The NRC must be able to rely on the Licensee and its employees to comply with NRC requirements, including the requirement to provide and maintain information that is complete and accurate in all material respects. Mr. LaRocque's action in causing the Licensee to violate its License and the Commission's regulations, his misrepresentations to the Licensee, and his prior actions as set forth in Section II of this Order, have raised serious doubt as to whether he can be relied upon to comply with NRC requirements, and to provide complete and accurate information to the NRC and its Licensees.

Consequently, I lack the requisite reasonable assurance that licensed activities can be conducted in compliance with the Commission's requirements and that the health and safety of the public would be protected if Mr. LaRocque were permitted at this time to be involved in NRC-licensed activities. Therefore, the public health, safety and interest require that Mr. LaRocque be prohibited from any involvement in NRC-licensed activities for a period of one year from the effective date of this Order. If Mr. LaRocque is involved in NRC-licensed activities on the effective date of the Order, Mr. LaRocque must immediately cease such activities, and inform the NRC of the name, address, and telephone number of the employer, and provide a copy of this Order to the employer. Additionally, Mr. LaRocque is required to notify the NRC of his first employment in NRC-licensed activities following the prohibition period.

IV

Accordingly, pursuant to Sections 81, 161b, 161i, 182 and 186 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 C.F.R. 2.202, 10 C.F.R. 30.10, and 10 C.F.R. 150.20, *it is hereby ordered* That:

1. Mr. Lee LaRocque is prohibited for one year from the effective date of this Order from engaging in NRC-licensed activities. NRC-licensed activities are those activities that are conducted pursuant to a specific or general license issued by the NRC, including, but not limited to, those activities of Agreement State licensees conducted pursuant to the authority granted by 10 C.F.R. 150.20.

2. If, on the effective date of this Order, Mr. LaRocque is involved in NRC-licensed activities, he must immediately cease those activities, and inform the NRC of the name, address, and telephone number of the employer, and provide a copy of this Order to the employer.

3. For a period of one year after the one-year period of prohibition has expired, Mr. LaRocque shall, within 20 days of his acceptance of each employment offer involving NRC-licensed activities or his becoming involved in NRC-licensed activities, as defined in Paragraph IV.1 above, provide notice to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, of the name, address, and telephone number of the employer or the entity where he is, or will be, involved in the NRC-licensed activities. In the first notification, Mr. LaRocque shall include a statement of his commitment to compliance with regulatory requirements and the basis why the Commission should have confidence that he will now comply with applicable NRC requirements.

The Director, Office of Enforcement, may, in writing, relax or rescind any of the above conditions upon demonstration by Mr. LaRocque of good cause.

V

In accordance with 10 C.F.R. 2.202, Mr. LaRocque must, and any other person adversely affected by this Order may, submit an answer to this Order, and may request a hearing on this Order, within 20 days of the date of this Order. Where good cause is shown, consideration will be given to extending the time to request a hearing. A request for extension of time must be made in writing to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission Washington, D.C. 20555, and include a statement of good cause for the extension. The answer may consent to this Order. Unless the answer consents to this Order, the answer shall, in writing and under oath or affirmation, specifically admit or deny each allegation or charge made in this Order and shall set forth the matters of fact and law on which Mr. LaRocque or other person adversely affected relies and the reasons as to why the Order should not have been issued. Any answer or request for a hearing shall be submitted to the Secretary, U.S. Nuclear Regulatory Commission, Attn: Chief, Rulemakings and Adjudications Staff, Washington, DC 20555. Copies also shall be sent to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555, to the Deputy Assistant General Counsel for Enforcement at the same address, to the Regional Administrator, NRC Region I, U.S. Nuclear Regulatory Commission, 475 Allendale Road, King of Prussia, Pennsylvania 19406, and to Mr. LaRocque if the answer or hearing request is by a person other than Mr. LaRocque. If a person other than Mr. LaRocque requests a hearing, that person shall set forth with particularity the manner in which that person's interest is adversely affected by this Order and shall address the criteria set forth in 10 C.F.R. 2.714(d).

If a hearing is requested by Mr. LaRocque or a person whose interest is adversely affected, the Commission will issue an Order designating the time and place of any hearing. If a hearing is held, the issue to be considered at such hearing shall be whether this Order should be sustained.

In the absence of any request for hearing, or written approval of an extension of time in which to request a hearing, the provisions specified in Section IV above shall be final 20 days from the date of this Order without further order or proceedings. If an extension of time for requesting a hearing has been approved, the provisions specified in Section IV shall be final when the extension expires if a hearing request has not been received.

Dated at Rockville, Maryland this 24th day of February 1999.

For the Nuclear Regulatory Commission.

Malcolm R. Knapp,

Deputy Executive Director for Regulatory Effectiveness.

[FR Doc. 99–5871 Filed 3–9–99; 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 12, 1999, through February 26, 1999. The last biweekly notice was published on February 24, 1999 (FR 64 PR 9183).

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 9, 1999, 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. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of amendment request: January 29, 1999.

Description of amendment request: The amendments would allow credit for containment overpressure to assist in providing net positive suction head (NPSH) for the emergency core cooling system pumps for a period of greater than 8 hours. The current licensing basis recognizes credit given only to 8 hours after a design-basis loss-of-coolant accident and the licensee has determined this to be an unreviewed safety question.

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

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

The proposed amendment involves the available containment overpressure (COP) following a design basis loss of coolant accident (DBA–LOCA) and the resulting NPSH available to the RHR [residual heat removal] and CS [core spray] pumps. While this change affects the ability of these pumps to perform their required functions following a DBA–LOCA, it does not affect the reactor recirculation piping or the reactor coolant pressure boundary, which are the initiators of the DBA–LOCA. Therefore, the proposed amendment does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the set points at which these actions are initiated. The proposed change permits limited COP to be credited in the calculation of available NPSH for the RHR and CS pumps following a DBA–LOCA.

The proposed change is supported by calculations, which demonstrates that adequate COP will be available to ensure the RHR and CS systems will be capable of performing their required safety functions. Therefore, the proposed amendment does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed amendment permits limited COP to be credited in the calculation of available NPSH for the RHR and CS pumps following a DBA–LOCA. This amendment does not involve a physical alteration of the plant. The proposed amendment is supported by calculations, which demonstrate that adequate COP will be available to ensure the RHR and CS systems will be capable of performing their required safety functions. This amendment will not alter the manner in which the RHR and CS systems are initiated, nor will the function demands on the RHR or CS system be changed. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed amendment permits limited COP to be credited in the calculation of available NPSH for the RHR and CS pumps following a DBA–LOCA. Crediting an incremental amount of overpressure does not result in a significant reduction in the margin of safety, because conservative analyses demonstrate that adequate COP will be available to ensure the RHR and CS systems will be capable of performing their required safety functions. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: February 18, 1999.

Description of amendment request: The proposed amendments would revise the joint Technical Specifications (TSs): (1) Surveillance Requirement (SR) 3.6.16.1—This SR incorrectly characterizes the access openings (there are five of them) to the reactor building as each having a double-door design, when in reality there is a single door for each opening; the proposed revision would change the wording to correctly characterize the actual design. (2) SR 3.6.16.3—This SR specifies that the reactor building structural integrity inspection be performed every 40 months to 50 months and during shutdown; the proposed revision would change this frequency to three times every 10 years coinciding with containment visual examinations required by SR 3.6.1.1. (3) Administrative Control 5.5.2—The proposed revision would add wording to specify that containment visual examinations required by Regulatory Guide c.3 will be conducted three times every 10 years including during each shutdown for SR 3.6.1.1.

The proposed amendments would only revise the SRs and Administrative Controls specified above; no physical change to any plant design is involved.

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:

First Standard

Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. Approval of this amendment will have no significant effect on accident probabilities or consequences. The containment and reactor building are not accident initiating systems or structures; therefore, there will be no impact on any accident probabilities by the approval of this amendment. The containment and reactor buildings serve an important function to mitigate consequences of postulated accidents previously evaluated and the examination frequencies proposed in this amendment will not result in a reduction in their capacity to meet their intended function. Therefore, there will be no impact on the consequences of any accident previously evaluated.

Second Standard

Implementation of this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No changes are being made to the plant that will introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators, since the containment and reactor building function primarily as accident mitigators.

Third Standard

Implementation of this amendment would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation, including the performance of the containment and reactor building. These components are already capable of performing as designed, and their functions are verified by visual examination and leakage rate testing. The ability of the containment and reactor building to perform their design function will not be impaired by the implementation of this amendment at Catawba Nuclear Station. Consequently, no safety margin will be impacted.

