRC Staff on Safety Evaluations, FSAR Updates and Incorporation of Risk Insights (Public Meeting).

Friday, June 5

10:00 a.m.—Briefing by EPRI on the Status of their Advanced Light Water Reactor (ALWR) Program (Public Meeting).
11:30 a.m.—Affirmation Session (Public Meeting); a: Private Fuel Storage, L.L.C.; Ruling by Chief Judge of the Atomic Safety and Licensing Board Panel to Establish a Second Board, LBP–98–8 (April 24, 1998).

Week of June 8—Tentative

Thursday, June 11

11:30 a.m.—Affirmation Session (Public Meeting) (if needed).

Friday, June 12

10:00 a.m.—Briefing by Reactor Vendors Owners' Groups (Public Meeting) (Contact: Bryan Sheron, 301–415–1274).

Week of June 15—Tentative

Wednesday, June 17

10:00 a.m.—Briefing by National Mining Association on Regulation of the Uranium Recovery Industry (Public Meeting).
11:30 a.m.—Affirmation Session (Public Meeting) (if needed).
2:00 p.m.—Meeting with Advisory Committee on Medical Uses of Isotopes (ACMUI) and Briefing on Part 35 QM Rule (Public Meeting) (Contact: Larry Camper, 301–415–7231).

Week of June 22—Tentative

Thursday, June 25

9:30 a.m.—Briefing by IG on Results of NRC Organization Safety Culture and Climate Survey (Public Meeting).
11:30 a.m.—Affirmation Session (Public Meeting) (if needed).
2:00 p.m.—Briefing on EEO Program (Public Meeting).

* The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Bill Hill (301) 415–1661.

The NRC Commission Meeting Schedule can be found on the Internet at:

http://www.nrc.gov/SECY/smj/schedule/htm

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 it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, D.C. 20555 (301–415–1661). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: May 29, 1998.

William M. Hill, Jr.,

Secy, Tracking Officer, Office of the Secretary.

[FR Doc. 98–14821 Filed 6–1–98; 11:15 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 May 11, 1998, through May 21, 1998. The last biweekly notice was published on May 20, 1998 (63 FR 27757).

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

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not; (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

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 Administration 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 NRC Public

Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By July 6, 1998, 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 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. 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.

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: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and 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 Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

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

Date of amendment request: March 6, 1998.

Description of amendment request: The proposed change will revise the H. B. Robinson, Unit 2 Technical Specifications to allow use of the Post Accident Monitoring (PAM) source range (SR) neutron flux detector as a compensatory measure in the event that one of the two required BF3 detectors become inoperable while the plant is in MODE 6.

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

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

The proposed change to Technical Specifications is only applicable during the refueling mode of operation (MODE 6). Neither the BF3 SR nor PAM neutron flux monitors provide an automatic initiation signal for the operation of plant systems or components but are only relied upon to provide indication of core reactivity. Since the proposed change to Technical Specifications does not alter the design or operation of plant equipment or systems, there is no change in the initiating mechanisms for any accidents previously analyzed. Therefore this change does not involve a significant increase in the probability for an accident previously analyzed.

The UFSAR [Updated Final Safety Analysis Report] identifies two accidents that credit the SR monitoring capability in MODE 6, the boron dilution accident and the fuel handling accident. No other accidents were found to rely on SR monitoring in MODE 6. The proposed change will continue to require BF3 SR visual indication of core reactivity in the control room and a BF3 SR neutron flux monitor audible indication in containment. This change will not result in a significant reduction in operator capability to detect unexpected changes in core reactivity and perform actions credited with termination of those events, therefore the proposed change does not involve a significant increase in the consequences of an accident previously analyzed.

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

The proposed change to Technical Specifications does not involve any physical alteration of plant systems, structures or components or changes in parameters governing plant operations. The proposed change will not result in a significant reduction in monitoring capability since two BF3 SR channels of SR visual indication in the control room and audible SR indication in the containment are required during core alterations and positive reactivity changes. The use of the PAM SR neutron flux monitor as a compensatory measure does not introduce any new accident initiation scenarios since the SR instruments are for monitoring and criticality assessment only and are not relied upon to initiate automatic accident mitigation measures. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously analyzed.

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

The proposed change will maintain two BF3 SR monitoring means for visually monitoring core reactivity as currently discussed in the bases for the affected Technical Specifications. Audible indication provided by one BF3 SR neutron flux monitor will still be required and fulfilled by the remaining BF3 SR neutron flux monitor. The PAM SR neutron flux monitors use fission chambers as detectors which have a sensitivity of 4 cps/neutron-volts (cps/nv) for thermal neutrons and 2 cps/nv for fast neutrons. The BF3 SR neutron flux monitors have a sensitivity of 9 cps/nv. The PAM SR neutron flux monitor has comparable range and accuracy (i.e., range of 1E–01 cps to 1E+05 cps with an accuracy of 2% of full scale) to that of BF3 SR neutron flux monitor (i. e., range of 1E–00 cps to 1E + 06 cps with an accuracy of 3% of full scale) which meets the Technical Specifications Section 3.9.2 Bases requirements of 6 decades of indication and 5% accuracy. Therefore, this change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

Attorney for licensee: William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Project Director: P. T. Kuo, Acting.

