Participation by the NRC staff and industry is anticipated.

D. *Risk-Informed, Performance-Based Regulation*—The Committee will review recent agency initiatives on risk-informed, performance-based regulation.

E. *Meeting with NRC's Director, Division of Waste Management, Office of Nuclear Material Safety and Safeguards*—The Committee will meet with the Director to discuss recent developments within the division such as developments at the Yucca Mountain project, rules and guidance under development, available resources, and other items of mutual interest.

F. *Preparation of ACNW Reports*—The Committee will discuss planned reports, including risk-informed, performance-based regulation, waste related research, regulatory guides dealing with decommissioning, and other topics discussed during this and previous meetings as the need arises.

G. *Committee Activities/Future Agenda*—The Committee will consider topics proposed for future consideration by the full Committee and Working Groups. The Committee will discuss ACNW-related activities of individual members.

H. *Miscellaneous*—The Committee will discuss miscellaneous matters related to the conduct of Committee activities and organizational activities and complete discussion of matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACNW meetings was published in the **Federal Register** on September 2, 1997 (62 FR 46382). In accordance with these procedures, oral or written statements may be presented by members of the public, electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Committee, its consultants, and staff. Persons desiring to make oral statements should notify the Chief, Nuclear Waste Branch, Mr. Richard K. Major, as far in advance as practicable so that appropriate arrangements can be made to schedule the necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during this meeting will be limited to selected portions of the meeting as determined by the ACNW Chairman. Information regarding the time to be set aside for taking pictures may be obtained by contacting the Chief, Nuclear Waste Branch, prior to the meeting. In view of the possibility that the schedule for ACNW meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should notify Mr. Major as to their particular needs.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Richard K. Major, Chief, Nuclear Waste Branch (telephone 301/415–7366), between 8:00 A.M. and 5:00 P.M. EST.

ACNW meeting notices, meeting transcripts, and letter reports are now available on FedWorld from the ''NRC MAIN MENU.'' Direct Dial Access number to FedWorld is (800) 303–9672; the local direct dial number is 703–321–3339.

Dated: March 6, 1998.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 98–6286 Filed 3–10–98; 8:45 am]

BILLING CODE 7590–01–P

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 February 13, 1998, through February 27, 1998. The last biweekly notice was published on February 25, 1998 (63 FR 9589).

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 April 10, 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.

Commonwealth Edison Company, Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of amendment request: January 14, 1998, which superseded the September 3, 1997, submittal.

Description of amendment request: The proposed amendments would revise the Technical Specifications to reduce the allowable Unit 1 Reactor Coolant System Dose Equivalent Iodine-131 from 0.35 microCuries/gram to 0.05 microCuries/gram thru the end of Unit 1, Cycle 7.

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.

Generic Letter 95–05, ''Voltage-Based Repair Criteria For Westinghouse Steam Generator Tubes Affected By Outside Diameter Stress Corrosion Cracking,'' allows lowering of the RCS [Reactor Coolant System] DE–131 [Dose Equivalent Iodine-131] activity as a means for accepting higher projected leak rates if justification for equivalent I–131 below 0.35 microCuries/gram is provided. Four methods for determining the impact of a release of activity to the public were reviewed to provide this justification. These four methods are as follows:

Method 1: NRC NUREG 0800, Standard Review Plan (SRP) Methodology

Method 2: Methodology described in a report by J.P. Adams and C.L. Atwood, ''The Iodine Spike Release Rate During a Steam Generator Tube Rupture,'' Nuclear Technology, Vol. 94, p. 361 (1991) using Braidwood Station reactor trip data.
Method 3: Methodology described in a report by J.P. Adams and C.L. Atwood, ''The Iodine Spike Release Rate During a Steam Generator Tube Rupture,'' Nuclear Technology, Vol. 94, p. 361 (1991) using normalized industry reactor trip data.
Method 4: Methodology described in a draft EPRI Report TR–103680, Revision 1, November 1995, ''Empirical Study of Iodine Spiking in PWR Plants''.

The effect of reducing the RCS DE I–131 activity limit on the amount of activity released to the environment remains unchanged when the maximum site allowable primary-to-secondary leak rate is proportionately increased and the iodine release rate spike factor is assumed to be 500 in accordance with the SRP. With an RCS DE I–131 activity limit of 1.0 microCuries/gram, the maximum site allowable leakage limit was calculated, in accordance with the NRC SRP methodology, to be 6.64 gpm at room temperature and pressure. ComEd has evaluated the reduction of the RCS DE I–131 activity to 0.05 microCuries/gram along with the increase of the allowable leakage to 132.8 gpm at room temperature and pressure and has concluded:

—assuming a spike factor of 500, the maximum activity released is not changed, and
—the offsite dose, including the iodine spiking factor, will be less than the 10 CFR 100 limits.

Based on the NRC SRP methodology for dose assessments and assuming the iodine spike factor of 500 is applicable at the new 0.05 microCuries/gram RCS DE I–131 activity limit, the Control Room dose, the Low Population Zone dose, and the dose at the Exclusion Area Boundary continue to satisfy the appropriately small fraction of the 10 CFR 100 dose limits.

An evaluation of the Control Room dose, attributed to an MSLB accident concurrent with steam generator primary-to-secondary leakage at the maximum site allowable limit, was performed in support of a license amendment request for application of a 1.0 volt Interim Plugging Criteria. This evaluation concluded that the activity released to the environment during an eight (8) hour time period from an MSLB accident (812 Curies for a Pre-accident iodine spike and 888 Curies for an accident-initiated iodine spike) is bounded by the activity released to the environment from the Loss of Coolant design basis accident (1290 Curies). Therefore, the Control Room dose, due to the MSLB accident scenario, is bounded by the existing Loss of Coolant Accident (LOCA) analysis. The maximum site allowable primary-to-secondary leakage is limited by the offsite dose at the Exclusion Area Boundary due to an accident-initiated spike.

The report by J.P. Adams and C.L. Atwood, ''The Iodine Spike Release Rate During a Steam Generator Tube Rupture,'' Nuclear Technology, Vol. 94, p. 361 (1991), concluded that the NRC SRP methodology, which specifies a release rate spike factor of 500 for iodine activity from the fuel rod to the RCS, is conservative when the RCS DE I–131 concentration is greater than 0.3 microCuries/gram. In order to evaluate whether a release rate spike factor of 500 is conservative below 0.3 microCuries/gram, actual operating data from the previous reactor trips of Braidwood Units 1 and 2, with and without fuel defects, were reviewed and analyzed using the methodology presented in Section II.C of the Adams and Atwood report (Method 2). The same five data screening criteria described in the Adams and Atwood report were applied to the Braidwood data to ensure consistency and validity when comparing the Braidwood results to the data in the Adams and Atwood report. Of the reactor trip events at Braidwood Units 1 and 2, seventeen (17) met the five data screening criteria.

Seven (7) of the seventeen (17) Braidwood trips occurred during cycles with no fuel defects. In all seven of these instances, the calculated spike factor was much less than the spike factor of 500 assumed in the NRC SRP methodology. Braidwood Unit 1 Cycle 7 is currently operating with no fuel defects and an RCS DE I–131 activity of approximately 3E–4 microCuries/gram. The seven previous trips with no fuel defects had steady-state iodine values that are reasonably close to the current operating conditions. It is therefore reasonable to conclude that, assuming continued operation with little to no fuel defects, the calculated spike factors from these events would reflect an actual event for Unit 1 Cycle 7, i.e. the spike factor will be less than 500.

Since some of the Braidwood spike factors were greater than 500 when the RCS DE I–131 activity prior to the accident was less than 0.3 microCuries/gram, ComEd examined the conservatisms in the current release rate calculation. The primary reason for the high spiking factors contained in the Adams and Atwood report (up to 12,000), is not because the absolute post-trip release rate is high (factor numerator), but rather because the steady-state release rate (factor denominator) is low. The Braidwood specific data resulted in six (6) events with a calculated release rate spike factor greater than 500. It is not expected based upon the Unit 1 Cycle 7 fuel conditions that a spiking factor greater than 500 would occur. The revised RCS DE I–131 activity limit will also ensure that the operating cycle will not continue if significant fuel defects develop.

In order to evaluate the Braidwood specific data against the NRC SRP methodology, the release rate for a steady-state RCS DE I–131 activity of 1.0 microCuries/gram was calculated. Using the Braidwood specific data, the pre-trip steady-state release rate is 27.5 Ci/hr. Using a release rate spike factor of 500 for the accident-initiated spike, the post-trip maximum release rate would be 13,733 Ci/hr (SRP Methodology). The highest post-trip iodine release rate from the Braidwood trip data, Event 15, was 1335 Ci/hr, it is important to remember that this number is determined by conservatively increasing the post-trip RCS DE I–131 activity by a factor of three (3), in accordance with the Adams and Atwood report.

The purpose of this amendment request is to reduce the TS [Technical Specification] RCS DE I–131 limit by a factor of twenty as compared to the original TS RCS DE I–131 limit of 1.0 microCuries/gram. By decreasing the TS RCS DE I–131 activity by a factor of twenty the maximum iodine release rate is 686.7 Ci/hr, (13,733 Ci/hr divided by 20). Two (2) of the seventeen (17) Braidwood data points exceed this value. Both occurred during cycles with fuel defects. Braidwood Unit 1 is currently operating with no fuel defects. Fifteen (15) of the 168 data points in the Adams and Atwood report exceed 686.7 Ci/hr. For the combined database of 185 data points, of which 17 exceeded 686.7 Ci/hr, only two of these seventeen (17) data points had a pre-trip RCS DE I–131 activity below 0.05 microCuries/gram. The 95% confidence prediction for the combined data sets bounded one (1) of these two (2) data points. This data indicates that the possibility for a post-trip iodine fuel release rate to exceed 686.7 Ci/hr, when the pre-trip RCS DE I–131 concentration is at or below 0.05 microCuries/gram, is small. The conservatisms mentioned in the following sections will reduce the possibility of exceeding a small fraction of the 10 CFR 100 limits should a fuel release greater than 686.7 Ci/hr occur.

If the Braidwood data were plotted with the Adams and Atwood data, the conclusions of the Adams and Atwood report would not be compromised. Where the Braidwood data contains spike factors greater than 500, the RCS DE I–131 concentrations are below 0.05 microCuries/gram. Since the Braidwood data includes very few data points near 0.05 microCuries/gram (the requested new TS limit), it is appropriate to use the Braidwood database combined with the Adams and Atwood database near 0.05 microCuries/gram to determine if a spike factor of 500 is appropriate. The combined databases contain seventy-nine (79) data points with a Pre-Trip RCS DE I–131 activity between 0.01 microCuries/gram and 0.10 microCuries/gram. Sixty-two (62) of these seventy-nine (79) data points (78%) have spike factors less than 500. Using the entire Braidwood database combined with the Adams and Atwood database, 141 of the 185 data points (76%) have an iodine spike factor less than 500. Therefore, it is reasonable to assume that a spike factor of 500 would not be exceeded for a majority of the events if an MSLB accident were to occur while the RCS DE I–131 activity is at or below 0.05 microCuries/gram. The highest spike factor seen in the Adams and Atwood report near a Pre-Trip RCS DE I–131 activity of 0.05 microCuries/gram was 773 (at 0.05 microCuries/gram). The corresponding release rate for this event was 368 Ci/hr which is less than the calculated Braidwood maximum release rate of 686.7 Ci/hr.