The NRC staff 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: York County Library, 138 East Black Street, Rock Hill, South Carolina

Attorney for licensee: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Project Director: Herbert N. Berkow.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Nuclear Generating Plant, Unit No. 3 (CR–3), Citrus County, Florida

Date of amendment request: January 27, 1999.

Description of amendment request: The proposed amendment would provide a one-time extension of the inspection interval for the Once Through Steam Generator (OTSG) tubes specified in the Crystal River Unit 3 (CR–3) Improved Technical Specifications (ITS) to coincide with the planned operating cycle. CR–3 ITS 5.6.2.10 requires the OTSG inspection interval to be 24 calendar months for Category C–2 inspection results. However, due to a previous extended maintenance outage, the next OTSG inspection at CR–3, which is planned for the October 1999 refueling and maintenance outage, will be approximately 26 calendar months since the last inspection. Florida Power Corporation indicated that the total interval between inspections would correspond to less than 21.6 months of plant operation at a temperature of 500°F or above (measured at the hot leg side of the OTSG). The licensee stated that the conclusions reached in the operational assessments for the OTSGs show leakage and structural integrity are maintained by substantial margins until the end of the planned operating cycle.

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

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

The last Crystal River Unit 3 (CR–3) Once Through Steam Generator (OTSG) tube surveillance was completed in August 1997. Both standard and enhanced eddy current techniques were used to inspect 100% of the OTSG tubes. Operational assessments performed for CR–3 provide reasonable assurance that the OTSG performance criteria meet the leakage and structural requirements in Draft Regulatory Guide-1074. These performance criteria will be maintained until the end of the planned operating cycle. These operational assessments demonstrate that operation is acceptable for an operating cycle length of up to 21.6 months of operating time at a temperature of 500°F or above (measured at the hot leg side).

The operational assessments concluded that the projected cumulative leakage for the limiting OTSG would be less than 1 gallon per minute (gpm) under the limiting accident conditions at the end of the planned operating cycle. Thus, the accident analysis assumptions bound the condition of the OTSGs, and structural and leakage integrity will be maintained for the proposed operating cycle. Therefore, the proposed one-time change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Create the possibility of a new or different kind of accident from previously evaluated accidents?

No new failure modes or accident scenarios are created by changing the inspection from a frequency based on calendar months, to a one-time interval based on up to 21.6 months of operating time at a temperature of 500'F or above (measured at the hot leg side). Plant systems and components will not be operated in a different manner as a result of this change. Thus, this change does not increase the risk of a plant trip or present a challenge to any other safety system. For all known degradation mechanisms in the CR–3 OTSGs, the most recent operational assessments bound the probability of tube burst and project primary-to-secondary leakage at accident conditions for the end of Operating Cycle 11 to be less than 1 gpm. Therefore, the proposed one-time change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Improved Technical Specification (ITS) Bases 3.4.12 contains relevant information pertaining to the limitations on reactor coolant system (RCS) leakage. The ITS Bases discuss the 1 gpm primary-to-secondary leakage assumed for a main steam line break accident, as well as for a steam generator tube rupture accident. The evaluation provided by this license amendment request shows that tube structural integrity is maintained, thus the required structural margins specified in NRC Regulatory Guide 1.121 are satisfied. The operational assessments performed show the maximum accident leakage, assuming all these indications leak, is less than 1 gpm. Therefore, all known OTSG tube degradation mechanisms have been assessed, and the proposed one-time change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428

Attorney for licensee: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC—A5A, P. O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: January 18, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Table 3.7–6, ''Area Temperature Monitoring,'' by increasing the temperature limits for the fuel building fuel pool pump cubicles and fuel building general area. The amendment would also change the Millstone Unit 3 licensing basis by incorporating into the Millstone Unit 3 Final Safety Analysis Report (FSAR) a revision to describe the full core off-load condition as a normal evolution. In addition, the amendment would increase the maximum bulk spent fuel pool (SFP) temperature from 140° F to 150° F, allow the crediting of evaporative cooling as a decay heat removal mechanism for the SFP (use of the ONEPOOL computer code), and allow the use of Holtec's quality assurance validated DECOR computer code as a method for predicting decay heat loads in the SFP pool.

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:

In accordance with 10 CFR 50.92, NNECo has reviewed the proposed changes and has concluded that the changes do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not [satisfied]. The proposed changes do not involve an SHC because the changes would not:

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

The proposed license amendment will permit NNECo to conduct full core off-loads as a normal evolution through the end of plant life. This amendment request does not affect: (1) the number of spent fuel assemblies allowed in the spent fuel pool, (2) Spent Fuel Pool (SFP) criticality analysis, (3) structural analysis of the spent fuel pool or (4) radiological release scenarios.

The proposed license amendment permits the use of ORIGEN2 based DECOR and ONEPOOL codes for the analysis of the Unit 3 SFP. The ORIGEN2 based DECOR code more accurately predicts decay heat loads from the spent fuel in the SFP. The ONEPOOL code credits the effect of evaporative cooling on the SFP bulk temperature. The use of these codes improves the accuracy of predicting SFP bulk temperatures during normal and abnormal refueling scenarios.

The analysis of decay heat removal permits the discharge of fuel from the reactor vessel to the SFP [to] start as early as 132 hours (depending on cooling water temperature) after reactor shutdown at a rate of 3 assemblies per hour. The existing accident analysis for a dropped spent fuel bundle during refueling bounds this situation as the analysis assumed a decay time of 100 hours after reactor shutdown.

The increase in pool temperature from 140° F to 150° F does not significantly impact the structural integrity of the fuel handling equipment. The temperature increase does not create a new failure of the fuel handling equipment that has not been previously analyzed.

The increased SFP temperature results in higher ambient temperatures in the Fuel Building. However, the duration of an increased pool temperature event is limited. The effect on the environmental qualification (EQ) of electrical equipment is an increase in the Maximum Normal and Abnormal Excursion temperatures, which are based on short duration excursions from the predicted summer maximum temperatures. This is reflected in the proposed Technical Specification (TS) temperature changes. The temperature limits within TS, 3.7.14, ''Plant Systems: Area Temperature Monitoring,'' Table 3.7–6, for the Fuel Pool Pump Cubicles and Fuel Pool General Area increase from 110° F to 119° F, and from 104° F to 108° F respectively, based upon the revised environmental conditions. The proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously analyzed as the Fuel Building Ventilation System is qualified for the increased temperature and humidity conditions. There are no changes in the EQ of equipment.

A comprehensive review of the design of the SFP, Spent Fuel Pool Cooling and Purification system and other associated systems, structures and components has been completed. All systems, structures and components are fully qualified at the higher SFP temperature of 150° F for a full core off-load as a normal operation.

Therefore, based on the above, this change will not involve a significant increase in the probability or consequence of an accident previously analyzed.

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

The proposed license amendment will permit NNECo to conduct full core off-loads as a normal evolution through the end of plant life. There are no physical plant changes. The SSCs [systems, structures, and components] supporting the SFP and Spent Fuel Pool Cooling are fully qualified for operation at 150° F. The higher Fuel Pool Pump Cubicles and Fuel Pool General Area temperatures do not create the possibility of a new or different kind of accident from any previously evaluated. Thus the changes do not create the possibility of an accident of a different type than previously evaluated.

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

The proposed license amendment will permit NNECo to conduct full core off-loads as a normal evolution through the end of plant life. The proposed changes allow a higher heat load in the SFP which results in a higher calculated maximum temperatures than the current analysis. In addition, several changes have been made with respect to the analysis methods used in calculating the maximum temperatures.

The new analysis demonstrates that the SFP cooling configuration will maintain the SFP pool bulk temperature at or below 150° F with a single train of spent fuel pool cooling. This temperature is above the SRP [Standard Review Plan] guidance of 140° F but is well below the 212° F limit permitted for abnormal core off-loads as defined in the Standard Review Plan (NUREG -0800). This temperature guideline of 140° F was one of the acceptance criteria credited by the NRC staff during their review of the adequacy of the design of the SFP Cooling System within the NRC Safety Evaluation Report (SER) for Millstone Unit 3 (NUREG–1031) and consequently requires prior review and approval.