Commonwealth Edison Company, Docket No. 50–249, Dresden Nuclear Power Station, Unit 3, Grundy County, Illinois

Date of amendment request: May 6, 1998.

Description of amendment request: The proposed amendment would amend Technical Specification (TS) 4.6.E to allow a one-time extension of the 40-month requirement to pressure set test or replace all Main Steam Safety Valves (MSSVs) to a maximum interval of 60 months as currently allowed by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code).

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. Involve a significant increase in the probability or consequences of an accident previously evaluated because of the following:

The proposed changes request a one-time change to the surveillance requirement for the MSSVs. The surveillance interval between safety valve testing is not a precursor assumed in any previously analyzed accident. Therefore, the probability of a previously evaluated accident has not been increased.

The proposed extension is consistent with the ASME Code requirement to test all valves within 60 months. The proposed changes are also consistent with NUREG–1433 and do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. Operating experience and superior materiel condition of the MSSVs support the expectation that they will continue to perform their intended function. Therefore, the consequences of a previously evaluated accident have not been increased.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated because:

No new equipment is required, nor will the MSSVs be operated in a different manner during the period of the extended surveillance interval. The proposed change is consistent with NUREG–1433 requirements for safety valve surveillance intervals as well as the ASME Code for requirements testing safety valves. Operating experience and superior materiel condition of the MSSVs support the expectation that they will continue to perform their intended function. Therefore, the possibility of a new or different accident has not been increased.

3. Involve a significant reduction in the margin of safety because:

The proposed amendment represents an extension to the current TS requirements, but would otherwise be provided generically by ASME Code. The proposed changes are also consistent with NUREG–1433, request a shorter total interval than previously granted by the Staff (Reference b), [J.F. Stang (NRC) to D. L. Farrar, SER dated October 8, 1996] and do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. The proposed changes have been evaluated and found to be acceptable for use at Dresden based on system safety analysis requirements and operational performance. The MSSV provisions continue to be adequately maintained during plant operation. The proposed changes to the MSSV surveillance interval do not significantly reduce existing plant safety margins since excellent materiel condition and acceptable surveillance test results support the expectation that no significant degradation will occur over the extended interval.

The proposed changes are based on NRC accepted provisions at other operating plants that are applicable at

Dresden and maintain necessary levels of system or component reliability.

The proposed amendment for Dresden will not reduce the availability of systems required to mitigate accident conditions; therefore, the proposed 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 requested amendments involve no significant hazards consideration.

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Stuart A. Richards.

Illinois Power Company, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of amendment request: May 4, 1998.

Description of amendment request: The proposed amendment would incorporate Technical Specifications Requirements for the protection systems for the new static VAR compensators being installed onsite to address degraded electrical grid voltage.

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 changes addressed by this amendment request involve the addition of SVCs and their associated protection systems to the onsite circuit connections for the plant offsite electrical power sources, i.e., to the RAT and ERAT. As noted throughout this request, the addition of the SVCs will help to maintain voltage at the site for both of the offsite electrical power sources consistent with the ''capacity and capability'' requirements of GDC 17. Further, the regulating effect of the SVCs will compensate for the voltage drop that can occur without the SVCs when the plant trips off-line (and thus no longer supports grid voltage) during normal or accident conditions. This supports compliance with the GDC 17 requirement to minimize the probability of losing electric power from the offsite supplies as a result of, or coincident with, the loss of power from the offsite supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit. Consequently, the likelihood of transferring to the onsite emergency power supplies (diesel generators) during an accident will be reduced. At the same time, as also addressed in this amendment request, incorporation of the SVCs into the CPS auxiliary power system requires consideration of failure modes that could be introduced by the SVCs wherein such failure modes could involve a significant increase in the probability or consequences of any accident previously evaluated.

By supplying each of the SVCs with an enhanced protection system, consisting of dual, redundant protection subsystems, either of which will isolate the SVC from the bus (by automatically opening the SVC main circuit breakers) in response to postulated SVC failures or associated abnormal conditions, the potential for such conditions or failures to adversely affect the plant safety busses, the associated plant loads, or the onsite emergency electrical power sources is reduced to a very low probability. The protection systems designed for the SVCs include consideration of failure modes or abnormal conditions that may be postulated or expected to occur with some degree of probability for the offsite electrical sources or grid with or without the presence of the SVCs, (such as a sustained degraded voltage condition), as well as consideration of any new or other failure modes or abnormal conditions potentially introduced by the SVCs that would be less likely to occur in the offsite electrical network without the presence of the SVCs (such as the introduction of harmonics). The proposed change to the CPS Technical Specifications to incorporate requirements for the SVC protection systems will ensure that the SVC protection systems are adequately maintained in an operable condition to perform their intended function of protecting against such conditions or failure modes. Operable SVC protection systems will reduce the probability of an SVC failure event that leads to equipment damage and subsequent core damage to a level that makes such an event incredible.