The predominant factors in calculating the offsite dose are the post-trip iodine release rate from the fuel and the flowrate at which the activity is being released to the environment, not whether the spike factor is greater than or less than 500. The post-trip DE I–131 release rate will determine the level of activity in the RCS that will be released. The flowrate will determine at what rate this activity is released to the environment. Method 3, which used an approach in the Adams and Atwood report, concluded that, at a 95% confidence of a 85 percentile, the post-trip iodine release rate was bounded by 0.608 Ci/hr-MWe. For Braidwood Station, which has a MWe rating of 1175, the post-trip iodine release rate, at a 95% confidence of a 85 percentile, should not exceed 714 Ci/hr. Two (2) of the seventeen (17) reactor trips from Braidwood exceeded 714 Ci/hr. These two (2) reactor trips had post-trip iodine release rates of 1335 Ci/hr (spike factor of 3471) and 802 Ci/hr (spike factor of 1483). Both occurred during cycles with fuel defects. Braidwood Unit 1 is currently operating with no fuel defects.

In the fourth method, the results from a Draft Electric Power Research Institute (EPRI) Report TR–103680, Rev. 1, November 1995, ''Empirical Study of Iodine Spiking In PWR Power Plants'' were applied. The objective of the EPRI study was to quantify the iodine spiking in a postulated Main Steam Line Break/Steam Generator Tube Rupture (MSLB/SGTR) accident sequences. In the EPRI report, an iodine spike factor between 40 and 150 was determined to match data from existing plant trips. The maximum iodine spike factor value of 150 was applied to a steady-state equilibrium RCS DE I–131 activity of 0.33 microCuries/gram. The resulting two-hour average iodine concentration for a postulated MSLB/SGTR accident sequence was determined to be 3.1 microCuries/gram. Since the EPRI report is based on industry data and the EPRI method predicted a post-accident iodine activity, which is a small fraction of the activity predicted by the NRC SRP methodology, it can be expected that, for the proposed 0.05 microCuries/gram limit under an MSLB/SGTR accident sequence, the post-accident iodine activity would typically be a small fraction of the RCS DE I–131 activity predicted by the NRC SRP methodology. For Braidwood, using the SRP methodology with an RCS DE I–131 activity of 1.0 microCuries/gram and a spike factor of 500, the Post-Trip RCS activity two hours after the event would be near 38 microCuries/gram. At an RCS DE I–131 activity of 0.05 microCuries/gram, it would require a spike factor of nearly 10,000 to obtain a Post-Trip RCS DE I–131 activity near 38 microCuries/gram. With a Post-Trip RCS DE I–131 activity of 38 microCuries/gram, an increase in the allowable leak rate could impact the 10 CFR 100 limits. To accommodate for an increase in the allowable leak rate by a factor of twenty, the resultant activity would need to be below 1.9 microCuries/gram. Two (2) of the seventeen (17) post-trip data points from Braidwood exceeded 1.9 microCuries/gram. Both occurred during cycles with fuel defects. Braidwood Unit 1 is currently operating with no fuel defects. The conservatisms mentioned below will reduce the possibility of exceeding a small fraction of the 10 CFR 100 limits should the post-trip iodine exceed 1.9 microCuries/gram.

Based on evaluations by the four methods above, Braidwood can conclude that the current methodology (Method 1) used to predict iodine spiking is conservative. Although dose projections indicate with confidence that the iodine spiking factor limit will be met, the conservatisms in the offsite dose calculation and current Braidwood Unit 1 operating conditions listed below, provide added assurance that the 10 CFR 100 limits, General Design Criteria (GDC) 19 criteria, and the requirements of NRC Generic Letter 95–05 will be satisfied if the iodine spike factor exceeds 500 or the post-trip fuel release rate exceeds 686.7 Ci/hr.

As further assurance that the 10 CFR 100 and GDC 19 limits are not exceeded, several conservatisms are inherent to the offsite dose calculation. These conservatisms include, but are not limited to:

1. The meteorological data used is at the fifth percentile. It is expected that the actual dispersion of the iodine would result in less exposure at the site boundary than the 30 Rem limit of 10 CFR 100.

2. Iodine partitioning is not accounted for in the faulted SG. With the high pH of the secondary water, some partitioning is expected to occur. An iodine partition factor of 0.1 is more realistic (per Table 15.1–3 of Reference 8 [the Braidwood Updated Final Safety Analysis Report]) than the 1.0 valued (no partitioning) used in the offsite dose calculation. This reduces calculated dose by 90%.

3. The activity in the RCS is not expected to increase instantaneously with the spike in iodine released from the defective fuel.

4. The results from the Braidwood tube pull data indicate that the projected Interim Plugging Criteria leak rate is conservative.

In addition, the current Braidwood Unit 1 operating conditions provide defense in depth and provide further assurance that the 10 CFR 100 and GDC 19 limits will not be exceeded:

1. Braidwood Unit 1 is currently operating with a debris resistant fuel design which is less likely to develop fuel defects.

2. As evidenced by industry data, if debris related fuel failures are going to occur they are most likely to be occur early in the cycle. Braidwood Unit 1 has operated approximately 6 months into its current cycle and has seen no signs of fuel defects. Therefore, fuel failure prior to completion of the current cycle is not likely.

3. The RCS DE I–131 activity is likely to be less than the TS limit. With the current Braidwood Unit 1 RCS DE I–131 activity near 3E–4 microCuries/gram with no fuel defects, the spike factor is expected to be considerably smaller than the 500 value.

4. It is unlikely, for the short time period this amendment is being requested (remainder of Cycle 7), that an accident-initiated iodine spike for Braidwood Unit 1 would be greater than the NRC SRP assumed value.

5. Primary-to-secondary leakage is likely to be less than the TS limit (150 gpd) in each of the four SGs prior to the event. Currently, minimal primary-to-secondary leakage (less than 5 gpd) exists at Braidwood Unit 1.

These proposed changes do not result in a significant increase in the consequences of an accident previously analyzed.

The RCS DE I–131 activity limit is not considered as a precursor to any accident. Therefore, this proposed change does not result in a significant increase in the probability of an accident previously analyzed.

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

The changes proposed in this amendment request conservatively reduce the Unit 1 RCS DE I–131 activity limit at which action needs to be taken. The changes do not directly affect plant operation. These changes will not result in the installation of any new equipment or systems or the modification of any existing equipment or systems. No new operating procedures, conditions or configurations will be created by this proposed amendment.

Accordingly, this proposed change does not create 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.

NRC Generic Letter 95–05 allows lowering of the RCS dose equivalent iodine as a means for accepting higher projected leakage rates provided justification for the RCS DE I–131 activity below 0.35 microCuries/gram is provided. Four methods for determining the fuel rod iodine release rates and spike factors during an accident were reviewed. Each of these methods utilized actual industry data, including Braidwood Units 1 and 2, for pre- and post-reactor trip RCS DE I–131 activities. Each of the methods demonstrated that the actual fuel rod iodine release rates are a small fraction of the release rate as calculated using the NRC SRP methodology. Although these values are a small fraction of that determined by the NRC SRP Method, Braidwood is also requesting an increase in the allowable primary-to-secondary leak rate during MSLB. By decreasing the TS RCS DE I–131 activity limit by a factor of twenty and increasing the allowable leak rate by a factor of twenty, the activity released to the public would be equal to or less than the activity calculated by the SRP method for each of the seventeen reactor trip events reviewed at Braidwood. The predicted end-of-cycle 7 leak rate is 122.3 gpm (Room T/P [temperature and pressure]). The calculated site boundary dose due to this leakage is 27.63 Rem. This dose meets the requirements of 10 CFR 100 and GDC 19. All design basis and off-site dose calculation assumptions remain satisfied. This proposed change would not result in 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 requested amendments involve no significant hazards consideration.

Local Public Document Room location: Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

NRC Project Director: Robert A. Capra.

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois

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

Date of amendment request: September 24, 1997.

Description of amendment request: The proposed amendment would revise Technical Specification Surveillance Requirement 4.3.4.2 to change the frequency of turbine throttle and governor valve testing from monthly to quarterly and incorporate corresponding administrative changes. Bases 3/4.3.4 will be changed to update a referenced vendor document and incorporate corresponding administrative changes.

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:

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

The Bases change is a reference update, which is administrative in nature. Additional administrative changes necessitated by a change in the presentation of the surveillance requirements are proposed. The changes are consistent with Generic Letter 93–05 and NUREG–1366. This change reduces the frequency of testing that is likely to cause transients or excessive wear of equipment. An evaluation of these changes indicates that there will be a benefit to plant safety. The evaluation, documented in NUREG–1366, considered (1) unavailability of safety equipment due to testing, (2) initiation of significant transients due to testing, (3) actuation of engineered safety features that unnecessarily cycle safety equipment, (4) importance to safety of that system or component, (5) failure rate of that system or component, and (6) effectiveness of the test in discovering the failure.

As a result of the decrease in the testing frequencies, the risk of testing causing a transient and equipment degradation will be decreased, and the reliability of the equipment will not be significantly decreased.

The initial conditions and methodologies used in the accident analyses remain unchanged. The proposed changes do not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident. Therefore, accident analyses results are not impacted. Appropriate testing will continue to assure that equipment and systems will be capable of performing the intended function. The frequency of testing is not a precursor for any analyzed accidents.

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

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

The proposed changes modify allowable intervals between turbine throttle and governor valve surveillance tests. The proposed changes do not affect the design or operation of any system, structure, or component in the plant. The safety functions of the related structures, systems, or components are not changed in any manner, nor is the reliability of any structure, system, or component reduced by the revised surveillance or testing requirements. Appropriate testing will continue to assure that the system is capable of performing its intended function.

The changes do not affect the manner by which the facility is operated and do not change any facility design feature, structure, system, or component. No new or different type of equipment will be installed.

The turbine valve testing surveillances will be changed to account for a frequency change from monthly to quarterly for the throttle valves and for the governor valves.

Since there is no change to the facility or operating procedures, and the safety functions and reliability of structures, systems, or components are not affected, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

All of the proposed Technical Specification changes are compatible with plant operating experience and are consistent with the guidance provided in Generic Letter 93–05 and NUREG–1366. The changes reduce the frequency of testing that increases the risk of transients and equipment degradation. There is no impact on safety limits or limiting safety system settings. The Bases change is a vendor reference update, which is administrative in nature.

Certain reload designs can be such that power differences between the top and bottom of the core are more sensitive to control and can develop divergent xenon oscillations when the power reduction occurs during the middle of core life. Near the end of core life, stabilizing even larger differences in axial power distribution becomes more of a problem because of the larger temperature coefficient, lower boron concentration and larger differential xenon transient. In the Safety Evaluation Report related to the Prairie Island Amendment Numbers 86 and 79 in regard to the discussion above, the NRC wrote, ''Based on the above, the staff has concluded that the margin of safety is reduced when the plant is undergoing turbine valve testing.''

Since this amendment reduces the number of turbine tests while still maintaining acceptable equipment reliability, the proposed changes result in an increase 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: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

NRC Project Director: Robert A. Capra.

Detroit Edison Company, Docket No. 50–16, Enrico Fermi Atomic Power Plant, Unit 1, Monroe County, Michigan

Date of amendment request: December 15, 1997 (Reference NRC–97–0115).