A single active failure will cause the loss of one of the two trains of spent fuel pool cooling. The complete loss of cooling to the Spent Fuel Pool is not a creditable occurrence in that the Fuel Pool Cooling System is designed to be able to withstand the worst single failure and still be able to perform its intended function. However, a loss of cooling analysis indicates that several hours are available during a refueling, and over thirteen hours are available during normal operations for operators to respond to the loss of cooling prior to the Spent Fuel Pool reaching its structural design temperature of 200° F.

A comprehensive review of the design of the SFP, Spent Fuel Pool Cooling and Purification System and other associated systems, structures and components has been completed for qualification at the higher pool temperature of 150° F. All systems, structures and components are fully qualified at the higher Technical Specification Fuel Pool Pump Cubicles and Fuel Pool General Area temperatures, and at the increased SFP temperature, and are therefore qualified for a full core off-load as a normal operation.

The ORIGEN2 based DECOR code more accurately predicts decay heat loads from the spent fuel in the SFP. The ONEPOOL code credits the effect of evaporative cooling on the SFP bulk temperature. The use of these codes improves the accuracy of predicting SFP bulk temperatures during normal and abnormal refueling scenarios. The use of these computer codes as a method for predicting decay heat loads and crediting evaporative cooling as a decay heat removal mechanism have not previously been evaluated for Unit 3, and therefore, require[s] prior NRC review and approval.

Therefore, based on the above, this license amendment to permit NNECo to conduct full core off-loads as a normal evolution, increase the maximum SFP pool bulk temperature from 140° F to 150° F, use the ORIGEN2 based DECOR and ONEPOOL computer codes to calculate the decay heat load and determine the effects of evaporative cooling respectively, and increase the TS Fuel Pool Pump Cubicles and General Area temperatures, does not involve a significant reduction in the margin of safety.

Thus, it is concluded that the proposed amendment does not involve a significant reduction in the margin of safety.

In conclusion, based on the information provided, it is determined that the proposed amendment does not involve an SHC.

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: February 10, 1999.

Description of amendment request: The proposed amendment would incorporate alternative inspection requirements into Technical Specification Surveillance Requirement 3/4.4.10, ''Structural Integrity,'' for the reactor coolant pump flywheel.

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:

NNECO has reviewed the proposed revision in accordance with 10 CFR 50.92 and has concluded that the revision does not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not satisfied. The proposed revision does not involve a SHC because the revision would not:

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

This proposed revision to the Millstone Unit No. 3 Technical Specifications incorporates alternative reactor coolant pump flywheel inspection requirements into Surveillance 4.4.10 based on Topical Report WCAP–14535A. WCAP–14535A provided a technical basis for the elimination of inspection requirements for reactor coolant pump flywheels based on industry data. The industry data indicated that no indications that would affect the integrity of flywheels was [sic] revealed during 729 examinations of 217 flywheels at 57 plants (including Millstone Unit No. 3). The NRC, during their review and approval of the WCAP required continued inspections on a ten year interval to protect against events and degradation that were not anticipated and had not been considered in the WCAP analysis. The proposed alternate inspection requirements are consistent with the conclusions of an NRC review and generic approval of Topical Report WCAP–14535A. Thus, it is concluded that the proposed revision does not significantly increase the probability of an accident.

Additionally, the performance of reactor coolant pump flywheel surveillances does not increase the consequence of an accident previously evaluated.

Therefore, it is concluded that the proposed revision does not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

This proposed revision to the surveillance does not change the operation of any plant system or component during normal or accident conditions. The proposed change incorporates alternate inspection requirements for the reactor coolant pump flywheels that were generically approved for use by licensees by the NRC. This change does not include any physical changes to the plant.

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

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

This proposed revision to the Millstone Unit No. 3 Technical Specifications incorporates alternative reactor coolant pump flywheel inspection requirements into Surveillance 4.4.10 that are consistent with the conclusions of an NRC review and generic approval of Topical Report WCAP–14535A. The current inspection requirements of Surveillance 4.4.10 and the NRC review of WCAP–14535A were both based on the recommendations of Regulatory Guide 1.14.

Thus, it is concluded that the proposed revision does not involve a significant reduction in a margin of safety.

In conclusion, based on the information provided, it is determined that the proposed revision does not involve an SHC.

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

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

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

NRC Project Director: Elinor G. Adensam.

Northern States Power Company, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of amendment requests: February 5, 1999.

Description of amendment requests: The proposed amendments would modify the technical specifications (TS) to incorporate Revision 3 of the ABB Combustion Engineering, Inc.'s topical report, CEN–629–P, ''Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leaktight Sleeves'', dated September 1998 (proprietary and nonproprietary documents available). The current TS requires that steam generator tube repair using the Combustion Engineering Inc.'s welded sleeves shall be in accordance with the methods and criteria described in Revision 2 of CEN–629–P and Addendum 1, Revision 1 of CEN–629–P. Incorporation of Revision 3 of CEN–629–P would involve the following TS changes: (1) editorial/administrative change to TS.4.12.D.3 to reflect adoption of Revision 3 of CEN–629–P, and deletion of reference to Addendum 1, Revision 1 of CEN–629–P since Revision 3 incorporates Addendum 1, Revision 1 of CEN–629–P; (2) changes in sleeve installation practices that incorporate improvements gained by prior experiences; and (3) more restrictive change to the sleeve repair limit as specified in TS.4.12.D.1.(f) from 31 percent of the nominal sleeve wall thickness to 25 percent.

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 amendment[s] will not involve a significant increase in the probability or consequences of an accident evaluated.

Editorial changes have no effect on probability or consequences of accidents previously evaluated. Changes in installation practices incorporate improvements gained by experience in installing sleeves. Further, the changes in the installation practices will change neither the final configuration of installed sleeves nor the post-installation NDE [nondestructive examination] from that which is already approved. Accident induced steam generator tube leakage is not [a]ffected by these changes. Post installation non-destructive examination will be conducted using VT, UT, and ET as previously licensed. The changes in repair limits have [led] to repair limits that are more conservative than those which have been previously approved. Thus, none of these changes will create the possibility of a new or different kind of accident from any accident previously analyzed.

2. The proposed amendment[s] will not create the possibility of a new or different kind of accident previously analyzed.

Editorial changes cannot create the possibility of a new or different kind of accident. Changes in installation practices incorporate improvements gained by experience in installing sleeves. Further, changes in installation practices do not change the final configuration of installed sleeves from that which is already approved. The changes in repair limits have [led] to repair limits that are more conservative than those which have been previously approved.

Thus, none of these changes will create the possibility of a new or different kind of accident from any accident previously analyzed.

3. The proposed amendment[s] will not involve a significant reduction in the margin of safety.

Editorial changes have no effect on the margin of safety. Changes in installation practices incorporate improvements gained by experience in installing sleeves. Further, changes in installation practices do not change the final configuration of installed sleeves from that which is already approved. The changes in repair limits have [led] to repair limits that are more conservative than those which have been previously approved. None of these changes will affect the tube plugging assumptions used in the PINGP [Prairie Island Nuclear Generating Plant] accident analyses. Thus, none of these changes will 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 10 CFR 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: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

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

NRC Project Director: Cynthia A. Carpenter.

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: January 15, 1999.

Description of amendment request: The proposed changes revise calibration requirements for the local power range monitors (LPRM).

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:

Operation of the FitzPatrick plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

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

This change proposes to remove the listed requirement for the method of calibration of the LPRM Signal from TS Table 4.1–2 because the definition for Instrument Channel Calibration provides the necessary guidance.

Other changes to the bases and adopting signal calibration frequency units of MWD/T [Megawatt Days per Ton] vice effective full power hours is consistent with STS [Standard Technical Specification].

The proposed changes do not increase the probability of an accident because the proposed surveillance requirements still ensure that the LPRM signal is adequately calibrated. The proposed change provides assurance that the associated Reactor Protection System (RPS) functions are tested consistent with the analysis assumptions. As a result, the consequences of an accident are not affected by this change. This change will not alter assumptions relative to the mitigation of an accident or transient event. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes will not physically alter the plant. As such, no new or different types of equipment will be installed. The methods governing normal plant operation and testing are consistent with current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change removes specific calibration method information in Table 4.1–2 regarding the LPRM signal which is adequately addressed in the definition for Instrument Channel Calibration.

Other changes to the Bases and adopting a signal calibration Frequency units of MWD/T vice effective full power hours is consistent with STS.

The proposed changes still provide the necessary control of testing to ensure operability of the RPS instrumentation. The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Project Director: S. Singh Bajwa, Director.