It should be noted that tripping of the SVCs in response to an SVC failure or abnormal condition does not result in a loss of power from the offsite sources. Thus, the probability of a loss of offsite power, which is an analyzed event in the plant safety analyses, will not be significantly increased by the SVC protection systems.

As noted previously, the proposed change to the Technical Specifications to incorporate SVC protection system operability and testing requirements would ensure that plant safety systems or components are not electrically affected by the SVCs in an adverse manner. In addition, except where the SVCs are physically located and connected to the ERAT and RAT via bus ducts, plant safety-related structures and supporting systems would not be mechanically affected by the SVCs. Separation, clearance and related requirements to ensure no other interaction with the RAT, ERAT and offsite source connections, as well as for maintaining offsite source independence would be maintained. On this basis, the safety functions of systems for preventing or mitigating analyzed events or accidents would not be impacted by the SVCs.

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

(2) In consideration of the potential adverse impacts that the SVCs may have on plant systems, structures or components, such impacts are primarily confined to potential electrical faults or abnormal conditions. As noted above, the SVCs have no mechanical impact on safety-related plant systems, structures or components. Thus, no new failure modes or precursors to potentially new and unanalyzed events would be introduced via any mechanical means.

With respect to potential adverse electrical impacts, the potential electrical failure modes or abnormal conditions postulated for the SVCs include conditions or events that, although could be considered possible for the offsite sources (i.e., the grid), were not in fact considered credible and therefore previously evaluated for the offsite electrical sources. These conditions or events, such as the introduction of harmonics or excessive overvoltage or phase imbalance caused by an SVC failure, would have the potential to degrade plant safety-related equipment connected to the busses at the time of the SVC failure if no protection for such conditions was provided. However, enhanced protection systems are provided for the SVCs to ensure that such failures cannot damage plant equipment. As noted previously, the probability of an event involving an SVC failure that leads to equipment damage and subsequent core damage has been calculated to be 1.5 x 10^{-8}/year. This low probability makes such an event incredible just as comparable events that could be postulated for the offsite electrical power sources were not previously considered credible and therefore were not considered to be design basis events. The calculated probability of 1.5 x 10^{-8}/year for an SVC failure event involving core damage is an order of magnitude lower than the threshold probability criterion specified in Section 2.2.3 of the Standard Review Plan (NUREG 0800) for design basis events involving an offsite hazard that can lead to core damage and radioactive release with dose consequences in excess of the limits specified in 10 CFR Part 100.

The proposed change to the Technical Specifications incorporates requirements for maintaining operability of the SVC protection systems. On this basis and as described above, no new credible accidents that could be associated with the SVCs (i.e., failure of the SVCs) are thus introduced, so that the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) As noted previously, incorporation of the SVCs into the CPS auxiliary power system will support or regulate plant bus voltage for both of the offsite sources. Specifically, analysis has shown that the SVCs will recover reduced margin that has occurred or would occur in the future (without the SVCs) with respect to the voltage required for plant safety loads and the minimum expected offsite voltage, under normal and accident conditions (i.e., under steady-state and transient voltage conditions). This also means that the SVCs will enhance the capability and capacity of the offsite sources such that, when compared to the configuration of not having the SVCs, either source will be more likely to reset the safety bus degraded voltage relays in the event of an accident, thus permitting the preferred offsite sources to remain connected (and not causing a transfer to the diesel generators). These desirable results constitute a significant increase in the margin of safety with respect to voltage requirements for plant loads.

Based on the above, IP has concluded that the proposed change to the Technical Specifications to support use of the SVCs and their protection systems does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, IL 61727.

Attorney for licensee: Leah Manning Stetzner, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, IL 62525.

NRC Acting Project Director: Ronald R. Bellamy.

North Atlantic Energy Service Corporation, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: April 22, 1998.

Description of amendment request: The proposed change would revise selected Technical Specification (TS) surveillance requirements to accommodate fuel cycles of up to 24 months for surveillances that are currently performed at each 18-month or other specified outage interval. Specifically, the following TS surveillance requirements would be revised by the proposed change: 4.1.2.2.b and c, ''Boration Systems Flow Paths—Operating;'' 4.3.3.5.2, ''Remote Shutdown System;'' 4.4.3.2, ''Pressurizer;'' 4.4.4.1, ''Relief Valves;'' 4.4.6.2.2.a and b, ''Operational Leakage;'' 4.4.11.2, ''Reactor Coolant System Vents;'' 4.5.1.1.d.1 and 2, ''Accumulators;'' 4.5.2.d, e, g.2), and h, ''Emergency Core Cooling System (ECCS) Subsystems—Tavg Greater Than or Equal to 350°F;'' 4.6.3.2, ''Containment Isolation Valves;'' and 4.7.1.2.1.c, ''Auxiliary Feedwater System.'' In conjunction with the proposed change, components addressed in the following TS surveillance requirements have been evaluated to support an extension in frequency to accommodate fuel cycles of up to 24 months: 4.6.3.1 and 3, ''Containment Isolation Valves;'' 4.7.1.2.2, ''Auxiliary Feedwater System;'' 4.7.1.5, ''Main Steam Line Isolation Valves;'' and 4.7.1.6, ''Atmospheric Relief Valves.'' In addition, the proposed change would delete the restriction ''during shutdown'' in those TS surveillance requirements where this restriction is stated.