Description of amendment request: The proposed amendment will revise License Condition A to delete references to letters dated May 17, 1985, July 23, 1986, September 15, 1986, September 25, 1987, September 15, 1988, and December 22, 1988, and replace them with the Enrico Fermi Atomic Power Plant, Unit 1, Safety Analysis Report (F1SAR) as the licensing basis.

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 using the standards in 10 CFR 50.92(c). The licensee's analysis is presented below:

(1) *Does the proposed change significantly increase the probability or consequences of an accident previously evaluated?*

No, the proposed submittal of the F1SAR as the facility's licensing basis document does not significantly increase the probability of an accident. The F1SAR is a compilation of previously submitted information and other information gathered on the condition of the facility. Compilation of current information and imposition of the new Fire Protection and Quality Assurance Program requirements will not increase the probability of an accident. These additional controls would reduce the probability of an event. The proposed addition of a hypothetical secondary sodium accident scenario identifies one possible previously unidentified potential cause of a primary sodium release and/or liquid waste tank release. The previous submittal assumed the cause of the primary sodium release to be a fire or other catastrophic event. The cause of the liquid waste tank rupture was assumed to be an earthquake. Recognition of a cause being the reaction of secondary sodium does not significantly increase the probability of a primary sodium release or liquid waste release. A catastrophic event would still need to occur to cause the postulated scenario, so there is no discernible increase in the probability of the primary sodium or liquid waste accident compared to the existing licensing basis. For the reasons discussed above, substituting the F1SAR as the licensing basis for Fermi 1 will not significantly increase the probability of an accident.

The proposed submittal of the F1SAR as the Fermi 1 licensing basis document will have no impact on the consequences of an accident. Consolidating current information on the plant and previous submittals does not change the amount of radioactivity at the facility or the potential magnitude of any release during an accident. Since the potential accident source terms were not updated as part of the submittal, the consequences of the accidents contained in the F1SAR match the consequences in the previous submittal. Though a new postulated hypothetical accident scenario was added, the secondary sodium involved in that accident is not radioactive, per previous submittals, and so the only potential radiological consequences of that scenario occur if the primary sodium or liquid waste is released and those consequences have already been reviewed in the NRC safety analysis for Amendment No. 9 to the Fermi 1 license. Therefore, the adoption of the F1SAR as the facility's licensing basis will not significantly increase the consequences of an accident at Fermi 1.

(2) *Will the proposed amendment create the possibility of a new or different kind of accident from any accident previously analyzed?*

No, establishment of the F1SAR as the Fermi 1 licensing basis document will not create a new type of accident. The F1SAR is mainly a compilation of the previous licensing basis documents, information on the facility condition and additional controls. It does not involve operating in any new type of mode and so cannot create a new or different type of accident. The new hypothetical secondary sodium accident contained in the F1SAR is a sodium accident. One of the existing licensing basis accidents is the primary sodium accident resulting in release of the primary sodium and its activity. The hypothetical secondary sodium accident as analyzed may lead to the release of the primary sodium or liquid waste and so it is a potential precursor of an already identified accident.

(3) *Will the proposed change significantly reduce the margin of safety at the facility?*

No, adopting the new F1SAR as the licensing basis document for Fermi 1 will not decrease the margin of safety. It will establish an up-to-date licensing basis, so future changes can be appropriately evaluated against an updated safety analysis report. The F1SAR better describes the current condition of the plant. No physical changes will be implemented based on the submittal of the F1SAR. Some additional administrative requirements will be established in the new Quality Assurance program and in the need to keep the F1SAR updated biannually. No new types of accidents are discussed in the F1SAR—the discussion of the hypothetical secondary sodium event is a more detailed discussion of what potentially could happen during a catastrophic event leading to a sodium reaction. A total primary sodium release was already established as a licensing basis event. Because the F1SAR will not, in itself, lead to physical changes, but will be the new standard to which future changes are compared, establishment of this updated document as the Fermi 1 licensing basis will not significantly reduce the margin of safety of the facility.

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, NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161.

Attorney for licensee: John Flynn, Esquire, Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Branch Chief: John W. N. Hickey.

Duquesne Light Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of amendment request: December 19, 1997.

Description of amendment request: The proposed amendment would revise the requirements for the source range neutron flux channels in Mode 2 (Below P–6), 3, 4, and 5 to incorporate the guidance provided in NUREG–1431, the NRC's Improved Standard Technical Specifications (ISTS) with some modifications to address plant-specific design features. This change would allow (1) the use of alternate detectors provided the required functions are provided, and (2) plant cooldown with inoperable detectors provided the shutdown margin accounts for the temperature change. This change would also modify the Unit 2 Technical Specifications (TS) Table 3.3–1 Channels To Trip and Minimum Channels Operable requirements to 0 and 1, respectively. This portion of the amendment would make these Unit 2 requirements consistent with the current Unit 1 requirements. For both Units 1 and 2, TS Table 4.3–1 would be modified to include a notation exempting the alternate source range detectors from surveillance testing until they are repaired for operability.

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 amendment would modify the reactor trip system instrumentation requirements to permit the use of alternate detectors in place of inoperable source range detectors. The alternate detectors will be connected to the source range circuits to provide the required indications and functions. The alternate detectors are not required to be tested to satisfy the surveillance requirements until they are connected to the source range circuits and required to be operable. The alternate detectors must have the accuracy and sensitivity required to adequately monitor changes in the core reactivity levels. The alternate detectors will provide neutron flux monitoring in place of the source range detectors thus assuring core monitoring at a level consistent with the current technical specification requirements. Therefore, there is no loss of function or need for additional compensatory actions and the operators can perform required plant evolutions while relying on the alternate detectors.

Two operable detectors are required when the control rods are capable of withdrawal. Rod withdrawal and boron dilution add positive reactivity which can significantly affect the reactivity condition of the core, therefore, two monitors are required operable during startup evolutions. Redundant detectors are required to ensure that two source range neutron flux detectors are available to detect changes in core reactivity. These changes provide those indications and functions consistent with the current technical specification requirements where at least two source range detectors are operating and capable of providing the required functions. The function of the source range detectors is to provide direct neutron flux monitoring of the core to detect changes in reactivity which would result in a loss of the required shutdown margin.

One source range or alternate detector is required when the control rods are fully inserted and are not capable of withdrawal. Plant cooldown is recognized as a positive reactivity addition, however, this is accounted for in the shutdown margin calculations. The shutdown margin remains essentially unchanged and will be available to preclude a criticality event during this evolution. Inadvertent control rod withdrawal is not a concern, therefore, one source range or alternate detector can adequately monitor the core neutron flux. The action statements have been modified to address the NUREG–1431 Improved Standard Technical Specification (ISTS) requirements along with incorporating the ability to use alternate detectors in place of the source range detectors.

Bases 3/4.3.1 and 3/4.3.2, Protective and Engineered Safety Features (ESF) Instrumentation, has been revised to include the modifications to the source range detector requirements including the use of alternate source range detectors. The alternate detectors must provide sufficient accuracy and sensitivity to adequately monitor changes in core reactivity during Modes 2 (Below P–6), 3, 4, and 5.

The operability requirements of the source range neutron flux instrumentation will continue to be met when using an alternate detector in place of a source range neutron flux detector. No changes are being incorporated that would act to increase the probability of a positive reactivity addition event, therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The function of the source range detectors is to provide direct neutron flux monitoring of the core to detect positive reactivity additions which would result in a loss of the required shutdown margin. The alternate detectors must provide the accuracy and sensitivity required to adequately monitor changes in the core reactivity levels during shutdown and startup activities. The alternate monitors will be connected to the source range circuits to provide the required indications and functions. Therefore, there is no loss of function or need for additional compensatory actions and plant shutdown and startup activities can be continued while relying on the alternate detectors.

Control rod withdrawal is a method capable of providing rapid positive reactivity addition with boron dilution being a much slower positive reactivity addition method. With the control rods capable of withdrawal, a rod withdrawal event could rapidly initiate core criticality so redundant source range detectors are required operable. This ensures adequate monitoring capability is available to alert the operators of a rapid increase in the core reactivity condition. The maximum reactivity addition due to the boron dilution is slow enough to allow the operator to determine the cause and take corrective action before the shutdown margin is lost. These changes will not affect the operability or reliability of the source range instrumentation to provide the required indications and functions. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will continue to ensure the required source range instrumentation functions are available during shutdown and startup conditions. This change will not reduce the reliability of the source range detectors to monitor the core reactivity condition and provide the appropriate indications or affect the required shutdown margin. Plant operation will continue to be maintained within the shutdown margin requirements of [Technical] Specification 3.1.1.1 and 3.1.1.2. The required indications and functions are still maintained in accordance with current technical specification requirements and the shutdown margin is unaffected, therefore, the proposed change will 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: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

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

NRC Project Director: John F. Stolz.

Duquesne Light Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit 2, Shippingport, Pennsylvania

Date of amendment request: January 29, 1998.

Description of amendment request: The proposed amendment would revise the Beaver Valley Power Station, Unit No. 2, Updated Final Safety Analysis Report (UFSAR) calculated doses to address a non-conversative assumption regarding control room emergency pressurization fan flow during the Locked Rotor accident and include new X/Q values in calculating the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) doses.

This change is not the result of hardware changes to the plant or a change in operating practices. It reflects corrected analysis results only and allows correction of the licensing basis to reflect conservative assumptions used in the revised dose analysis for a Locked Rotor event.

The proposed amendment would also revise USFAR Tables 15.0–13, 15.6–15 and 15.6–16 to modify calculation parameters and UFSAR Section 15.6.5.5 to include editorial changes to ensure that descriptions of the Small Break Loss of Coolant Accident (SBLOCA) radiological consequences are clear. The following items in the UFSAR description of the SBLOCA radiological consequences analysis were changed: (1) a new lower minimum control room emergency pressurization fan flow rate and (2) a new lower minimum air bottle discharge rate.

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?

[Locked Rotor Accident]

The proposed amendment would revise the calculated control room doses for a Locked Rotor accident to address a non-conservative assumption for the fan pressurization system flow rate. The proposed amendment does not affect the capability of the control room habitability system to maintain control room dose within the limits of General Design Criterion (GDC) 19 in Appendix A of the Code of Federal Regulations Title 10 Part 50. The control room habitability system is an accident mitigation system and will continue to operate as designed. The system has no accident prevention function nor does it interact with systems that have such a function. The proposed change does not alter plant systems, structures or components.

The proposed amendment would also revise calculated offsite doses resulting from a locked rotor accident. This change in doses is not due to physical plant changes, but results mainly from use of more conservative assumptions used in calculating doses.

The proposed change does not affect the manner in which the plant is operated. The physical plant equipment and operating practices are not changed; therefore, the probability of an accident previously evaluated remains unchanged.

The performance requirements of the plant systems which are required to minimize the radiological consequences of a Locked Rotor accident remain unchanged. The proposed change slightly increases calculated control room doses due to an analysis input change for filtration fan flow rate. This slight increase remains below the limits required by GDC 19. The proposed change does not involve a significant increase in the consequences of an accident previously evaluated since adequate control room radiation protection continues to be provided to ensure actions can be taken to operate the plant safely under accident conditions. The radiological consequences to the environment from a Locked Rotor accident remain unchanged since the performance of plant systems remains unchanged. Although slightly increased, revised calculated offsite doses remain less than 10 CFR 100 limits.