Public Service Electric & Gas Company, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: February 2, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 5.6, ''Fuel Storage, Criticality,'' to change the maximum unirradiated fuel assembly enrichment value for new fuel storage from 4.5 to 5.0 weight percent Uranium-235 and to allow the use of equivalent criticality control to that provided by the current TS requirement of 2.35 mg of Boron-10 per linear inch loading in the Integral Fuel Burnable Absorber (IFBA) pins.

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. Will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(a) Fuel Assembly Drop.

There is no increase in the probability of a fuel assembly drop accident because the mass of a fuel assembly does not increase when the fuel enrichment is increased. This amendment affects only the isotopic composition within the fuel pellets of a fuel assembly without involving any changes to the outward physical characteristics or structural integrity of the assembly.

The radiological consequences of a new fuel assembly drop accident do not increase as a consequence of the proposed change to new fuel enrichment. Because it has not been irradiated, there are no significant radiological consequences associated with fresh fuel. The radiological consequences of an irradiated fuel assembly drop were previously evaluated and approved in the Spent Fuel license amendment numbers 151/131 (Units 1 & 2 respectively).

(b) Misplaced Fuel Assembly in New Fuel Storage Vault or Spent Fuel Storage Racks.

There is no increase in the probability of a misplaced fuel assembly in the New Fuel Storage Vault or Spent Fuel Storage Racks. The proposed change does not alter the physical structure of the New Fuel Storage Vault or the Spent Fuel Storage Racks. All new fuel assembly movements will continue to be made in accordance with approved procedures.

There is no increase in the consequences of misplacing a fuel assembly in the new fuel storage racks. The normally-dry new fuel vault K_{eff} is very small (approximately 0.65), as such, there is sufficient reactivity margin to the 0.95 limit to bound any possible misplacement. The double contingency principle does not require consideration of a second unlikely event. Since a misplaced bundle constitutes the first unlikely event, presence of moderator in the normally dry new fuel storage racks (a second unlikely event) is not assumed in evaluating the event.

The inadvertent misplacement of a fresh fuel assembly in the spent fuel storage racks has the potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble boron. Administrative procedures to assure the presence of soluble boron during fuel handling operations will preclude the possibility of the simultaneous occurrence of the two independent accident conditions. The analyses supporting Amendments 151/131 demonstrated that 600 ppm of soluble boron is adequate to compensate for a mis-loaded fuel event, while plant procedures require the concentration to be maintained at least 2300 ppm. The proposed change to allow reduced IFBA B–10 loading does not invalidate these prior analyses since equivalent reactivity hold down to the 2.35 mg/linear inch B–10 loading will be maintained.

(c) Introduction of Moderator to the New Fuel Vault

There is no increase in the probability of any accident involving moderator introduction to the new fuel storage vault. The proposed change affects only the enrichment within the fuel assemblies. No other plant systems or components are affected by this change.

There is no increase in the consequences of introducing a moderator to the new fuel storage vault resulting from increased fuel enrichment. The new fuel storage vault has been analyzed for storage of fuel assemblies with nominal enrichments of 4.65 w/o U^{235} at the fully flooded condition and 5.00 w/o U^{235} at the optimum moderation condition, as described in the attached Criticality Analysis (Attachment 2). As long as the requirement for the number of IFBA pins versus assembly enrichment is met, calculated K_{eff} (including uncertainties and biases) does not exceed 0.95 under full density conditions and does not exceed 0.98 under optimum moderation conditions.

These analyses demonstrate that 5.0 w/o enrichment fuel storage in the New Fuel Storage Vault complies with criticality acceptance criteria for all moderation conditions. Therefore, based on the conclusions of the above analyses, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do not create the possibility of a new or different kind of accident from any accident previously analyzed.

The proposed Technical specification changes do not involve any physical changes to the plant or any changes to the method in which the plant is operated. No physical changes to the new fuel or spent fuel storage racks are required, nor any changes in the process or procedures to place fuel in the racks. The enrichment limits and reactivity hold-down requirements ensure that the assumptions used in the criticality analyses remain bounding. As such, these changes do not affect the performance or qualification of safety-related equipment. Therefore, the possibility of a new or different type of accident than previously considered i[s] not created.

3. Do not involve a significant reduction in a margin of safety.

The new fuel storage vault has been analyzed for storage of fuel assemblies with nominal enrichments of 4.65 w/o U^{235} at the fully flooded condition and 5.00 w/o U^{235} at the optimum moderation condition, as described in the attached Criticality Analysis (Attachment 2). As long as the requirement for the number of IFBA pins versus assembly enrichment in Equation 1 is met, calculated K_{eff} (including uncertainties and biases) does not exceed 0.95 under full density conditions and does not exceed 0.98 under optimum moderation conditions.

For the 5.00 w/o U^{235} enrichment requested, Equation 2, which bounds Equation 1, will be used in the Technical Specifications related to new fuel storage.

Therefore, since the calculated values of K_{eff} have been shown to be below the regulatory limits (including uncertainties and biases) and because they reflect a substantial subcritical configuration under adverse conditions, the proposed changes will not result in a significant reduction in the plant's margin of safety.

Previous analyses provided in support of Amendments 151/131 demonstrate that the addition of new fuel having IFBA pins with a loading of 2.35 mg B–10 per linear inch to the spent fuel racks does not result in a reduction in the margin of safety. Thus, providing for reactivity hold down for IFBA pins which is equivalent to a nominal 2.35 mg B–10/linear inch loading in fresh fuel in the spent fuel storage racks maintains the current 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: Salem Free Public Library, 112 West Broadway, Salem, NJ 08079.

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

NRC Project Director: Elinor G. Adensam.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of amendment request: February 5, 1999.

Description of amendment request: The proposed amendments would change the Technical Specifications (TSs) to incorporate some of the generic changes to the Improved Technical Specifications that have been previously approved by the NRC. In addition, a TS has been added that would test the Unit 1 automatic scram relay on a periodic 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, which is presented below:

1. The proposed changes do not involve a significant increase in the probability or the consequences of a previously evaluated event for the following reasons:

Proposed Change One

The majority of primary containment isolation valves (PCIVs) should be in the closed position following an accident to prevent the release of radiation to the environment. Locked PCIVs are verified to be in the closed position prior to being locked. Therefore, it is unnecessary for these valves to be verified closed under the provisions of Surveillance Requirements (SRs) 3.6.1.3.2 and 3.6.1.3.3. The fact that the valves are secured closed assures they will be in the safe position following an accident. Furthermore, per Plant Hatch procedure, locked valves are periodically verified to be in their correct position. This provides additional assurance the valves will remain in the correct position. For these reasons, the proposed change does not involve a significant increase in the probability or the consequences of a previously evaluated event.

Proposed Change Two

This proposed change does not affect the function of the control rods, the control rod drive (CRD) system, or the control rod housing. Thus, the probability of the control rod drop accident (CRDA) is not increased. Also, this change does not affect the function of the rod worth minimizer (RWM). As with the present Technical Specification, no control rods will be moved (via SRs 3.1.3.2 and 3.1.3.3) when below the low power setpoint (LPSP) to limit interference with respect to the RWM's function in limiting the consequences of a CRDA. Additionally, no other systems designed to prevent or mitigate the consequences of any other transient or accident are affected.

Proposed Change Three

This proposed change merely deletes a redundant specification in the control rod operability section. The requirement to electrically disarm an inoperable withdrawn control rod ensures the validity of banked position withdrawal sequence (BPWS) is maintained, thus ensuring the mitigation of the consequences of the CRDA. This proposed change in no way affects the BPWS, the RWM, or the structures of the control rods and control rod drive. Thus, the probability, or the consequences, of a previously evaluated event are not increased by this proposed change.

Proposed Change Four

Any physical deterioration of a station service battery that can cause degradation of battery performance will result in failure of the SR, with the ensuing inoperable declaration of the battery. A determination that battery performance is not degraded, or will not degrade, will result from evaluation of the particular abnormality found while performing the Surveillance. This is the intent of the Technical Specification as clarified in the Bases.

Accordingly, the safety function of the station service batteries is not compromised as a result of this proposed change. Thus, the consequences of a previously evaluated event are not affected by this proposed revision. The proposed revision does not affect any system needed to prevent the occurrence of previously analyzed events; therefore, the probability of occurrence of a previously evaluated event is not increased.

Proposed Change Five

The purpose of the primary containment air interlock is to provide access to the primary containment while maintaining containment integrity. Extending the Surveillance Frequency on the airlock to once per 24 months will not increase the likelihood of occurrence of any previously evaluated event, since no change in the operation or testing of any system designed for the prevention of accidents and transients is being made.