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 have no adverse affect on accident initiators or precursors nor alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. The proposed changes do not alter or prevent the ability of structures, systems, or components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). The proposed changes are administrative in nature and do not change the level of programmatic controls or the procedural details associated with aforementioned surveillance requirements.

Changing the frequencies of the aforementioned surveillance requirements from at least once per 18 months to at least once per refueling interval does not change the basis for the frequencies. The frequencies were chosen because of the need to perform these verifications under the conditions that are normally found during a plant refueling outage, and to avoid the potential of an unplanned transient if these surveillances were conducted with the plant at power.

Equipment performance over several operating cycles was evaluated to determine the impact of extending the surveillance intervals. This evaluation included a review of surveillance results, preventative maintenance records, and the frequency and type of corrective maintenance activities, and a failure mode analysis. The evaluations conclude that the subject SSCs are highly reliable, presently exhibiting no time dependent failure modes of significance, and that there is no indication that the proposed extension could cause deterioration in the condition or performance of the subject SSCs. There are no known mechanisms that would significantly degrade the performance of the evaluated equipment during normal plant operation. Although there have been generic or repetitive failures of some components in the past, which may have affected the ability of the SSCs to consistently and successfully perform their safety function, those items have been resolved through design changes and rework such that they have not recurred. There have been no repetitive failures or time dependent failures that were significant in nature which would have prevented the SSCs from performing their intended safety function.

Deletion of the restriction ''during shutdown'' where this restriction is stated will permit performance of certain maintenance and testing activities during conditions or modes other than shutdown. North Atlantic will ensure, through the implementation of administrative controls that proper regard to their effect on safe operation of the plant is given prior to conduct of a particular surveillance in a condition or mode other than shutdown.

Since the proposed changes only affect the surveillance intervals for SSCs that are used to mitigate accidents, the changes do not affect the probability or consequence of a previously analyzed accident. While the proposed changes will lengthen the intervals between surveillances, the increase in intervals has been evaluated. Based on the reviews of the surveillance tests, inspections, and maintenance activities, it is concluded that there is no significant adverse impact on the reliability or availability of these SSCs.

Since there are no changes to previous accident analyses, the radiological consequences associated with these analyses remain unchanged, therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. There are no changes to the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences in the Seabrook Station UFSAR. Existing system and component redundancy is not being changed by the proposed changes. The proposed changes have no adverse impact on component or system interactions. The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details associated with the aforementioned surveillance requirements. Therefore, since there are no changes to the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated and surveilled, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

There is no adverse impact on equipment design or operation and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details associated with the aforementioned surveillance requirements.

From the evaluations performed on the subject SSCs there are no indications that potential problems would be cycle-length dependent or that potential degradation would be significant for the time frame of interest and, therefore, increasing the surveillance interval to the bounding limit of 30 months (24 months plus 25%) will have little, if any, impact on safety.

The proposed changes to the surveillance intervals are still consistent with the basis for the intervals and the intent and method of performing the surveillance is unchanged. Deletion of the restriction ''during shutdown'' where this restriction is stated will permit performance of certain maintenance and testing activities during conditions or modes other than shutdown. North Atlantic will ensure, through the implementation of appropriate administrative controls, that proper regard to their effect on safe operation of the plant is given prior to conduct of a particular surveillance in a condition or mode other than shutdown. In addition, use of the subject SSCs during normal plant operation, combined with their previous history of availability and reliability, provide assurance that the proposed changes will not affect the reliability of the subject SSCs. Thus, it is concluded that the subject SSCs would be available upon demand to mitigate the consequences of an accident and, therefore, there is no 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270

NRC Project Director: Cecil O. Thomas.

Northeast Nuclear Energy Company (NNECO), et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: April 29, 1998.

Description of amendment request: The proposed amendment would allow NNECO to revise the Updated Final Safety Analysis Report (UFSAR) for Millstone Unit 2 by deleting the diversity requirement for the two low-range pressurizer pressure transmitters, PT–103 and PT–103–1.

NNECO proposes to replace PT–103 and PT–103–1 with transmitters that are more accurate in a post-accident environment to provide assurance that entry into shutdown cooling in a post-accident environment is not compromised and to provide relief for the reactor coolant system pressure/ temperature curves. NNECO further indicates that only a single model series of Rosemount transmitters meet the revised design requirements and has specifically requested to delete the diversity requirement in the UFSAR, Section 4.3.8.2.3, ''Pressurizer Pressure.'' NNECO has determined that this deviation from the current design basis constitutes an unreviewed safety question as defined in 10 CFR 50.59.