[SBLOCA]

The proposed amendment would revise the control room dose analysis parameters for a Small Break Loss of Coolant Accident (SBLOCA) to include more conservative assumptions for the pressurization system flow rate. The proposed amendment does not affect the capability of the control room habitability system to maintain control room dose within the limits of General Design Criterion (GDC) 19 in Appendix A of the Code of Federal Regulations Title 10 Part 50. The control room habitability system is an accident mitigation system and will continue to operate as designed. The system has no accident prevention function nor does it interact with systems that have such a function. The proposed change does not alter plant systems, structures or components.

The proposed change does not affect the manner in which the plant is operated. The physical plant equipment and operating practices are not changed; therefore, the probability of an accident previously evaluated remains unchanged.

The performance requirements of the plant systems which are required to minimize the radiological consequences of a SBLOCA remain unchanged. The proposed change slightly decreases calculated control room doses due to analysis input changes. Calculated doses remain below the limits required by GDC 19.

Based on the above discussion, it is concluded that th[e] proposed change[s] [do] not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

[Locked Rotor Accident]

The proposed change does not alter the method of operating the plant nor does it pose additional challenges to the design or function of the control room habitability system. The control room habitability system will continue to operate as designed. The control room habitability system will continue to maintain the control room dose consequences within the limits specified in GDC 19. Adequate control room radiation protection will continue to be provided to ensure actions can be taken to operate the plants safely under accident conditions. The proposed change to the control room dose is only the result of a change in analysis input parameters. Plant performance has not been modified in any way which affects doses to the public.

[SBLOCA]

The proposed change does not alter the method of operating the plant nor does it pose additional challenges to the design or function of the control room habitability system. The control room habitability system will continue to operate as designed. The control room habitability system will continue to maintain the control room dose consequences within the limits specified in GDC 19. Adequate control room radiation protection will continue to be provided to ensure actions can be taken to operate the plants safely under accident conditions. The proposed change to the control room dose is only a result of an analysis being revised. Plant performance has not been modified in any way which affects doses to the public.

Therefore, the proposed change[s] [do] not create the possibility of a new or different kind of accident from any accident previously evaluated. Although no new types of accidents are created, the analysis represents a new methodology different than any evaluated previously by the NRC.

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

[Locked Rotor Accident]

The slight increase in calculated control room dose as a result of assuming increased fan flow does not result in exceeding the limits prescribed in GDC 19. Calculated doses to the public are slightly increased, but not as a result of physical changes. The proposed change will not result in any additional challenges to plant equipment including the fuel and reactor coolant system pressure boundary since adequate control room radiation protection will continue to be provided. The control room habitability system will continue to provide adequate radiation protection to ensure actions can be taken to operate the plant safely under accident conditions. The offsite doses increase slightly; however, the calculated dose results remain less than 10 CFR 100 limits. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

[SBLOCA]

The slight decrease in calculated control room dose as a result of the revised analysis does not result in exceeding the limits prescribed in GDC 19. The proposed change will not result in any additional challenges to plant equipment including the fuel and reactor coolant system pressure boundary since adequate control room radiation protection will continue to be provided. The control room habitability system will continue to provide adequate radiation protection to ensure actions can be taken to operate the plant safely under accident conditions. [Therefore, the NRC staff concludes that the revision to the SBLOCA analysis does 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: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

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

NRC Project Director: John F. Stolz.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2 (NMP2), Oswego County, New York

Date of amendment request: February 5, 1998.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) to update the terminology and references to 10 CFR 50.55a(f) and (g) consistent with the 1989 edition of Section XI of the American Society of Mechanical Engineer Boiler and Pressure Vessel Code (ASME Code). These changes, in effect, provide for consistency between (1) the NMP2 TS, (2) the second 10-year interval of the Inservice Inspections (ISI) and Inservice Testing (IST) Program Plans for NMP2, and (3) the requirement of 10 CFR 50.55a that the ISI/IST activities conducted during successive 10-year intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code that was in effect 12 months before the start of the 10-year interval.

Specifically, TS 4.0.5 would be changed to reference 10 CFR 50.55a(f) for the second 10-year IST Program and 10 CFR 50.55a(g) for the second 10-year ISI Program. The proposed changes to TS Table 4.3.7.5–1 and TS 4.4.3.2.2 would replace the references to ASME Section XI with references to criteria in the IST Program. The changes to TS 3.4.9.1 and 3.4.9.2 would add the phrase ''system leakage'' to notes that identify testing conditions when the shutdown cooling mode loop may be removed from service. Changes to TS 4.8.1.1.2.h.2 would correct a typographical error for which a reference to ASME Code Section II should refer to Section XI. Appropriate changes would be made to the TS index. Editoral changes to several other TS (i.e., TS 3/4.4.6.1, TS Figure 3.4.6.1–1, TS 3/4.10.7, TS Bases 3/4.4.6, TS Bases 3/4.10.7, and TS Table 5.7.1–1) would make references to ''hydrostatic testing'' and ''leak testing'' conform to the terminology to be used in the second 10-year ISI/IST Programs.

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

1. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The changes to the TS will ensure that TS reflect the correct 10CFR references and the terminology of the second NMP2 10-year ISI/IST program. The proposed revisions replace references to ASME Section XI with references to criteria in the Inservice Testing Program. The performance of system leakage testing is added to notes that identify conditions when the shutdown cooling mode loop may be removed from service. The other changes are editorial changes only to ensure that TS reflect the second 10-year ISI/IST program. One of the changes corrects a typographical error. These proposed changes do not affect the inspections or tests performed under the ISI/IST Program and will not result in any changes to the plant. None of the precursors of previously evaluated accidents are affected and therefore, the probability of an accident previously evaluated is not increased.

The changes will not affect the safety function of any equipment covered by the ISI/IST program. Therefore, these changes will not involve a significant increase in the consequences of an accident previously evaluated.

2. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of

a new or different kind of accident from any accident previously evaluated.

The changes to the TS will ensure that TS reflect the correct 10CFR references and the terminology of the second NMP2 10-year ISI/IST program. One of the changes corrects a typographical error. No physical modification of the plant is involved and no changes to the methods in which plant systems are operated are required. These changes do no affect the inspections or tests performed under the ISI/IST Program. The changes do not introduce any new failure modes or conditions that may create a new or different accident. Therefore, the changes do not by themselves create the possibility of a new or different kind of accident [from any accident] previously evaluated.

3. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The changes to the TS will ensure that TS reflect the correct 10CFR references and the terminology of the second NMP2 10-year ISI/IST program. One of the changes corrects a typographical error. No physical modification of the plant is involved and no changes to the methods in which plant systems are operated are required. The changes do not adversely affect any physical barrier to the release of radiation to plant personnel or to the public. These changes do not affect the inspections or tests performed under the ISI/IST Program. Therefore, these changes do not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

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

NRC Project Director: S. Singh Bajwa.

Philadelphia Electric Company, Docket Nos. 50–352 and 50–353, Limerick Generating Station (LGS), Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: January 27, 1998.

Description of amendment request: The proposed changes to the LGS, Units 1 and 2 Technical Specifications (TS) will revise the TS Table 3.6.3–1, ''Part A—Primary Containment Isolation Valves,'' by removing the numerical maximum stroke time for penetration 210, ''HPCI [High Pressure Coolant Injection] Turbine Exhaust,'' and adding a notation that the isolation time is not required.

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 Technical Specifications changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Changes to Technical Specifications regarding the removal of the High Pressure Coolant Injection (HPCI) Turbine Exhaust Valve maximum stroke times do not change the frequency or consequences of any accident previously evaluated.

The proposed changes do not change the function of the HPCI system nor any safety function of the valve as described in the SAR [Safety Analysis Report]. The isolation stroke times are not limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. The stroke times do not detect or indicate an abnormal degradation of the reactor coolant pressure boundary. The stroke times are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The stroke times are not part of a component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The stroke times are not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Therefore, the changes will not increase the probability or consequences of an accident previously evaluated.

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

The proposed Technical Specifications changes regarding the removal of the High Pressure Coolant Injection (HPCI) Turbine Exhaust Valve maximum stroke times do not affect the probability of a malfunction of equipment important to safety. Safety related HPCI system operation occurs with the subject valve passively open. This valve would only be manually closed under events where there was a need to isolate the HPCI system from the suppression pool. The manual closing of the valve may occur under these events and is controlled by station procedures. Given that these procedurally mandated valve isolations are all via remote manual means, valve isolation time is not a critical parameter requiring specific acceptance criteria.

The Inservice Testing (IST) Program will still maintain an IST program basis maximum stroke time for HV–055–1(2)F072 to establish action and alert levels for valve performance monitoring. These performance based values, in conjunction with diagnostic test criteria, are used for motor operated valve material condition monitoring and trending. Therefore, eliminating the subject maximum isolation time requirement from TS will not increase the probability of malfunction of the valve since the principal means of monitoring valve performance remains unchanged.

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

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

There is no defined margin of safety for remote manual valve isolation times discussed in Technical Specification Bases. In addition, the valve maximum stroke time will be retained in the IST program.

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.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, PA 19464.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, PA 19101.

NRC Project Director: John F. Stolz.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of amendment requests: June 30, 1997.

Description of amendment requests: The licensee proposes to delete SONGS Unit 2 License Condition 2.C.(19)b, ''Shift Manning,'' and revise SONGS Units 2 and 3 Technical Specifications (TS) 3.3.1, ''Reactor Protective Instrumentation (RPS)-Operating,'' TS 3.3.2, ''Reactor Protective Instrumentation (RPS)-Shutdown,'' TS 3.3.5, ''Engineered Safety Features Actuation System (ESFAS) Instrumentation,'' TS 3.3.10, ''Fuel Handling Isolation Signal (FHIS),'' TS 3.3.11, ''Post Accident Monitoring Instrumentation,'' TS 3.4.7, ''RCS Loops—Mode 5, Loops Filled,'' TS 3.4.12.1, ''Low Temperature Overpressure Protection (LTOP) System,'' TS 3.7.5, ''Auxiliary Feedwater (AFW) System,'' TS Section 5.5.2.10, ''Inservice Testing Program,'' and TS Section 5.5.2.11, ''Steam

Generator (SG) Tube Surveillance Program.'' The proposed changes are required to either: reinstate provisions of the SONGS Units 2 and 3 TS, revised as part of NRC Amendment Numbers 127 and 116, make corrections to the TS, or remove information inadvertently added that is not applicable.

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.

Proposed Technical Specification Change Number NPF–10/15–475 (PCN–475) addresses modifications to the Technical Specifications for San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 approved by NRC Amendment Nos. 127 and 116. NRC Amendment Numbers 127 and 116 approved changes to adopt the recommendations of NUREG–1432, ''Standard Technical Specifications Combustion Engineering Plants,'' requested through Proposed Technical Specification Change Number NPF–10/15–299 (PCN–299). The proposed changes were identified during drafting of the procedure changes required to implement NRC Amendment Numbers 127 and 116, and during the self-assessment performed by Southern California Edison (SCE).

The proposed change is required to either: reinstate provisions of the SONGS Units 2 and 3 Technical Specifications, revised as part of NRC Amendment Numbers 127 and 116, for SONGS Units 2 and 3, make corrections to the Technical Specifications, or remove information inadvertently added that is not applicable.