Extending the Frequency of the airlock interlock Surveillance does not increase the consequences of any accident or transient, since the proposed change does not affect any system designed to mitigate the consequences of a previously analyzed event. In fact, the extended Frequency will challenge the airlock interlock less; thus, the likelihood of a loss of primary containment integrity will decrease.

Proposed Change Six

This proposed change to the Safety Function Determination Program (SFDP) description in LCO [Limiting Condition for Operation] 3.0.6 is more restrictive than the existing version. Requiring an SFDP evaluation upon entry into LCO 3.0.6, as stated in the Bases, will not increase the probability of occurrence or the consequences of a previously evaluated event, since this is purely an administrative change to clarify the intent of LCO 3.0.6 and provide consistency with the Bases.

Proposed Change Seven

This proposed administrative change merely relocates the review requirements for the Offsite Dose Calculation Manual (ODCM) to licensee controlled documents. This change does not affect any system designed for the prevention or mitigation of previously analyzed events or any assumptions regarding transient and accident analyses.

Proposed Change Eight

This proposed administrative change eliminates some of the redundant reporting requirements for safety limit violations listed in the Technical Specifications. This change does not affect any systems designed for the prevention or mitigation of any previously evaluated accident or transient. Additionally, the change does not affect any assumptions of previously evaluated accidents or transient analyses.

Proposed Change Nine

This change adds a footnote to Unit 1 Technical Specifications Table 3.3.1.1–1 to ensure the auto scram relays (K14s) are tested as part of the manual scram Functional Test. This change does not adversely affect the ability of the reactor protection system (RPS) to perform its safety function. In fact, the added testing requirement enhances the ability to detect and correct problems with the RPS. Successful testing of the K14s on a weekly basis for many years has demonstrated that the additional testing requirements do not impose an undue burden on the system. No other systems designed for the prevention or mitigation of accidents are affected by this change. Therefore, the probability, or the consequences, of a previously evaluated event are not increased.

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

Proposed Change One

Removing the SR to verify locked valves are in their ''safe'' position does not increase the likelihood of occurrence or consequences of a new type of event, since no new modes of operation are introduced. All plant systems will continue to be operated within their design basis. Since the valves are verified to be in their safe position prior to locking, and are periodically verified to be in that position per the locked valve procedure, the valves will be in the position assumed by accident analyses should an event occur.

Proposed Change Two

This proposed change does not affect the function of either the CRD system or the RWM. These systems, as well as all other systems designed for the prevention or mitigation of accidents, will continue to function per their design basis. Also, the BPWS will continue to be used for control rod withdrawal. Thus, no new modes of operation that would cause a type of failure different from any previously analyzed are introduced.

Proposed Change Three

Deleting Required Action B.1 of Technical Specification 3.1.3 does not eliminate any Required Actions, since the subject Required Action is redundant. Deleting the redundant specification does not prevent any of the control rod control systems from performing their functions per their design bases. Therefore, no new modes of operation are introduced, and the probability of a new type event is also not introduced by this proposed change.

Proposed Change Four

No changes to the operation, maintenance, or testing of the batteries are proposed. The batteries will continue to operate within their design basis. As a result, no new modes of operation are introduced, and thus, the probability of occurrence of a new type event is not created.

Proposed Change Five

This change is administrative in the sense that it does not result in the airlock being operated or tested outside of its design. The proposed revision only includes a change to the Frequency of SR 3.6.1.2.2, which tests the interlock's ability to prevent the two primary containment airlock doors from opening at the same time. This change does not affect how the test is to be performed or how the doors are operated. Therefore, the probability of occurrence of a new type event is not increased by the proposed change.

Proposed Change Six

This proposed administrative change to the SFDP description does not involve the operation of any safety-related system. Furthermore, this change does not involve accident or transient analyses; thus, no changes to the assumptions for the analyses are made. As a result, the probability of occurrence of a new type event is not increased.

Proposed Change Seven

This administrative change merely relocates the review requirements for the ODCM to licensee controlled documents. This change does not affect any system designed for the prevention or mitigation of previously analyzed events or any assumptions regarding transient and accident analysis. Accordingly, the possibility of a new type event is not created.

Proposed Change Eight

This administrative change eliminates some of the redundant reporting requirements for safety limit violations listed in the Technical Specifications. This change does not affect any systems designed for the prevention or mitigation of any previously evaluated accident or transient. Additionally, the change does not affect any assumptions of previously evaluated accident or transient analyses. Accordingly, the possibility of a new type event is not created.

Proposed Change Nine

Adding a requirement to test the auto scram relays (K14s) on a weekly basis does not create a new mode of operation for the RPS. Also, no other safety-related systems are affected by this change, and as a result, the possibility of occurrence of a new type accident is not created.

3. The changes do not significantly reduce the margin of safety.

Proposed Change One

Not requiring position surveillance on PCIVs locked in position does not reduce the margin of safety, because the valves are verified to be in their ''safe'' position prior to locking. This ensures the valve will remain in the ''safe'' position until it is unlocked again. The position of these locked valves is verified periodically by the Operations Department. Furthermore, a ''malicious'' unlocking of the valves is unlikely to take place, since the keys to the valves are controlled by the shift supervisor (SS). Anyone wanting to check out a key must obtain SS approval. Also, the locked valves are periodically verified to be in their proper position whenever Operations Management deems it necessary. For these reasons, the margin of safety is not significantly reduced.

Proposed Change Two

Moving the Technical Specification 3.1.3 Note from the Required Action column to the Completion Time column will not affect the safety function of the RWM system. The RWM will continue to function through the power ranges where the control rod drop accident is of concern. The change does not affect the safety function of the RWM in any way. Thus, the margin of safety is not reduced.

Proposed Change Three

This proposed change only eliminates a redundant Specification. Adherence to the requirements of the BPWS will still be maintained during plant startups. Also, the operation of the RWM system remains unaffected by this proposed change. For these reasons, the margin of safety for the CRDA is not reduced.

Proposed Change Four

This proposed change clarifies that the purpose of SR 3.8.4.3 is to determine whether a physical deterioration that could affect battery performance exists. This is already stated in the Plant Hatch Technical Specifications Bases; thus, the proposed revision is merely a clarification of the Specification. Adding this clarification does not reduce the margin of safety with respect to battery performance, because an engineering evaluation must be performed to document that the particular deficiency will not prevent the battery from performing its safety function.

Proposed Change Five

This proposed change to extend the Frequency of SR 3.6.1.2.2 reduces the number of challenges to primary containment integrity. The nature of the Surveillance is such that the primary containment (drywell) interlock is challenged. With that challenge, the likelihood of a primary containment breach is increased. Therefore, reducing the Frequency of this SR actually increases the safety of margin, since normal entry and exit procedures do not permit challenging the interlock.

Proposed Change Six

This purely administrative change clarifies the definition of the SFDP in LCO 3.0.6. The Technical Specifications margin of safety is enhanced, since the new wording, together with the existing wording in the Bases, makes it clear that the SFDP must be performed any time LCO 3.0.6 is entered.

Proposed Change Seven

This proposed change merely allows relocation of the review and approval functions for the ODCM revisions from the Technical Specifications to owner-controlled documents. The purely administrative change does not affect any Technical Specifications required system, test, or function. Changes to the ODCM will continue to receive the level of review necessary to ensure any proposed changes are accurate and complete. Therefore, the margin of safety is not reduced.

Proposed Change Eight

This purely administrative change eliminates redundant reporting requirements with respect to a safety limit violation. The change has no effect on any Technical Specifications required system, test, or function, or on any other safety-related system. Accordingly, the margin of safety is not reduced.

Proposed Change Nine

This proposed change ensures the Unit 1 auto scram relays (K14s) are tested on a weekly basis. General Electric recognizes this as an optimum test frequency for these scram contactors. In this respect, the margin of safety is increased, since this change ensures the relays will be tested at the optimum recommended Frequency. Also, at Plant Hatch, the K14 relays and contacts have been tested at this Frequency for many years. As a result, placing this requirement on the relays will not pose an undue burden on the RPS. No other safety-related systems are affected by this proposed change. For the above reasons, this proposed change does not 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 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: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia.

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC.

NRC Project Director: Herbert N. Berkow.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas.

Date of amendment request: January 26, 1999.

Description of amendment request: The amendment would revise part of the Inservice Inspection requirements for the Reactor Coolant Pump flywheel from an in-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and keyway at approximately 3-year intervals and a surface examination of all exposed surfaces and complete ultrasonic volumetric examination at approximately 10 year intervals to ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius once every 10 years.