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

The replacement of the low-range pressurizer pressure transmitters with non-diverse transmitters will reduce the instrument uncertainties post—SBLOCA [small break loss-of-coolant accident] or MSLB [main steamline break]. The probability of a post-accident intersystem LOCA [loss-of-coolant accident] as a result of aligning the SDC [shutdown cooling] system to RCS [reactor coolant system] pressures beyond its design pressure is reduced due to the reduced uncertainties of low-range pressurizer pressure signals to the SDC suction valve interlocks. The reduced uncertainty associated with the low-range pressurizer pressure transmitters in a harsh environment will not significantly reduce the probability of previously evaluated accidents relative to the use of the transmitter signal as an input variable to the ICC [inadequate core cooling] system, or to the other functions of LTOP [low temperature overpressure protection], SIT [safety injection tank] interlock, and Hot Shutdown Panel indication since these are functions not required in post-accident design bases. With respect to ICC, other parameters are available to the operator to determine adequate cooling of the core is taking place and saturation conditions are being approached or are occurring.

The loss of diversity in manufacturer and operating principle results in a small increase in the susceptibility of the replacement transmitters to common cause events that are primarily linked to internal failures of the transmitters verses failure that result from external events or conditions. To some extent, externally related common cause failures that can result from calibration or maintenance errors can be expected to also increase slightly because of commonality of procedures. Common cause failure increase is considered for the identified functions and all accidents. Because of the slight increase in the probability of common cause failures, the probability of exceeding RCS pressure/temperature (P/T) curves at temperatures [less than] 275°F is slightly increased (assuming a common cause failure of the replacement transmitters that would result in indicating a pressure lower than actual RCS pressure). Also, the potential for exceeding ASME Section III, Appendix G, pressure/temperature limit curves on cooldown and heatup is also slightly increased due to the slight increase in potential for common cause failure. This small increase in the potential for common cause failure will not significantly affect the probability of previously evaluated accidents. The reasons for this are:

—Exceedance of P/T limit curves does not in and of itself result in an accident initiator.

—Internally caused common cause failures are not expected to have a significant impact on the overall common cause potential of the transmitters. Typically, the majority of common cause potential is due to external reasons. Further, many times simultaneous internal failures of instrumentation can be recognized by direct comparison at which time alternative means can be sought, if available. Unless failure of the replacement transmitters was simultaneous and resulted in consistent output signals, transmitter failure would likely be recognized before requiring LTOP.

—The narrow-range pressurizer pressure loops (1500–2500 psia [pounds per square inch absolute]) are fully qualified Class 1E with transmitters manufactured by a different vendor. They provide a check against the low-range pressurizer pressure transmitters in the overlapped range of 1500–1600 psia.

—The non-class 1E wide-range pressurizer pressure loop (0–3000 psig [pounds per square inch gauge]), although not environmentally qualified, utilizes a qualified type transmitter manufactured by a different vendor and has been demonstrated to be reliable. It provides a check against the low-range pressurizer pressure transmitters in the overlapped range of 0–1600 psia.

The probability of an intersystem LOCA will not increase, due to the multiple means of determining that RCS pressure is beyond the SDC system design pressure and the multiple failures that would have to occur. All other functions evaluated (ICC, Hot Shutdown Panel and the SIT interlock) would not increase the probability of a previously evaluated accident.

Because the function and output of the replacement transmitters is the same as the existing transmitters and the transmitter failure types has not changed, the radiological consequences of previously evaluated accidents are not affected by the proposed change. The exception to this occurs when considering consequences of accidents that result in a harsh environment inside Containment and requiring SDC for long term cooling. In these cases, (SBLOCAs and MSLBs inside containment) given that transmitter accuracy is improved, the ability of getting onto SDC improves. This allows getting onto SDC more consistently. By doing so, a reduction in the radiological consequences of these accidents may be improved.

Therefore, there is no 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.

The function and output of the replacement transmitters are the same as the existing transmitters, and the transmitter failure types have not changed.

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

3. Involve a significant reduction in a margin of safety.

Improvement of the transmitters' ability to provide an accurate interpretation of RCS pressures in the operating range of 0–1600 psia (post-accident harsh environment in Containment) results in a positive benefit in the ability to control cooldown rates, establish SDC, and operate at the proper RCS pressures. However, the Margin of Safety is not impacted when the original transmitters/uncertainty calculations are compared to the proposed replacement transmitters/uncertainty calculations with regard to RCS P/T curves.

Therefore, the change will not involve a reduction in a margin of safety.

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

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

NRC Deputy Director: Phillip F. McKee.

Power Authority of The State of New York, Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: June 25, 1997.

Description of amendment request: The proposed amendment would change the Indian Point 3 Technical Specifications to allow the use of zirconium alloy or stainless steel filler rods in fuel assemblies to replace failed or damaged fuel rods.

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:

Consistent with the criteria of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously analyzed?