Proposed Change 1 would delete License Condition 2.C.(19)b for SONGS Unit 2 only. Presently, overtime restrictions are specified in both the license condition and the Topical Report. Through NRC Amendment Numbers 127 and 116, the shift manning requirements were modified and subsequently moved to the Section 5.5.2.e, with details moved to the Topical Report.

In addition, in the NRC's Safety Evaluation Report related to the ''Issuance of Amendment for San Onofre Nuclear Generating Station, Unit No. 2 (TAC No. M86191) and Unit No. 3 (TAC No. M86192),'' dated February 9, 1996, it is stated that the staff has determined on a generic basis, that specific overtime limits need not be specified in technical specifications, as they are not required by 10 CFR 50.36 (c)(5). The staff also concluded that control of this matter through administrative procedures provides reasonable assurance that personnel overtime would not jeopardize safe plant operation and that specific overtime limits and associated procedures could be described in the UFSAR, or other licensee controlled documents incorporated in the UFSAR by reference for which further changes can be made pursuant to 10 CFR 50.59.

Retaining a separate license condition provides no function, is inconsistent with the Topical Report, and therefore, should be deleted. There can be no increase in the probability or consequences of any accident previously evaluated as a result of this change, as the change does not revise or reduce commitments, it is solely for clarity.

Proposed change 2 would revise TS 3.3.1, ''Reactor Protective Instrumentation (RPS)—Operating,'' to delete the exception of the power range neutron flux channels from Surveillance Requirement (SR) 3.3.1.7. TS 3.3.1 requires that four RPS trip and operating bypass removal channels for each function covered by this specification be operable in the applicable Modes. SR 3.3.1.7 requires that a channel functional test be performed on each RPS channel, except the power range neutron flux channels. Therefore, the proposed change would delete the exception to SR 3.3.1.7 for the power range neutron flux channels. Under the former Technical Specifications, the power range neutron flux channels were not exempt from the channel functional test.

Proposed change 3 would revise SR 3.3.2.5 of TS 3.3.2, ''Reactor Protective Instrumentation (RPS)-Shutdown.'' SR 3.3.2.5 requires that the RPS response time be verified within limits every 24 months on a staggered test basis. SR 3.3.1.13 of TS 3.3.1 also requires that response time tests be performed every 24 months on a staggered test basis. However, neutron detectors presently are excluded from response time testing in Modes 1 and 2. Therefore, the proposed change will add a note to SR 3.3.2.5 to allow exclusion of neutron detectors from response time testing. Under the former Technical Specifications, the neutron detectors were exempt from response time testing.

Proposed change 4 would revise SR 3.3.5.4. SR 3.3.5.4 requires that a channel calibration of the Recirculation Actuation Signal (RAS), including the bypass removal function, be performed. However, a bypass removal function is not part of the RAS design. A change is required therefore, to delete the bypass removal function, as it is not a part of the RAS function. Because the RAS function does not utilize the bypass removal function, eliminating the words from the SR cannot increase the probability or consequences of any accident previously evaluated as a result of this change.

Proposed change 5 would revise Technical Specification (TS) 3.3.10, ''Fuel Handling Isolation Signal (FHIS).'' Specifically, the proposed change would revise the allowable value specified in SR 3.3.10.2 for the required FHIS monitor, from ''less than or equal to 6E4 cpm above background,'' to ''Sufficiently high to prevent spurious alarms/trips, yet sufficiently low to assure an alarm/trip should an inadvertent release occur.''

The 6E4 cpm setpoint does not provide adequate margin above and beyond background during a normal refueling outage. Thus, the proposed setpoint, which can be set greater than the highest ambient background level, but remains well below the calculated monitor response to a fuel handling accident, would provide that margin, and was previously specified in the former Technical Specifications.

The proposed change would permit relocation of the allowable value for the monitors from the Technical Specifications to the administrative control procedures. This change is consistent with the existing Containment Airborne Radiation Monitor Specification. This change will not prevent the radiation monitors from performing their intended function following a design basis accident.

The consequences of a Fuel Handling Accident inside the FHB have been evaluated, assuming no FHB isolation. The results of the calculation indicated off-site, and control room doses with control room isolation within three minutes, are well within the limits established by the NRC guidelines.

Compliance with this statement would provide suitable confirmation that the monitors will be capable of performing their intended function, and is further justified by the fact that no credit was given to the monitors in the radiological dose analysis.

This change will not involve a significant increase in the probability of any accident previously evaluated because the setpoint is not an accident initiator. The consequences of an accident would not be increased either as the administrative value would be set sufficiently low to assure an alarm/trip should an inadvertent release occur. The actual values would be administratively controlled by quality-affecting procedures (i.e., changes to procedures will be evaluated under 10 CFR 50.59).

In addition, a typographical error in SR 3.3.10.3 would be corrected. The SR Note would be revised to refer to ''initiation relay,'' not ''ignition relay.'' This change will not involve a significant increase in the probability of any accident previously evaluated because it corrects a typographical error only.

Proposed change 6 would revise Function 6 of Table 3.3.11–1. Currently, Function 6 refers to Containment Sump Water Level (wide range). However, Function 6 is the combined function of the wide range emergency sump level transmitters, and the containment area level transmitters. Therefore, the description of the combination should not be the description of the function of the single transmitter. There can be no increase in the probability or consequences of any accident previously evaluated as a result of this change, as the change does not revise or reduce commitments, it is solely for clarity.

Proposed change 7 would revise Surveillance Requirement 3.4.7.2 of TS 3.4.7. The change would remove an inconsistency between what is specified in the Limiting Condition for Operation (LCO), and what is required to be verified by the SR. The proposed change conservatively removes the inconsistency by revising SR 3.4.7.2 to specify that the required steam generator secondary side water level be verified greater than 50% (wide range). This change is for clarity only, and is consistent with existing station procedures and operation of the facility.

Proposed change 8 would revise TS 3.4.12.1, ''Low Temperature Overpressure

Protection (LTOP) System.'' Specifically, the Applicability would be revised to clarify the Mode 6 applicability. The Applicability should read ''Mode 6 when the head is on the reactor vessel and the RCS is not vented.'' This change is intended to clarify the Applicability of TS 3.4.12.1 in Mode 6, and also reflects the previous requirements of former TS 3/4.4.8.3.1, ''Overpressure Protection Systems RCS Temperature less than or equal to 256'F.'' This change is editorial only and there can be no increase in the probability or consequences of any accident previously evaluated as a result of this change.

Proposed change 9 would revise SR 3.7.5.3 and SR 3.7.5.4 of TS 3.7.5, ''Auxiliary Feedwater (AFW) System.'' Presently, SR 3.7.5.3 requires that AFW automatic valves actuate to their correct position on an actual or simulated signal when in Mode 1, 2, or 3 (except valves HV–8200 and HV–8201) and SR 3.7.5.4 requires that each AFW pump starts automatically on an actual or simulated signal when in Mode 1, 2, or 3. The Bases, however, for these SRs makes it clear that the tests are a refueling surveillance which should be performed in Mode 5. The proposed change will delete the reference to Modes 1, 2, and 3 from both SR 3.7.5.3 and 3.7.5.4.

The intent of the wording for the SR is to perform the test in Mode 5 in order to demonstrate the operability of the system in Modes 1, 2, and 3. This change would also be consistent with the former SRs which previously specified that the surveillances were required to be performed at least once per refueling interval during shutdown. Therefore, there can be no increase in the probability or consequences of any accident previously evaluated as a result of this change.

Proposed change 10 would revise Section 5.5.2.10, ''Inservice Testing Program.'' The change will clarify that this section applies not only to the Inservice Testing Program, but includes the Inservice Inspection Program as well. This change is editorial in that it correctly identifies the intent of this section. As this is an editorial change only, there can be no increase in the probability or consequences of any accident previously evaluated as a result of this change.

Proposed change 11 would revise Section 5.5.2.11 to correct typographical errors. A table is provided that identifies supplemental sampling requirements for steam generator tube inspections. However, the table is numbered incorrectly. The proposed change would correct the table number.

In addition, under the table heading ''Action Required'' for both the first ''1st Sample Inspection'' and ''2nd Sample Inspection,'' for result C–3, notification is to be made to the NRC, and an incorrect reference to 10 CFR 50.72 is made. The proper notification is pursuant to 10 CFR 50.73. The proposed change would correct this reference. Also under the ''Action Required'' heading for the ''1st Sample Inspection'' for Result C2, is a typographical error. It is currently written, ''Plug defective tubes and inspect an additional 25 tubes in this SG.'' However, the statement should read, ''Plug defective tubes and inspect an additional 2S tubes in this SG.'' The proposed requirement is consistent with the requirement of the former TS 3/4.4.4, ''Steam Generators.''

Operation of the facility would remain unchanged as a result of the proposed changes as the changes correct typographical errors. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any 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 would either: reinstate provisions of the former SONGS Units 2 and 3 Technical Specifications, make corrections to the Technical Specifications, or remove information inadvertently added that is not applicable to SONGS Units 2 and 3.

Proposed change 1 deletes the SONGS Unit 2 license condition regarding shift manning requirements as it conflicts with the requirements contained in the revised Technical Specifications and the Topical Report. Operation of the facility would remain unchanged as a result of the proposed changes and could not create the possibility of a new or different kind of accident from any previously evaluated.

Proposed change 2 would revise TS 3.3.1, ''Reactor Protective Instrumentation (RPS)-Operating,'' to delete the exception of the power range neutron flux channels from Surveillance Requirement (SR) 3.3.1.7. SR 3.3.1.7 requires that a channel functional test be performed on each RPS channel, except the power range neutron flux channels. Therefore, the proposed change would delete the exception to SR 3.3.1.7 for the power range neutron flux channels. This change will not create the possibility of a new or different kind of accident from any previously evaluated.

Proposed change 3 would revise SR 3.3.2.5 of TS 3.3.2, ''Reactor Protective Instrumentation (RPS)-Shutdown.'' SR 3.3.2.5 requires that the RPS response time be verified within limits every 24 months on a staggered test basis. SR 3.3.1.13 of TS 3.3.1 also requires that response time tests be performed every 24 months on a staggered test basis. However, neutron detectors presently are excluded from response time testing in Modes 1 and 2. Therefore, the proposed change will add a note to SR 3.3.2.5 to allow exclusion of neutron detectors from response time testing. The proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

Proposed change 4 would revise Surveillance Requirement (SR) 3.3.5.4. A change is required to delete the bypass removal function, as it is not a part of the RAS function. Because the RAS function does not utilize the bypass removal function, eliminating the words from the SR cannot create the possibility of a new or different kind of accident from any previously evaluated.

Proposed change 5 revises the FHIS the monitor allowable value. The value would be controlled by administrative procedures. This change would not alter the design and operational interface between the FHIS and existing plant equipment. As such, the monitors would continue to operate and perform their intended safety function to isolate the FHB following a design basis accident as before. In addition, the Note to SR 3.3.10.3 would be corrected to read ''* * * verification of the proper operation of each initiation relay.'' Therefore, operation of the facility in accordance with this proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Proposed change 6 revises the name of Function 6 of Table 3.3.11–1. Currently, Function 6 refers to Containment Sump Water Level (wide range), and is more correctly specified as the Containment Water Level (wide range). The proposed change cannot create the possibility of a new or different kind of accident from any accident previously evaluated as the change only revises the name of an instrument and is solely for clarity.