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

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

The proposed change increases the examination volume and revises the periodicity of the ultrasonic examination required by Regulatory Guide 1.14 regulatory position C.4.b(1) from 3-year intervals to 10-year intervals. This change is consistent with the conclusions of the NRC staff in the referenced safety evaluation of WCAP–14535. The NRC staff has determined that the evaluation methodology is appropriate and the criteria are in accordance with the design criteria of RG 1.14. There is no change in the method of plant operation or system design.

The proposed change revises the inspection process to eliminate 10-year surface examination of all exposed surfaces and complete ultrasonic volumetric examination required by Regulatory Guide 1.14 Regulatory Position C.4.b(2). An ultrasonic volumetric examination will be performed of a section of the flywheel once every 10 years. This change is consistent with the conclusions of the NRC staff in referenced safety evaluation of WCAP–14535. The NRC staff has determined that the evaluation methodology is appropriate and the criteria are in accordance with the design criteria of RG 1.14.

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

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

The proposed change increases the examination volume and revises the periodicity of the ultrasonic examination required by Regulatory Guide 1.14 regulatory position C.4.b(1) from 3-year intervals to 10-year intervals. This change is consistent with the conclusions of the NRC staff in the referenced safety evaluation of WCAP–14535. The only potential accident associated with this change is loss of the flywheel. Precautionary measures taken to preclude missile formation from Reactor Coolant Pump components assure that the pumps will not produce missiles under any anticipated accident condition. Each component of the primary pump motors has been analyzed for missile generation Any fragments of the motor rotor would be contained by the heavy stator. Effects on reactor coolant flow due to loss of functionality of a single Reactor Coolant Pump flywheel are enveloped by the analysis of the consequences of the Reactor Coolant Pump locked rotor event. There is no change in the method of plant operation or system design.

The proposed change revises the inspection process to eliminate 10-year surface examination of all exposed surfaces and complete ultrasonic volumetric examination required by Regulatory Guide

1.14 Regulatory Position C.4.b(2). An ultrasonic volumetric examination will be performed of a section of the flywheel once every 10 years. This change is consistent with the conclusions of the NRC staff in the referenced safety evaluation of WCAP–14535. The only potential accident associated with this change is loss of the flywheel. Precautionary measures taken to preclude missile formation from Reactor Coolant Pump components assure that the pumps will not produce missiles under any anticipated accident condition. Each component of the primary pump motors has been analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator. Effects on reactor coolant flow due to loss of functionality of single Reactor Coolant Pump flywheel are enveloped by the analysis of the consequences of the Reactor Coolant Pump locked rotor event.

Based on the above, this 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.

The proposed change increases the examination volume and revises the periodicity of the ultrasonic examination required by Regulatory Guide 1.14 Regulatory Position C.4.b(1) from 3-year intervals to 10-year intervals. This change is consistent with the conclusions of the NRC staff in the referenced safety evaluation of WCAP–14535. The NRC staff used deterministic methodology to review the WCAP and came to the conclusion that ASME margins would be maintained during the service period and a 10-year inspection period appears reasonable. There is no change in the method of plant operation or system design.

The proposed change revises the inspection process to eliminate the 10-year surface examination of all exposed surfaces and complete ultrasonic volumetric examination required by Regulatory Position C.4.b(2) of Regulatory Guide 1.14. An ultrasonic volumetric examination will be performed of a section of the flywheel once every 10 years. This change is consistent with the conclusions of the NRC staff in the referenced safety evaluation of WCAP–14535. Effects on reactor coolant flow due to loss of functionality of a single Reactor Coolant Pump flywheel are enveloped by the analysis of the consequences of the Reactor Coolant Pump locked rotor event.

Based on the above, this change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, N.W., Washington, DC 20036–5869.

NRC Project Director: John N. Hannon.

Tennessee Valley Authority, Docket No. 50–328, Sequoyah Nuclear Plant, Unit 2, Hamilton County, Tennessee

Date of application for amendments: August 27, 1998 (TS 98–04).

Brief description of amendments: The proposed amendment would change the Sequoyah (SQN) Technical Specifications (TSs) by adding a provision to Section 5.3, ''Reactor Core,'' authorizing a limited number of lead test assemblies (LTAs) to be installed in the core as described in the Framatome Cogema Fuels Report BAW–2328 entitled ''Blended Uranium Lead Test Assembly Design Report.''

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

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

The LTAs are identical to the other Mark-BW fuel assemblies with the exception of minor differences internal to the fuel rods. These differences will not adversely affect reactor neutronic or thermal-hydraulic performance; therefore, they do not significantly increase the probability of accidents while in the reactor.

The reload design analyses performed for SQN Unit 2 Cycle 10 accounts for any minor neutronic differences of the LTAs and confirms any effects on the reload core to be within established fuel design limits.

The pressure and temperature safety limits for the cycles in which the LTAs will be in the core are the same as those for the current operating cycle thus ensuring that the fuel will be maintained within the same range of safety parameters that form the basis for the FSAR [Final Safety Analysis Report] accident evaluation. The potential effects of the LTAs on plant operation and safety have been evaluated. This evaluation investigated both LOCA [loss-of-coolant accident] and non-LOCA events, and concluded that the current analyses remain bounding and that there will be no increase in the probability of occurrence for any design basis accident described in the FSAR.

The impact of the LTAs on key safety analysis parameters was examined and it was concluded that there will be an insignificant impact.

The impacts of the LTAs on the radiological consequences for all postulated events have been evaluated. The total calculated source term and the source term-activity of isotopes, which significantly contribute to operator and off-site accident exposure levels, were shown to be less than standard fuel assemblies, therefore, it will not increase the consequences of any accident previously evaluated.

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

The fuel assembly design for the LTAs is identical to the standard fuel assemblies. The main difference between the LTAs and the production fuel is that the concentration of the U^{234} and U^{236} isotopes will be higher in the LTA fuel pellets than that typically found in standard fuel. These isotopic differences will not affect the chemical, mechanical or thermal properties of the fuel pellet.

The LTAs meet the same design criteria and licensing basis criteria as the standard fuel assemblies and were manufactured with the same processes. The LTA skeleton is identical to the standard skeleton, which ensures that the loadings associated with normal operation, seismic events, LOCA events, and shipping and handling are not affected.

Pressure and temperature safety limits will be maintained the same as those for the current operating cycle, thus ensuring that the fuel will be maintained within the same range of safety parameters that form the basis for previous accident evaluations. No new performance requirements are being imposed on any system or component that exceed design criteria or cause the core to operate in excess of design basis operating limits. No credible scenario has been identified, which could jeopardize equipment that could cause intensify or mitigate events or accident sequences. Therefore, the LTAs will not create the possibility of accidents or equipment malfunctions of a different type than previously evaluated while in the reactor.

C. The proposed amendment does not involve a significant reduction in a margin of safety.

The LTAs will not adversely affect reactor neutronic or thermal-hydraulic performance. The LOCA acceptance criteria with LTAs installed in the core will continue to be met: peak cladding temperature of less than or equal to 2200 °F, peak cladding oxidation of less than or equal to 17 percent, average clad oxidation of less than or equal to 1 percent, and long-term coolability. The acceptance criteria for departure from nucleate boiling (DNB) events with the LTAs installed in the core will also continue to be met: 95 percent probability and 95 percent confidence interval that DNB is not occurring during the transient. Other acceptance criteria have also been demonstrated to remain within acceptable limits. The total calculated source term-activity and the source term-activity of isotopes, which significantly contribute to operator and off-site accident exposure levels of the LTAs, was determined to be less than that for the standard fuel assembly. All previously evaluated events remain bounding and valid. For these reasons, the proposed amendment does not involve a significant reduction in a margin of safety.

The NRC 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: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Project Director: Cecil O. Thomas.

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: August 14, 1998, as supplemented by letters dated October 13, 1998, and December 23, 1998.

Brief description of amendments: The amendments revised the Dresden Technical Specifications (TS) to reflect the use of Siemens Power Corporation ATRIUM–9B fuel. Specifically the amendments incorporated the following into the TS: (a) new methodologies that enhanced operational flexibility and reduced the likelihood of future plant derates; (b) administrative changes that eliminated the cycle-specific implementation of ATRIUM–9B fuel and adopted Improved Standard Technical Specification language where appropriate; and (c) changed the Minimum Critical Power Ratio.

Date of issuance: February 16, 1999

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

Amendment Nos.: 171; 166.