Response:

The proposed changes modify the technical specification only to the extent that the reconstitution is recognized as acceptable under limited circumstances. Reconstitution is limited to substitution of zirconium alloy or stainless steel filler rods, and must be in accordance with approved applications of fuel rod configurations. Although these changes permit reconstitution to occur without the need for a specific technical specification change, use of an approved methodology is required prior to its application. Since the changes will allow substitution of filler rods for leaking, potentially leaking rods or damaged rods, the changes may actually reduce the radiological consequences of an accident. It is noted that the specific changes requested in this letter have previously been found acceptable by the NRC in GL [Generic Letter] 90–02, Supplement 1. For these reasons, we conclude that the changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response:

The proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated because they will only affect the assembly configuration and can only be implemented if demonstrated to meet current plant requirements in accordance with an NRC-approved methodology. The other aspects of plant design, operation limitations, and responses to events will remain unchanged. It is noted that the changes have previously been determined acceptable by the NRC in GL 90–02, Supplement 1.

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

Response:

The proposed change will not involve a reduction in a margin of safety because the changes can only be implemented if demonstrated to meet current plant requirements in accordance with an NRC-approved methodology. It is noted that the changes have previously been determined acceptable by the NRC in GL 90–02, Supplement 1.

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.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Attorney for licensee: Mr. David Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Project Director: S. Singh Bajwa.

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: April 24, 1998.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 3/4.3.1.1, ''Reactor Protection System Instrumentation,'' TS Section 3/4.3.2.1, ''Safety Features Actuation System Instrumentation,'' TS Section 3/4.3.2.2, ''Steam and Feedwater Rupture Control System Instrumentation,'' and the associated TS bases. The TS tables of response time limits would be relocated to the Davis-Besse Technical Requirements Manual. Other changes in these TS sections consistent with the relocation are also proposed.

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

The Davis-Besse Nuclear Power Station (DBNPS) has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit Number 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiator, conditions or assumptions are affected by the proposed changes to Technical Specification (TS) 3/4.3.1.1, Reactor Protection System (RPS) Instrumentation, TS 3/4.3.2.1, Safety Features Actuation System (SFAS) Instrumentation, and TS 3/4.3.2.2, Steam and Feedwater Rupture Control System (SFRCS) Instrumentation and the associated TS Bases to relocate their tables of response time limits to the Technical Requirements Manual (TRM) of the DBNPS Updated Safety Analysis Report (USAR).

The RPS, SFAS and SFRCS response time limits and surveillance intervals currently prescribed in the TS are not changed under the proposed License Amendment. The RPS, SFAS and SFRCS will continue to function in the manner described in the DBNPS USAR. Therefore, the performance of these protection systems will remain within the bounds of the USAR accident analysis.

Under the proposed changes, the response time limits of the RPS, SFAS and SFRCS would continue to be tested in accordance with plant procedures in the same manner as in the past. The specific RPS, SFAS and SFRCS tables of response time limits will be relocated and remain controlled by the TRM of the DBNPS USAR following the guidance of the NRC's Generic Letter (GL) 93–08, ''Relocation of Technical Specification Tables of Instrument Response Time Limits,'' dated December 29, 1993. Any change to the relocated tables for response time limits will be subject to review and evaluation under Section 50.59, ''Changes, Tests, and Experiments,'' of Title 10 of the Code of Federal Regulation (10 CFR) prior to any changes being made.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no accident conditions or assumptions are affected by the proposed changes. As described above, the changes are consistent with the guidance of NRC GL 93–08. The proposed changes administratively relocate response time tables and do not alter the source term, containment isolation, or allowable releases. The proposed changes, therefore, will not increase the radiological consequences of a previously evaluated accident.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes, which involve only the administrative relocation of response time limit tables. No new accident scenarios, transient precursors, failure mechanisms, or limiting failures are introduced as a result of the proposed changes. As described above, the changes are consistent with the guidance of NRC GL 93–08. The proposed changes do not alter any accident scenarios and future changes to the response time limits will be subject to the regulatory requirements in 10 CFR 50.59.

3. Not involve a significant reduction in a margin of safety because the proposed changes only administratively relocate the response time tables from the TS to the USAR TRM, and do not reduce or adversely affect the capabilities of any plant structures, systems or components. No response times will be changed by this amendment request. Future changes to the response time limits will be subject to the regulatory requirements of 10 CFR 50.59. Accordingly, there is not a reduction in a margin of safety.

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

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, OH 43606.

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

NRC Acting Project Director: Ronald R. Bellamy.

Wisconsin Public Service Corporation, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of amendment request: May 14, 1998.

Description of amendment request: The proposed Technical Specification (TS) amendment would redefine the parent tube pressure boundary location for Westinghouse mechanical hybrid expansion joint (HEJ) steam generator (SG) tube sleeves. The proposed amendment would change the parent tube pressure boundary definition from a minimum required interference lip to a minimum required length of non-degraded hardroll engagement.

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:

The proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed change will not:

(1) Involve a significant increase in the probability or consequence of an accident previously evaluated.