Proposed change 7 would remove an inconsistency between what is specified in the LCO, and what is required to be verified by the SR. The proposed change conservatively removes the inconsistency by revising SR 3.4.7.2 to specify that the required steam generator secondary side water level be verified greater than 50% (wide range). This change is for clarity only, is consistent with existing station procedures, and consistent with operation of the facility. The proposed change cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

Proposed change 8 would revise TS 3.4.12.1, ''Low Temperature Overpressure Protection (LTOP) System.'' Specifically, the Applicability would be revised to clarify the Mode 6 applicability. The Applicability should read ''Mode 6 when the head is on the reactor vessel and the RCS is not vented.'' This change is intended to clarify the Applicability of TS 3.4.12.1 in Mode 6, and also reflects the previous requirements of former TS 3/4.4.8.3.1, ''Overpressure Protection Systems RCS Temperature less than or equal to 256°F.'' This change is editorial only and cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

Proposed change 9 would revise SR 3.7.5.3 and SR 3.7.5.4 of TS 3.7.5, ''Auxiliary Feedwater (AFW) System,'' to delete the requirements that the SRs be performed in Mode 1, 2, or 3. The intent of the wording for the SR is to perform the test in Mode 5 in order to demonstrate the operability of the system in Modes 1, 2, and 3. This change would also be consistent with the former SRs which previously specified that the surveillances were required to be performed at least once per refueling interval during shutdown. Therefore, the proposed change cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

Proposed change 10 would revise Section 5.5.2.10, ''Inservice Testing Program.'' The change will clarify that this section applies not only to the Inservice Testing Program, but includes the Inservice Inspection Program as well. This change is editorial in that it correctly identifies the intent of this section. As this is an editorial change only, and cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

Proposed change 11 would revise Section 5.5.2.11 to correct typographical errors. A table is provided that identifies supplemental sampling requirements for steam generator tube inspections. Operation of the facility in accordance with this proposed change will 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 will either: reinstate provisions of the SONGS Units 2 and 3 Technical Specifications, make corrections to the Technical Specifications, or remove information inadvertently added that is not applicable to SONGS Units 2 and 3. Operation of the facility would remain unchanged as a result of the proposed change. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

Proposed change 1 deletes the SONGS Unit 2 license condition regarding shift manning requirements as it conflicts with the requirements contained in the revised Technical Specifications and the Topical Report. The NRC staff has concluded that control of overtime restrictions through administrative procedures provides reasonable assurance that personnel overtime would not jeopardize safe plant operation and that specific overtime limits and associated procedures could be described in the UFSAR, or other licensee controlled documents incorporated in the UFSAR by reference for which further changes can be made pursuant to 10 CFR 50.59. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

Proposed change 2 would revise TS 3.3.1, ''Reactor Protective Instrumentation (RPS)—Operating,'' to delete the exception of the power range neutron flux channels from Surveillance Requirement (SR) 3.3.1.7. SR 3.3.1.7 requires that a channel functional test be performed on each RPS channel, except the power range neutron flux channels. Therefore, the proposed change would delete the exception to SR 3.3.1.7 for the power range neutron flux channels. This change will not involve a significant reduction in a margin of safety.

Proposed change 3 would revise SR 3.3.2.5 of TS 3.3.2, ''Reactor Protective Instrumentation (RPS)-Shutdown.'' SR 3.3.2.5 requires that the RPS response time be verified within limits every 24 months on a staggered test basis. SR 3.3.1.13 of TS 3.3.1 also requires that response time tests be performed every 24 months on a staggered test basis. However, neutron detectors presently are excluded from response time testing in Modes 1 and 2. Therefore, the proposed change will add a note to SR 3.3.2.5 to allow exclusion of neutron detectors from response time testing. The proposed change will not involve a significant reduction in a margin of safety.

Proposed change 4 would delete the bypass removal function, as it is not a part of the RAS function. Because the RAS function does not utilize the bypass removal function, eliminating the words from the SR cannot involve a significant reduction in a margin of safety.

Proposed change 5 would revise the FHIS monitor allowable values and would not alter the existing margin of safety. The change would only relinquish control of the allowable values from the TSs to quality-affecting (changes will require a 10 CFR 50.59 evaluation) procedures. In addition, the proposed change would correct a typographical error in the Note to SR 3.3.10.3. Therefore, operation of the facility will not involve a significant reduction in a margin of safety.

Proposed change 6 revises the name of Function 6 of Table 3.3.11–1. Currently, Function 6 refers to Containment Sump Water Level (wide range), and is more correctly specified as the Containment Water Level (wide range). The proposed change cannot involve a significant reduction in a margin of safety.

Proposed change 7 would remove an inconsistency between what is specified in the LCO, and what is required to be verified by the SR. The proposed change conservatively removes the inconsistency by revising SR 3.4.7.2 to specify that the required steam generator secondary side water level be verified greater than 50% (wide range). This change is consistent with existing station procedures, and consistent with operation of the facility. The proposed change cannot involve a significant reduction in a margin of safety.

Proposed change 8 would revise TS 3.4.12.1, ''Low Temperature Overpressure Protection (LTOP) System.'' Specifically, the Applicability would be revised to clarify the Mode 6 applicability. The Applicability should read ''Mode 6 when the head is on the reactor vessel and the RCS is not vented.'' This change is intended to clarify the Applicability of TS 3.4.12.1 in Mode 6, and also reflects the previous requirements of former TS 3/4.4.8.3.1, ''Overpressure Protection Systems RCS Temperature less than or equal to 256°F.''

Proposed change 9 would revise SR 3.7.5.3 and SR 3.7.5.4 of TS 3.7.5, ''Auxiliary Feedwater (AFW) System,'' to delete the requirements that the SRs be performed in Mode 1, 2, or 3. The intent of the wording for the SR is to perform the test in Mode 5 in order to demonstrate the operability of the system in Modes 1, 2, and 3. Therefore, the proposed change cannot involve a significant reduction in a margin of safety.

Proposed change 10 would revise Section 5.5.2.10, ''Inservice Testing Program.'' The change will clarify that this section applies not only to the Inservice Testing Program, but includes the Inservice Inspection Program as well. This change is editorial in that it correctly identifies the intent of this section. This is an editorial change only.

Proposed change 11 would revise Section 5.5.2.11 to correct typographical errors. Operation of the facility would remain unchanged as a result of the proposed changes and could not create the possibility of a new or different kind of accident from any previously evaluated.

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 requests involve no significant hazards consideration.

Local Public Document Room location: Main Library, University of California, Irvine, California 92713.

Attorney for licensee: T.E. Oubre, Esquire, Southern California Edison Company, P.O. Box 800, Rosemead, California 91770.

NRC Project Director: William H. Bateman.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: February 3, 1998.

Description of amendment request: The proposed changes will replace the augmented inspection requirements for the Reactor Coolant Pump flywheels specified by Regulatory Guide 1.14, ''Reactor Coolant Pump Integrity,'' Revision 1, dated August 1975, with those established by WCAP–14535A, ''Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination,'' dated November 1996, and will eliminate the inspection requirements for the flow straighteners.

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:

Virginia Electric and Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed changes for the North Anna Units 1 and 2 and determined that a significant hazards consideration is not involved.

(a) The elimination of the inspection requirements for the flow straighteners, and the reduction of the inspection requirements for the reactor coolant pump flywheels as granted by the NRC and supported by WCAP–14535A do not significantly increase the probability of an accident previously evaluated in the safety analysis report.

The surveillance frequency changes for the reactor coolant pump flywheels are based upon the technical basis of the Westinghouse Energy Systems Topical Report WCAP–14535A. The results of WCAP–14535A have been reviewed, evaluated, and accepted for referencing in license applications by the NRC in their letter entitled ''Acceptance for Referencing of Topical Report WCAP–14535, Topical Report on Reactor Coolant Pump

Flywheel Inspection Elimination'' dated September 12, 1996.

The proposed surveillance (inspection) requirements only reduce the inspection frequency for the reactor coolant pump flywheels and eliminate the inspection requirements for the flow [straighteners]. There is no change in the method of plant operation or system design. Therefore, the proposed changes do not increase the probability of occurrence or the consequences of any previously analyzed accident.

(b) The proposed changes for the elimination of the inspection requirements for the flow straighteners, and for the reduction in inspection requirements for the reactor coolant pump flywheels as granted by the NRC and supported by WCAP–14535A do not create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report.

The proposed surveillance (inspection) requirements only reduce the inspection frequency for the reactor coolant pump flywheels and eliminate the inspection requirements for the flow [straighteners] in Unit 1. There is no change in the method of plant operation or system design. Therefore, there are no new or different kinds of accident or malfunction from any accidents previously evaluated.

(c) The proposed changes for the elimination of the inspection requirements for the flow straighteners, and for the reduction in inspection requirements for the reactor coolant pump flywheels as granted by the NRC and supported by WCAP–14535A do not impact the accident analysis assumptions or the basis of any Technical Specification. The revised inspection requirements only reduce the examination frequency for the reactor coolant pump flywheels and eliminate the inspection requirements for the flow [straightener] in Unit 1. Therefore, the proposed changes in surveillance (inspection) frequency do 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 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: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: Gordon E. Edison, Acting.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: February 3, 1998.

Description of amendment request: The proposed changes will allow the reactor trip bypass breakers to be tested in the racked-in position. This change will continue to ensure the operability of the breakers and eliminate unnecessary movement caused by racking the breakers, thus reducing the wear and tear on the breakers and the possibility of a reactor trip. The operation of the Reactor Protection System and the reactor trip and the reactor trip bypass breakers are not being changed. The proposed changes in the test sequence for the reactor trip bypass breakers continue to provide assurance that the reactor trip bypass breakers will operate as designed to mitigate the consequence of any unsafe or improper reactor operation during steady-state or transient power operations when the bypass breakers are placed in service for reactor trip system testing or trip breaker maintenance.

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:

Virginia Electric and Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed changes for the North Anna Units 1 and 2 and determined that a significant hazards consideration is not involved.

(a) Operation and testing of the reactor trip breakers does not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

The testing sequence will continue to ensure that the reactor trip system will be operable to mitigate the consequences of any unsafe or improper reactor operation during steady state or transient power operations. Although the breaker is placed in service before it is tested, the breaker is tested as soon as practicable to reestablish operability prior to performing testing of the reactor trip system or maintenance on the reactor trip breakers. During the short period of time the breaker is closed before the local shunt trip device test, the operability of the breaker is established based on satisfactory breaker testing conducted during the previous surveillance interval. Changing the minimum channels operable requirement for the reactor trip bypass breakers does not affect the operation of the reactor trip system since only one reactor trip breaker can be inservice for testing or maintenance of the reactor protection system. Therefore, the proposed test sequence does not significantly increase the probability of occurrence or the consequences of any previously analyzed accident.

(b) The proposed Technical Specifications do not create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report.

The proposed test sequence change does not alter the actual test performed to establish operability of the reactor trip bypass breakers. The bypass breakers will be proven operable prior to reactor trip system testing or reactor trip breaker maintenance. Although the breaker is placed in service before it is tested, the breaker is tested as soon as practicable to reestablish operability prior to performing testing of the reactor trip system or maintenance on the reactor trip breakers. During the short period of time the breaker is closed before the local shunt trip device test, the operability of the breaker is established based on satisfactory breaker testing conducted during the previous surveillance interval. Changing the minimum channels operable requirement for the reactor trip bypass breakers does not affect the operation of the reactor trip system since only one reactor trip bypass breaker can be inservice for testing or maintenance of the reactor protection system. Therefore, it is concluded that no new or different kind of accident or malfunction from any previously evaluated has been created.