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

Date of initial notice in **Federal Register**: 63 FR 48258 (September 9, 1998) and 63 FR 59588 (November 4, 1998). The October 13 and December 23, 1998 submittals provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 16, 1999.

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 16, 1998.

Brief description of amendments: The amendments revise the Technical Specifications (TS) to lower the power level (from 30 percent to 25 percent rated thermal power) below which the turbine control valve (TCV) and turbine stop valve (TSV) closure scram signals and the end-of-cycle recirculation pump trip (EOC–RPT) signal are not in effect. The amendments also (1) delete from TSs the reference to turbine first stage pressure as a measure of rated thermal power, and (2) add a requirement to periodically verify that TCV and TSV scram trip functions and the EOC–RPT trip functions are not bypassed at greater than or equal to 25 percent rated thermal power.

Date of issuance: February 12, 1999.

Effective date: For Unit 1—Immediately, to be implemented within 90 days; for Unit 2—immediately, to be implemented prior to startup of L2C8.

Amendment Nos.: 130; 114.

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

Date of initial notice in **Federal Register**: November 18, 1998 (63 FR 54108). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 12, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Jacobs Memorial Library, 815 North Orlando Smith Avenue, Illinois Valley Community College, Oglesby, Illinois 61348–9692.

Commonwealth Edison Company, Docket No. 50–374, LaSalle County Station, Unit 2, LaSalle County, Illinois

Date of amendment request: November 9, 1998.

Brief description of amendment: The amendment revised Technical Specification 3/4.3.2, ''Isolation Actuation Instrumentation'' to add/revise various isolation setpoints for leak detection instrumentation. These changes are necessary due to modifications to the reactor water cleanup (RWCU) system to restore ''hot'' suction to the RWCU pumps and due to a re-evaluation of the high energy line break analysis. In addition, the amendment eliminated isolation actuation trip functions for the residual heat removal system steam condensing mode and shutdown cooling mode.

Date of issuance: February 16, 1999.

Effective date: February 16, 1999.

Amendment No.: 115.

Facility Operating License No. NPF–18: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 16, 1998 (63 FR 69335).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 16, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Jacobs Memorial Library, 815 North Orlando Smith Avenue, Illinois

Valley Community College, Oglesby, Illinois 61348–9692.

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of application for amendment: June 20, 1997 (NRC–97–0037), as supplemented July 2, 1997 (NRC–97–0066), and March 10 (NRC–98–0036) and April 9, 1998 (NRC–98–0083).

Brief description of amendment: The amendment revises the technical specifications by relocating surveillance requirement 4.4.1.1.2 for setting the reactor recirculation system motor-generator set scoop tube stops to the updated final safety analysis report (UFSAR), with modifications.

Date of issuance: February 8, 1999.

Effective date: February 8, 1999, with full implementation within 90 days. Implementation of this amendment shall include the relocation of surveillance requirement 4.4.1.1.2 from the technical specifications to the UFSAR as described in the licensee's application dated June 20, 1997, as supplemented on July 2, 1997, and March 10 and April 9, 1998, and evaluated in the staff's safety evaluation dated February 8, 1999.

Amendment No.: 130.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: July 16, 1997 (62 FR 38134)

The July 2, 1997, and March 10 and April 9, 1998, submittals provided additional clarifying information within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 8, 1999.

No significant hazards consideration comments received: No.

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

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of application for amendment: June 30, 1998, as supplemented by letter dated November 23, 1998.

Brief description of amendment: The amendment authorizes the licensee to modify the plant to correct a design deficiency with the plant protection system (PPS). This deficiency could have rendered the system vulnerable to a single failure (i.e., failure of a DC buss) with one channel in bypass. The proposed modification would ensure the required redundancy and independence for the PPS such that no single failure results in a loss of the protection function with a channel in indefinite bypass, and removal from service of any component or channel does not result in a loss of the minimum redundancy required by the Technical Specifications.

Date of issuance: February 17, 1999.

Effective date: This license amendment is effective as of its date of issuance to be implemented within six months following the facility's restart from refueling outage 2R14.

Amendment No.: 201

Facility Operating License No. NPF–6: Amendment revised the license to authorize a modification to the plant protection system.

Date of initial notice in **Federal Register**: December 2, 1998 (63 FR 66593).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 17, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: June 29, 1998, as supplemented by letter dated January 12, 1999.

Brief description of amendment: The amendment changes the Appendix A TSs by modifying TS 3.7.6.1, ''Control Room Emergency Air Filtration System'' in Modes 1–4, TS 3.7.6.2, ''Control Room Emergency Air Filtration System'' in Modes 5 and 6, TS 3.7.6.3, ''Control Room Air Temperature'' in Modes 1–4, TS 3.7.6.4, ''Control Room Air Temperature,'' in Modes 5 and 6, TS 3.7.6.5, ''Control Room Isolation and Pressurization,'' and its associated basis. This amendment also modifies TS Tables 3.3–6 and 4.3–3 for the Control Room Intake Monitors.

Date of issuance: February 17, 1999.

Effective date: This license amendment is effective as of its date of issuance, to be implemented within 60 days.

Amendment No.: 149.

Facility Operating License No. NPF–38: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 21, 1998 (63 FR 56247).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 17, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122.

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

Date of application for amendment: May 28, 1996, as supplemented by letter dated October 27, 1998.

Brief description of amendment: This amendment increases the test interval for reactor protection system instrumentation and anticipatory reactor trip system instrumentation.

Date of issuance: February 22, 1999.

Effective date: February 22, 1999.

Amendment No.: 230.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 31, 1996 (61 FR 40031). The supplemental information provided did not impact the proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 22, 1999.

No significant hazards consideration comments received: No.

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

FirstEnergy Nuclear Operating Company, Docket No. 50–440 Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of application for amendment: September 8, 1997, as supplemented by submittal dated October 27, 1998.

Brief description of amendment: This amendment revised Technical Specification 5.2.2.e, ''Organization—Unit Staff,'' by removing the reference to the NRC Policy Statement on working hours and incorporating a requirement for administrative procedures necessary to ensure that the working hours of unit staff who perform safety-related functions are limited and controlled.

Date of issuance: February 22, 1999.

Effective date: February 22, 1999.

Amendment No.: 98.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 19, 1997 (62 FR

61847) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 22, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, OH 44081.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of application for amendment: September 3, 1998, as supplemented by submittals dated December 3, and December 9, 1998 and January 12, and January 26, 1999.

Brief description of amendment: This amendment revised Technical Specification 3.8.1, ''AC Sources—Operating,'' by extending the emergency diesel generator (EDG) Completion Time from 72 hours to 14 days for the Division 1 and 2 EDG and allows performance of the EDG 24-hour test run in Modes 1 and 2. The amendment also establishes Technical Specification 5.5.13.1, ''Configuration Risk Management Program,'' an administrative program that assesses risk based on plant status.

Date of issuance: February 24, 1999.

Effective date: February 24, 1999.

Amendment No.: 99.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 21, 1998 (63 FR 56261)

The supplemental information provided clarifying information that did not change the initial no significant hazards consideration determination or alter the scope of the proposed action.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 24, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, OH 44081.

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

Date of application for amendment: January 22, 1999.

Brief description of amendment: The amendment revises Technical Specification Surveillance Requirement (SR) 3.8.1.7 to better match plant conditions during diesel generator (DG) testing by clarifying which voltage and frequency limits are applicable during the transient and steady state portions of the DG start. A Notice of Enforcement Discretion (NOED) related to SR 3.8.1.7 was issued verbally on January 20, 1999. The NOED is documented in a letter dated January 22, 1999.

Date of issuance: February 17, 1999.

Effective date: February 17, 1999, to be implemented within 30 days.

Amendment No.: 225.

Facility Operating License No. DPR–49: Amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (64 FR 4902 dated February 1, 1999). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided for an opportunity to request a hearing by March 3, 1999, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final NSHC determination are contained in a Safety Evaluation dated February 17, 1999.

Attorney for Licensee: Al Gutterman; Morgan, Lewis & Bockius, 1800 M Street NW, Washington, D.C. 20036–5869.

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

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

Date of application for amendment: October 22, 1998.