Mechanical testing shows inherent structural integrity of the HEJ upper joint such that the requirements of RG 1.121 are met even for 360 degree, 100 percent throughwall parent tube indications (PTIs). Structural test results are documented in WCAP–15050. Based on the test data, the structural recommendations of RG 1.121 are satisfied when there is a minimum length of non-degraded hardroll which measures 0.92 inch (plus an allowance for NDE measurement uncertainty) or more from the bottom of the hardroll upper transition (HRUT), as measured on the inside of the sleeve. Based on the structural integrity of the HEJ upper joint, it can be concluded that application of the revised parent tube pressure boundary will not result in a significant increase in the probability of an accident previously evaluated.

A conservatively bounding primary-to-secondary steam line break (SLB) leak rate of one gpm will be applied to the calculation for postulated SLB leakage. This leak rate encompasses all HEJs left inservice with PTIs located outside the revised parent tube pressure boundary. This one gpm is based on a normal operating leakage limit of 150 gpd. This leak rate is based on tests and analysis documented in WCAP–15050. Application of this leak rate to the postulated leakage calculation will ensure primary-to-secondary leakage will not exceed the current maximum allowable during a SLB event. Maintenance of the current maximum allowable primary-to-secondary leak rate during a SLB event ensures off-site doses will not exceed a small fraction of 10 CFR 100 and control room doses will not exceed GDC–19 criteria. Therefore, it can be concluded that the application of the revised parent tube pressure boundary will not increase the consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any previously evaluated.

Implementation of the revised parent tube pressure boundary will not introduce a change to the design basis or operation of the plant. The configuration of the currently installed sleeves is not physically changed. As with the initial installation of the sleeves and previous changes to the parent tube pressure boundary for HEJs, implementation of the revised parent tube pressure boundary does not interact with other portions of the reactor coolant system. Neither the sleeve design nor the implementation of the revised parent tube pressure boundary affects any other component or location of the tube outside of the immediate repaired area. Mechanical testing of representative specimens supports the conclusions that the joint retains structural integrity consistent with RG 1.121 and leakage integrity with regards to 10 CFR 100 and GDC–19. Any hypothetical accident as a result of potential PTIs is bounded by the existing steam generator tube rupture analysis. Therefore, application of the revised parent tube pressure boundary will not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Involve a significant reduction in the margin of safety.

The safety factors used in establishment of the HEJ sleeved tube pressure boundary are consistent with safety factors in the ASME Boiler and Pressure Vessel Code used in the SG design. Based on the sleeve-to-tube geometry, it is unrealistic to consider that application of the revised parent tube pressure boundary could result in single tube leak rates exceeding the normal makeup capacity during normal operating conditions. The parent tube pressure boundary developed in WCAP–15050 has been developed using the methodology of RG 1.121. The performance characteristics of postulated degraded parent of HEJ sleeve/tube joints have been verified through testing to retain structural integrity and preclude significant leakage during both normal operating and SLB conditions. The existing off-site and control room dose evaluation performed for KNPP established a faulted loop primary-to-secondary leak rate of 12.85 gpm. Combined leakage from all sources including the assumed leak rate for the voltage based repair criteria and for HEJs with PTIs that are left inservice will not exceed 12.85 gpm in the faulted loop. Maintenance of this limit will ensure off-site doses will not exceed a small fraction of the 10 CFR 100 guidelines nor will it exceed the GDC–19 criteria for control room dose. Therefore, the application of the revised parent tube pressure boundary will not result in a significant reduction in the margin of safety.

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

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, WI 54311–7001.

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

Acting NRC Project Director: Ronald R. Bellamy.

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, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

Carolina Power & Light Company, et al., Docket No. 50–325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of application for amendments: February 23, 1998, as supplemented March 27, 1998.

Brief description of amendment: The amendment modifies the values for the safety limit for the Minimum Critical Power Ratio (SLMCPR) in the TS and the associated action statement for Cycle 12 operation only. A reference in TS 6.9.3.2.c is also revised.

Date of issuance: May 11, 1998.

Effective date: May 11, 1998.

Amendment No.: 194.

Facility Operating License No. DPR–71: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 10, 1998 (63 FR 17900).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 11, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403–3297.

Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: September 26, 1997, as supplemented on April 7, 1998, and May 1, 1998.

Brief description of amendments: The amendments revise the Technical Specifications to upgrade the ventilation filter testing program to the current industry standards and specify that the auxiliary electric equipment room is required to be habitable during design bases accidents. Revisions related to drywell and suppression chamber purge and the editorial changes requested in the September 26, 1997, application were approved and issued under Amendment Nos. 125 and 110 dated April 27, 1998.

Date of issuance: May 13, 1998.

Effective date: Immediately, to be implemented prior to restart from L1F35 for Unit 1 and prior to restart from the current outage for Unit 2.

Amendment Nos.: 126 and 111.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 19, 1997 (62 FR 61840). The April 7 and May 1, 1998, submittals provided additional information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 13, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: July 8, 1994, as supplemented August 13, 1996, and February 12, 1998.

Brief description of amendment: The amendment revises Technical Specifications Sections 3.7 and 3.3.E to clarify offsite power availability requirement, revise emergency diesel generator fuel oil availability requirements and specify the configuration requirements for removing Component Cooling Pump 22 from service.