(c) The proposed Technical Specifications change does not result in a significant reduction in margin of safety.

The proposed change in the reactor trip bypass breaker test sequence provides assurance that the reactor trip system remains operable during normal operations or during reactor trip system testing and reactor trip breaker maintenance to mitigate the consequences of any unsafe or improper reactor operation. Changing the minimum channels operable requirement for the reactor trip bypass breakers does not affect the operation of the reactor trip system since only one reactor trip bypass breaker can be inservice for testing or maintenance of the reactor protection system. Therefore, the proposed change in the test sequence for the reactor trip bypass breaker does not significantly reduce 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 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: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: Gordon E. Edison, Acting.

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

Date of amendment request: October 13, 1997, as supplemented by a letter dated February 10, 1998.

Description of amendment request: The proposed amendment would revise the Kewaunee Technical Specifications (TS) to denote several changes. The proposed changes are: Relocating information to the Updated Safety Analysis Report (USAR), deleting redundant information, incorporating new references and deleting incorrect references, correcting errors, and augmenting existing requirements.

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 changes were revised in accordance with the provision of 10 CFR 50.92 to show no significant hazards exist. The proposed changes will not:

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

The likelihood that an accident will occur is neither increased nor decreased by these TS changes. The TS changes will not impact the function or method of operation of plant equipment. Thus, there is not a significant increase in the probability of a previously analyzed accident due to the changes. Since no plant practices have changed and no physical changes are being made, no systems, equipment, or components are affected by the proposed changes. Thus, the consequences of the malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR) are not increased by the changes.

The proposed changes are administrative in nature and, therefore, have no impact on accident initiators or plant equipment, and thus, do not affect the probabilities or consequences of an accident.

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

Operation of the facility in accordance with the proposed TS changes would not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve changes to the physical plant or operations. Since these administrative changes do not contribute to accident initiation, they do not produce a new accident scenario or produce a new type of equipment malfunction. Also, these changes do not alter any existing accident scenarios; they do not affect equipment or its operation, and thus, do not create the possibility of a new or different kind of accident.

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

Changes in the proposed amendment include relocating information to the USAR, deleting redundant information, incorporating new references, deleting incorrect references, correcting errors, and augmenting existing requirements. Operation of the facility in accordance with the proposed TS would not involve a significant reduction in a margin of safety. The proposed changes do not affect plant equipment or operation. Safety limits and limiting safety system settings are not affected by these proposed changes.

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.

NRC Project Director: Richard P. Savio.

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.

Duke Energy Corporation, Docket No. 50–270, Oconee Nuclear Station, Unit 2, Oconee County, South Carolina

Date of amendment request: January 15, 1998.

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) Table 4.1–1 and TS 4.5.2.1.2 to allow a one-time extension for specified Unit 2 refueling outage surveillances during operating cycle 16.

Date of publication of individual notice in the **Federal Register**: January 23, 1998 (63 FR 3593).

Expiration date of individual notice: February 23, 1998.

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: February 3, 1998.

Brief description of amendment request: The proposed amendment would change the operability requirement for the Standby Liquid Control system to Run/Power Operations and Startup.

Date of individual notice in **Federal Register**: February 26, 1998 (63 FR 9872).

Expiration date of individual notice: March 30, 1998.

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, SE., Cedar Rapids, Iowa 52401.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: February 3, 1998.

Brief description of amendment request: The proposed amendment would revise the definitions of Cold Condition and Cold Shutdown and add a new section, 3.17, Vessel Hydrostatic Pressure and Leak Testing, to the Technical Specifications to specifically allow reactor vessel hydrostatic pressure testing to be performed during plant shutdown.

Date of individual notice in **Federal Register**: February 26, 1998 (63 FR 9874).

Expiration date of individual notice: March 30, 1998.

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, SE., Cedar Rapids, Iowa 52401.

Southern Nuclear Operating Company, Inc, Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendment request: July 23, 1997, as supplemented September 30, October 27, and December 18, 1997, and February 12, 1998.

Description of amendment request: The July 23, 1997, application was previously noticed in the **Federal Register** on September 10, 1997 (62 FR 47699). In addition, the December 18, 1997, supplement provided additional information that revised the original licensee's evaluation of the no significant hazards consideration and, therefore, was noticed in the **Federal Register** on January 14, 1998 (63 FR 2281). The February 12, 1998, supplement provided additional information that revised the licensee's evaluation of the no significant hazards consideration. Therefore, renotification of the Commission's proposed determination of no significant hazards is necessary.

The proposed amendments would revise the Technical Specifications (TSs) by relocating the reactor coolant system (RCS) pressure and temperature limits from the TSs to the proposed Pressure Temperature Limits Report in accordance with the guidance provided by Generic Letter 96–03, ''Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits.'' TS 3.4.10.3 would be revised to require that two residual heat removal system suction relief valves be operable or that the RCS be vented at RCS indicated cold leg temperatures less than or equal to 325 °F. In addition, a new TS would be added to limit the operation of more than one reactor coolant pump below 110 °F.

Date of publication of individual notice in the **Federal Register**: February 23, 1998 (63 FR 9020).

Expiration date of individual notice: March 25, 1998.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302.

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.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of application for amendments: November 7, 1997.

Brief description of amendments: The amendments remove the 24/48 Volt direct current (Vdc) batteries and associated charger and distribution systems from the Unit 2 Technical Specifications. All safety-related loads associated with the 24/48 Vdc batteries for Unit 2 will be connected to other safety related battery systems which are in the TS.

Date of issuance: February 25, 1998.

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 165 and 160.

Facility Operating License Nos. DPR–19 and DPR–25: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 14, 1998 (63 FR 2277).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 25, 1998.

No significant hazards consideration comments received: No.

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

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of application for amendments: October 3, 1996.

Brief description of amendments: The amendments will correct a typographical error that was introduced into the Technical Specifications with the issuance of Amendment Nos. 150 and 145 issued on June 28, 1996.

Date of issuance: February 25, 1998.

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 166 and 161.

Facility Operating License Nos. DPR–19 and DPR–25: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 14, 1998 (63 Fr 2273).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 25, 1998.

No significant hazards consideration comments received: No.

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

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: October 15, 1997.

Brief description of amendments: The amendments eliminate unnecessary detail from the Accident Monitoring Instrumentation Surveillance Requirements (TS Table 4.3.7.5–1).

Date of issuance: February 17, 1998.

Effective date: Immediately, to be implemented prior to startup from L1F35 for Unit 1 and L2R07 for Unit 2.

Amendment Nos.: 123 and 108.

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 61841).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 17, 1998.

No significant hazards consideration comments received: No.

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

Duke Energy Corporation, Docket No. 50–270, Oconee Nuclear Station, Unit 2, Oconee County, South Carolina

Date of application for amendment: January 15, 1998.

Brief description of amendment: The amendment revises Technical Specifications (TS) Table 4.1–1 and Specification 4.5.2.1.2 to allow a one-time extension for specified Unit 2 refueling outage surveillances during operating cycle 16.

Date of issuance: February 23, 1998.

Effective date: As of the date of issuance to be implemented upon receipt.

Amendment No.: 228.

Facility Operating License No. DPR–47: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 23, 1998 (63 FR 3593).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 23, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691.

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 for amendments: February 2, 1998, as supplemented February 18, 1998.

Brief description of amendments: The amendments revise the wording used to specify refueling outage surveillances.

Date of issuance: February 26, 1998

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

Amendment Nos.: Unit 1–228; Unit 2–229; Unit 3–225.

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

Public comments requested as to proposed no significant hazards consideration: Yes. (63 FR 6784 dated February 10, 1998). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by March 12, 1998, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendments. The February 18, 1998, letter provided clarifying information that did not change the scope of the February 2, 1998, application and the no significant hazards consideration determination.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and a final no significant hazards consideration determination are contained in a Safety Evaluation dated February 26, 1998.

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC.

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina.

Florida Power and Light Company, et al., Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of application for amendment: August 22, 1997.

Brief description of amendment: Changes to the Technical Specifications (TS) to relocate the inservice testing program requirements from TS 4.0.5 to the Administrative Controls Section in the Unit 1 and 2 TS.

Date of Issuance: February 25, 1998.

Effective Date: February 25, 1998.

Amendment Nos.: 153 and 91.

Facility Operating License No. NPF–16: Amendment revised the TS.

Date of initial notice in **Federal Register**: September 24, 1997 (62 FR 50006).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 25, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Community College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34981–5596.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of application for amendment: October 21, 1997, as supplemented by letter dated February 3, 1998. The application superseded a previous application of May 16, 1997.

Brief description of amendment: This amendment revised administrative requirements regarding the unit staff positions of General Supervisor Operation and Manager Operations as stated in TS 6.2.2.i and 6.3.1.

Date of issuance: February 19, 1998.

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

Amendment No.: 160.

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

Date of initial notice in **Federal Register**: November 5, 1997 (62 FR 59916).

The February 3, 1998, letter provided clarifying information that did not change the no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 19, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

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

Date of application for amendment: March 27, 1997, as supplemented on September 25, 1997.

Brief description of amendment: The amendment revises Technical Specification (TS) Limiting Condition for Operation (LCO) 3.7.11 and Surveillance Requirement (SR) 4.7.11 for the ultimate heat sink. TS LCO 3.7.11 is changed to indicate that the ultimate heat sink is operable at a water temperature of less than or equal to 75 °F instead of an average value. The use of average when verifying the water temperature and the reference to a specific monitoring location are deleted in TS SR 4.7.11.a and .b. The TS Bases Section 3/4.7.11 is also modified to reflect the above changes.

A license condition was also included in Appendix B of the Operating license, which is a list of additional license conditions. This license condition was discussed with NNECO in a conference call on December 15, 1997, and NNECO agreed to the inclusion of the license condition for approving the amendment.

Date of issuance: February 9, 1998.

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

Amendment No.: 213.

Facility Operating License No. DPR–65: Amendment revised the Technical Specifications and Appendix B of Operating License.

Date of initial notice in **Federal Register**: April 23, 1997 (62 FR 19831).

The September 25, 1997, letter provided clarifying information that did not change the scope of the March 27, 1997, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 9, 1998.

No significant hazards consideration comments received: No.

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.

Northeast Nuclear Energy Company (NNECO), et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: August 29, 1997, as supplemented by letters dated September 25 and November 14, 1997.

Brief description of amendment: Based on a review and subsequent calculations of the cold overpressurization protection (COPS) enabling temperature and the emergency core cooling system (ECCS)/ charging system mode 3 requirements, NNECO proposes to reduce the COPS enabling temperature. As a result, NNECO proposed the following Technical Specifications (TS) changes: add new heatup and cooldown pressure/temperature limit curves and their associated requirements; add new power operated relief valve (PORV) setpoint curves and their associated requirements; revise the reactor coolant loops and coolant circulation, ECCS, boration systems, and COPS to incorporate the lower enabling temperature and new restrictions for cold overpressure protection system, PORV undershoot, and residual heat removal (RHR) relief valve bellows; add a footnote to allow a reactor coolant pump to substitute for an RHR pump during heatup from Mode 5 to 4, which is consistent with the improved standard technical specification (STS); reword TS 3/4.4.9.3 and its surveillance requirement to be consistent with the improved STS; and revise the affected Bases sections to be consistent with the proposed changes.

Date of issuance: February 12, 1998.

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

Amendment No.: 157.

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

Date of initial notice in **Federal Register***:* December 8, 1997 (62 FR 52583).

The September 25 and November 14, 1997, letters provided clarifying information that did not change the August 27, 1997, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 12, 1998.

No significant hazards consideration comments received: No.

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.

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: November 20, 1996, as supplemented by letter dated February 20, 1997, and submittal dated March 25, 1997.

Brief description of amendment: The amendment revised the technical specifications to reflect organizational changes and correct editorial and typographical inaccuracies. It also removed paragraph 3.D of the facility operating license that described the modification that increased the spent fuel pool storage capacity.

Date of issuance: February 3, 1998.

Effective date: February 3, 1998.

Amendment No.: 184.

Facility Operating License No. DPR–40: Amendment revised the Technical Specifications and Facility Operating License No. DPR–40.

Date of initial notice in **Federal Register***:* January 2, 1997 (62 FR 131) and April 9, 1997 (62 FR 17238). The March 25, 1997, submittal did not change the staff's original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 3, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102.

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

Date of application for amendments: October 4, 1995, as supplemented by letters dated July 17, 1996, August 20, 1996, and June 2, 1997.

Brief description of amendments: The amendments revise the technical specifications to relocate the requirements in 10 subsections of the technical specifications to licensee-controlled documents.

Date of issuance: February 3, 1998.

Effective date: February 3, 1998, to be implemented within 90 days of issuance.

Amendment Nos.: Unit 1—120; Unit 2—118.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Operating Licenses and the Technical Specifications.

Date of initial notice in **Federal Register***:* November 27, 1995 (60 FR 58404). The July 17, 1996, August 20, 1996, and June 2, 1997, supplemental letters provided additional clarifying information and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 3, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

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

Date of application for amendments: May 14, 1997, as supplemented by letter dated December 15, 1997.

Brief description of amendments: The amendments revised the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2 to revise Technical Specification (TS) 6.9.1.8.b.5 to replace reference WCAP–10266–P–A with WCAP–12945–P for best estimate loss-of-coolant accident (LOCA) analysis. The amendment also revises TS Bases 3/ 4.2.2 and 3/4.2.3 to change the emergency core cooling system (ECCS) acceptance criteria limit to state that there is a high level of probability that the ECCS acceptance criteria limits are not exceeded.

Date of issuance: February 13, 1998.

Effective date: February 13, 1998, to be implemented within 90 days of issuance.

Amendment Nos.: Unit 1—121; Unit 2—119.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 30, 1997 (62 FR 40855).

The December 15, 1997, supplemental letter provided additional clarifying information and did not change the staff's initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 13, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407

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

Date of application for amendments: December 9, 1996.

Brief description of amendments: The amendments revised the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP), Unit Nos. 1 and 2 to revise the surveillance frequencies from at least once every 18 months to at least once per refueling interval (nominally 24 months) for the reactor trip system (RTS) and engineering safety features actuation systems (ESFAS) instrumentation channels, and make certain changes in trip setpoints and allowance values due to a setpoint methodology change in support of the calibration extensions. Channel operational tests (COTs) and trip actuating device operational tests (TADOTs) associated with these channels are also being extended. Revisions to the appropriate TS Bases are being revised to support the TS revisions.

Date of issuance: February 17, 1998.

Effective date: February 17, 1998, to be implemented within 90 days of issuance.

Amendment Nos.: Unit 1—Amendment No. 122; Unit 2—Amendment No. 120.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 12, 1997 (62 FR 6577)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 17, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

Pennsylvania Power and Light Company, Docket Nos. 50–387 and 50–388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: May 31, 1996.

Brief description of amendments: These amendments delete, from the Technical Specifications, Section 4.7.2.d.2, the surveillance requirement for chlorine detection for the control room emergency outside air supply system as a result of the removal of bulk quantities of gaseous chlorine from the Susquehanna Steam Electric Station.

Date of issuance: February 19, 1998.

Effective date: Both units, as of date of issuance, to be implemented within 30 days.

Amendment Nos.: 172 and 145.

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

Date of initial notice in **Federal Register**: July 16, 1997 (62 FR 38137).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 19, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701.

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: August 8, 1997, as supplemented October 10, 1997, January 16, 23, and 27, 1998.

Brief description of amendments: The amendment changes Vogtle Electric Generating Plant, Units 1 and 2, Technical Specifications (TS) 3.7.17, ''Fuel Storage Pool Boron Concentration,'' TS 3.7.18, ''Fuel Assembly Storage in the Fuel Storage Pool,'' and TS 4.3, ''Fuel Storage,'' to allow credit for soluble boron, in the spent fuel pool, for maintenance of subcriticality associated with spent fuel storage.

Date of issuance: February 20, 1998.

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

Amendment Nos.: 99—Unit 1; 77—Unit 2

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

Date of initial notice in **Federal Register**: December 31, 1997 (62 FR 68136).

The January 16, 23, and 27, 1998, letters provided clarifying information that did not change the scope of the August 8, 1997, 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 February 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. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: September 17, 1997 (TS 97–02).

Brief description of amendments: The amendments change the Technical Specifications (TS) by modifying Surveillance Requirements (SRs) 4.6.2.1.1.b., 4.6.2.1.1.c,. 4.6.2.1.1.d, and 4.6.2.1.2.b to account for a plant modification to the containment spray system and to make the SRs more consistent with the Westinghouse Standard TS (NUREG–1431).

Date of issuance: February 20, 1998.

Effective date: February 20, 1998.

Amendment Nos.: 231 and 221.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise TS.

Date of initial notice in **Federal Register**: October 8, 1997 (62 FR 52589).

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: February 23, 1996, as supplemented by letters dated April 24, 1996, and November 15, 1996.

Brief description of amendment: The amendment revises the Callaway Plant, Unit 1 operating license to reflect Union Electric Company (UEC) as a wholly-owned operating subsidiary of Ameren Corporation at the closing of the contemplated merger between UEC and CIPSCO Incorporated.

Date of issuance: February 13, 1998.

Effective date: February 13, 1998.

Amendment No.: 120.

Facility Operating License No. NPF–30: The amendment revised the Operating License.

Date of initial notice in **Federal Register**: May 22, 1996 (61 FR 25713) The November 15, 1996, supplemental letter provided only clarifying information and did not change the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 13, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Missouri-Columbia, Elmer Ellis Library, Columbia, Missouri.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri.

Date of application for amendment: August 8, 1997.

Brief description of amendment: The amendment revises the Callaway Plant, Unit 1 surveillance requirements of Technical Specification 3/4.7.4, ''Essential Service Water System'' by removing the requirement to perform 4.7.4.b, 4.7.4.b.2 and 4.7.4.c during shutdown.

Date of issuance: February 24, 1998.

Effective date: February 24, 1998, to be implemented within 30 days from the date of issuance.

Amendment No.: 121.

Facility Operating License No. NPF–30: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 17, 1997 (62 FR 66143) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 24, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Missouri-Columbia, Elmer Ellis Library, Columbia, Missouri 65201–5149.

Dated at Rockville, Maryland, this 4th day of March 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–6085 Filed 3–10–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR WASTE TECHNICAL REVIEW BOARD

Panel Meeting: April 23–24, 1998—Albuquerque, New Mexico: The Department of Energy's Work on the Total System Performance Assessment for the Viability Assessment (TSPA–VA)

Pursuant to its authority under section 5051 of Public Law 100–203, the Nuclear Waste Policy Amendments Act of 1987, the Nuclear Waste Technical Review Board's Panel on Performance Assessment will hold a meeting April 23–24, 1998, beginning at 8:30 a.m. both days. The meeting, which is open to the public, will focus on the Department of Energy's work on the total system performance assessment for the viability assessment, or TSPA–VA. A detailed agenda will be available approximately two weeks prior to the meeting by fax or e-mail, or on the Board's web site at www.nwtrb.gov.

The meeting will be held at the Sheraton Uptown Albuquerque Hotel, 2600 Louisiana Boulevard, NE, Albuquerque, New Mexico 87110; Toll-free (800) 252–7772; Tel (505) 881–0000; Fax (505) 881–3736. Reservations for accommodations must be made by March 23, 1998, and you must indicate that you are attending the Nuclear Waste Technical Review Board's panel meeting to receive the preferred rate.

Time will be set aside on the agenda for comments and questions from the public. Those wishing to speak are encouraged to sign the Public Comment Register at the check-in table. A time limit may have to be set on the length of individual remarks; however, written comments of any length may be submitted for the record.

Transcripts of this meeting will be available on computer disk, via e-mail, or on a library-loan basis in paper format from Davonya Barnes, Board staff, beginning May 22, 1998. For further information, contact Frank Randall, External Affairs, 2300 Clarendon Blvd., Suite 1300, Arlington, Virginia 22201–3367; (Tel) 703–235–4473; (Fax) 703–235–4495; (E-mail) info@nwtrb.gov.

The Nuclear Waste Technical Review Board was created by Congress in the Nuclear Waste Policy Amendments Act of 1987 to evaluate the technical and scientific validity of activities undertaken by the DOE in its program to manage the disposal of the nation's high-level radioactive waste and commercial spent nuclear fuel. In that same legislation, Congress directed the DOE to characterize a site at Yucca Mountain, Nevada, for its suitability as a potential location for a permanent repository for the disposal of that waste.

Dated: March 6, 1998.

William Barnard,

Executive Director, Nuclear Waste Technical Review Board.

[FR Doc. 98–6209 Filed 3–10–98; 8:45 am]

BILLING CODE 6820–AM–M

SECURITIES AND EXCHANGE COMMISSION

[Investment Company Act Release No. 23058; 812–11016]

AMP Limited, et al.; Notice of Application

March 4, 1998.

AGENCY: Securities and Exchange Commission (''SEC'').

ACTION: Notice of application for exemption under section 6(c) of the Investment Company Act of 1940 (the ''Act'') from section 15(a) of the Act.

SUMMARY OF APPLICATION: Applicants seek an order to permit the implementation, without prior shareholder approval, of new sub-advisory agreements (''New Agreements'') between Seligman Henderson Co. (''Sub-adviser'') and J.&W. Seligman & Co. Incorporated (''Seligman'') relating to various registered investment companies (each a ''Fund'' and collectively, the ''Funds'') in connection with the acquisition of Henderson plc (''Henderson'') by AMP Limited (''AMP''). The order would cover a period of up to 150 days following the later of: (i) the date on which the assignment of the existing investment sub-advisory agreements (''Existing Agreements'') is deemed to have occurred (*i.e.*, the date AMP is deemed to control the issued share capital of Henderson (the ''Assignment Date'')), or (ii) the date upon which the requested order is issued (but in no event later than October 1, 1998) (''Interim Period''). The order also would permit the Sub-adviser to receive all fees earned under the New Agreements during the Interim Period following shareholder approval.

APPLICANTS: AMP, Henderson, and the Sub-adviser.

FILING DATES: The application was filed on February 18, 1998, and was amended and restated on March 3, 1998.

HEARING OR NOTIFICATION OF HEARING: An order granting the application will be issued unless the SEC orders a hearing. Interested persons may request a hearing by writing to the SEC's Secretary and serving applicants with a