Date of issuance: May 8, 1998.

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

Amendment No.: 196.

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

Date of initial notice in **Federal Register**: August 17, 1994 (59 FR 42336).

The August 13, 1996, and February 12, 1998, letters provided clarifying information that 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 May 8, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

Consumers Energy Company, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: January 18, 1996, as supplemented October 1, 1997, and January 29 and April 27, 1998.

Brief description of amendment: The amendment revises the technical specifications regarding inspection requirements for the reactor coolant pump (RCP) flywheels. The staff denied a portion of the amendment request regarding application to the flywheel testing program of the Surveillance Requirement 4.0.2 allowance for surveillance interval extension of up to 25%. A separate Notice of Partial Denial of Amendment to Facility Operating License and Opportunity for Hearing has been published in the **Federal Register**.

Date of issuance: May 15, 1998.

Effective date: May 15, 1998.

Amendment No.: 182.

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

Date of initial notice in **Federal Register**: November 5, 1997 (62 FR 59915).

The January 29 and April 27, 1998, letters provided additional clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423.

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: December 19, 1997, as supplemented March 6, 1998.

Brief description of amendment: This amendment changes the wording of Section 4.2.2, ''Terrestrial Ecology Monitoring,'' of the Environmental Protection Plan to include completion of the Salt Drift Monitoring Program.

Date of issuance: May 8, 1998.

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

Amendment No.: 111.

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

Date of initial notice in **Federal Register**: January 28, 1998 (63 FR 4321) The March 6, 1998, supplement provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 8, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

Southern Nuclear Power Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: January 22, 1998, as supplemented by letter dated March 18, 1998, April 21, 1998, and May 15, 1998.

Brief description of amendments: The amendments change the Technical Specifications to allow an extended allowed outage time for one emergency diesel generator of 14 days.

Date of issuance: May 20, 1998.

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

Amendment Nos.: Unit 1–100; Unit 2–78.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications and Operating Licenses.

Date of initial notice in **Federal Register**: February 11, 1998 (63 FR 6998) The March 18, 1998, April 21, 1998, and May 15, 1998, supplements provided clarifying information that did not change the scope of the January 22, 1998, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 20, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Burke County Library, 412 Fourth Street, Waynesboro, Georgia.

Tennessee Valley Authority, Docket Nos. 50–260 and 50–296, Browns Ferry Nuclear Plant , Units No. 2 and 3, Limestone County, Alabama

Date of amendment request: December 11, 1996, as supplemented by letter dated November 3, 1997 (TS–386).

Description of amendment request: The amendment modifies the Appendix A Technical Specifications (TSs) Limiting Safety System Setting 2.2.A, which relates to the main steam safety/relief valve set points and the set point tolerance. Specifically, the revision increases the set point tolerance to ±3% vice the current ± 11 pound per square inch (approximately 1% of set point value) tolerance. Bases 1.2 and 3.6D/4.6D also are revised.

Date of issuance: May 18, 1998.

Effective date: May 18, 1998.

Amendment No.: 251 and 210.

Facility Operating License Nos. DPR–33, DPR–52, and DPR–68. Amendment revised the TSs.

Date of initial notice in **Federal Register**: January 15, 1997 (62 FR 2194). The licensee's letter dated November 3, 1997, provided additional supporting information that 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 May 18, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Athens Public Library, 405 South Street, Athens, Alabama 35611.

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

Date of application for amendment: March 20, 1998

The licensee proposed to modify the licensing basis by limiting the time the large (18″) purge and vent valves may be open to 90 hours per year. This is a change to the Final Safety Analysis Report (FSAR) and technical specification bases.

Date of Issuance: May 14, 1998.

Effective date: May 14, 1998.

Amendment No.: 161.

Facility Operating License No. DPR–28: Amendment authorizes revision to the FSAR.

Date of initial notice in **Federal Register**: March 27, 1998 (63 FR 14976).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated May 14, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

Dated at Rockville, Maryland, this 27th day of May 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

Acting Director, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 98–14519 Filed 6–2–98; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

Submission for OMB Review; Comment Request for Revision of Information Collection, RI 20–64 & RI 20–64A

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (Pub. L. 104–13, May 22, 1995), this notice announces that the Office of Personnel Management will submit to the Office of Management and Budget a request for revision of the following information collection. RI 20–64, Former Spouse Survivor Annuity Election, is used by the Civil Service Retirement System to provide information about the amount of annuity payable after a survivor reduction and obtain a survivor benefits election form from annuitants who are eligible to elect to provide survivor benefits for a former spouse. RI 20–64A, Information On Electing A Survivor Annuity For Your Former Spouse, is a pamphlet that provides important information to retirees under the Civil Service Retirement System who want to provide a survivor annuity for a former spouse.

Approximately 30 RI 20–64 forms are completed annually. Each form takes about 45 minutes to complete. The annual estimated burden is 23 hours.

For copies of this proposal, contact Jim Farron on (202) 418–3208, or E-mail to jmfarron@mail.opm.gov