Brief description of amendment: The amendment revises Technical Specifications 3.3.2.1, ''Instrumentation—Engineered Safety Feature Actuation System Instrumentation''; 3.4.9.3, ''Reactor Coolant System—Overpressure Protection Systems''; and 3.5.3, ''Emergency Core Cooling Systems—ECCS Subsystems—Tavg < 300 [degrees] F.'' The amendment allows Millstone Unit No. 2 to prevent an automatic start of any high-pressure safety injection (HPSI) pump when the shutdown cooling system (SDCS) is in operation (Mode 4 and below). An inadvertent start of an HPSI pump could result in overpressurization of the SDCS.

Date of issuance: February 10, 1999.

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

Amendment No.: 227.

Facility Operating License No. DPR–65: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 2, 1998 (63 FR 66600)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 10, 1999.

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.

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: March 10, 1997, as supplemented by letters dated May 20, 1997; March 13, August 28, and October 22, 1998; and January 29 and February 2, 1999.

Brief description of amendments: The amendments revised the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2 that changed TS 3/4.4.5 and its associated Bases to allow the implementation of steam generator (SG) tube alternate repair criteria for axial indications in the Westinghouse explosive tube expansion (WEXTEX) region below the top of the tubesheet and below the bottom of the WEXTEX transition that may exceed the current TS depth-based plugging limit.

Date of issuance: February 19, 1999.

Effective date: February 19, 1999, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 1—129; Unit 2—127.

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

Date of initial notice in **Federal Register**: November 19, 1997 (62 FR 61843). The March 13, August 28, and October 22, 1998; and January 29 and February 2, 1999, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 19, 1999. 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.

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

Date of application for amendment: June 16, 1998.

Brief description of amendment: The amendment changes the Technical Specifications (TSs) by moving certain administrative requirements from the TSs to the Final Safety Analysis Report.

Date of issuance: February 25, 1999.

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

Amendment No.: 188.

Facility Operating License No. DPR–64: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: July 29, 1998 (63 FR 40560).

No significant hazards consideration comments received: No.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 25, 1999.

No significant hazards consideration comments received: No.

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

PP&L, Inc., Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of application for amendment: August 4, 1998, as supplemented by letters dated December 16, 1998, and January 12 and 28, 1999.

Brief description of amendment: This amendment would modify the Susquehanna Steam Electric Station, Unit 2 Technical Specifications to replace figures 2.1.1.2–1 and 2.1.1.2–2, and associated footnotes, with single value minimum critical power ratio Safety Limits of Section 2.1.1.2; remove references from Section 5.6.5 which do not directly support the generation of Core Operating Limits; remove references from Section 5.6.5 which were previously included to address the application of the ANFB–10 correlation to ATRIUM–10 fuel; include Siemiens Power Corporation ANFB–10 topical report in Section 5.6.5; and to change the Bases to reflect the inclusion of the ANFB–10 critical power correlation.

Date of issuance: February 17, 1999.

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

Amendment No.: 154.

Facility Operating License No. NPF–22. This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 9, 1998 (63 FR 48262). The December 16, 1998, and January 12, and 28, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 17, 1999.

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

Date of amendments request: October 12, 1998.

Brief Description of amendments: The amendments revise Technical Specification Section 6, ''Administrative Controls,'' to recognize the additional management positions associated with the steam generator replacement project. The new positions would provide the ability to approve procedures regarding this project, which may affect nuclear safety.

Date of issuance: February 19, 1999.

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

Amendment Nos.: Unit 1—141 and Unit 2—133.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register**: November 18, 1998 (63 FR 64122). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 19, 1999.

No significant hazards consideration comments received: No.

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

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 application for amendments: June 19, 1998, as supplemented by letters dated December 4, 1998, and January 13, 1999.

Brief description of amendments: The proposed changes would modify the technical specifications (TS) to (1) reduce the minimum RCS cold leg temperature (Tc); (2) convert the specified reactor coolant system (RCS) flow from mass units (lbm/hr) to volumetric units (gpm); and (3) eliminate the maximum RCS flow rate limit from the TS.

Date of issuance: February 12, 1999.

Effective date: February 12, 1999, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 2—149; Unit 3—141.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 9, 1998 (63 FR 48266). The supplemental letters dated December 4, 1998, and January 13, 1999, provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 12, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713.

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 application for amendments: November 23, 1998, as supplemented by letter dated January 13, 1999.

Brief description of amendments: The amendments revised the technical specifications (TS) to (1) reinstate the log power reactor trip at or above 4E–5% RATED THERMAL POWER (RTP); (2) reinstate reactor trips for Reactor Coolant Flow—Low (RCS flow), the Local Power Density—High (LPD), and the Departure from Nucleate Boiling Ratio—Low (DNBR); (3) remove the word ''automatically'' from notes (a) and (d) of Table 3.3.1–1 to clarify that the manual enable of the trip is permissible; and (4) clarify that the setpoints on Table 3.3.1–1 are set relative to logarithmic power.

Date of issuance: February 12, 1999.

Effective date: February 12, 1999, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 2—150; Unit 3—142.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 30, 1998 (63 FR 71973). The January 13, 1999, supplemental information 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 12, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: November 23, 1998.

Brief description of amendments: Relocates descriptive design information from Technical Specification (TS) Section 3.7.1.1, Table 3.7–2, regarding orifice sizes for main steam line Code safety valves, to the Bases section for this TS.

Date of issuance: February 24, 1999.

Effective date: This license amendment is effective as of its date of issuance, and shall be implemented within 30 days of issuance.

Amendment Nos.: Unit 1—Amendment No. 103; Unit 2—Amendment No. 90.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 30, 1998 (63 FR 71974).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 24, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

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

Date of application for amendment: December 10, 1998.

Brief description of amendment: The amendment corrects an error in the technical specifications by changing to the use of ''hydrogen balance air'' rather than the incorrect ''hydrogen balance nitrogen'' for calibration of the Augmented Offgass System hydrogen monitors.

Date of Issuance: February 12, 1999.

Effective date: February 12, 1999, to be implemented within 30 days.

Amendment No.: 166.

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

Date of initial notice in **Federal Register**: December 30, 1998 (63 FR 71975).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated February 12, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 20, 1997, as supplemented on September 18, 1997, and October 31, 1997.

Brief description of amendment: The amendment makes administrative changes to the Technical Specifications to add and revise reference to NRC-approved methodologies which will be used to generate the cycle-specific thermal operating limits in the Vermont Yankee Core Operating Limits Report.

Date of Issuance: February 23, 1999.

Effective date: February 23, 1999, to be implemented within 30 days.

Amendment No.: 167.

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

Date of initial notice in **Federal Register**: March 25, 1998 (63 FR 14489).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated February 23, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 10, 1996, as supplemented on January 22, 1999.

Brief description of amendment: The amendment makes changes to the Technical Specifications regarding fire protection requirements as recommended by NRC Generic Letters 86–10 and 88–12. This includes relocating certain fire protection requirements to the Vermont Yankee Fire Protection Plan, Technical Requirements Manual, and Final Safety Analysis Report.

Date of Issuance: February 24, 1999.

Effective date: February 24, 1999, to be implemented within 30 days.

Amendment No.: 168.

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

Date of initial notice in **Federal Register**: February 26, 1997 (62 FR 8801).

The January 22, 1999, supplement did not change the original proposed no significant hazards consideration determination, or expand the scope of the amendment request as initially noticed.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated February 24, 1999.

No significant hazards consideration comments received: No.

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

Virginia Electric and Power Company, et al., Docket Nos. 50–280 and 50–281, Surry Power Station, Units 1 and 2, Surry County, Virginia

Date of application for amendments: November 4, 1998.

Brief Description of amendments: These amendments revise the Technical Specifications (TS) to change Emergency Diesel Generator start and load time testing requirements in TS 4.6.A.1.b. The TS Basis Section 3.16 is also revised to reflect the basis for the new TS requirements. The TS changes are in a conservative direction, and are being made to bring the TS and the Updated Final Safety Analysis Report into conformance with each other.

Date of issuance: March 1, 1999.

Effective date: March 1, 1999.

Amendment Nos.: 218 and 218.

Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Technical Specifications.

Date of initial notice in **Federal Register**: January 27, 1999 (64 FR 4161).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 1, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Swem Library, College of William and Mary, Williamsburg, Virginia 23185

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

Date of application for amendment: March 4, 1998, as supplemented September 21, 1998.

Brief description of amendment: This amendment revises the Technical Specifications to provide a one-hour limiting condition for operation that will permit a safety injection pump to be used for the addition of make-up fluid to safety injection accumulators during power operation.

Date of issuance: February 23, 1999.

Effective date: February 23, 1999.

Amendment No.: 143.

Facility Operating License No. DPR–43: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 8, 1998 (63 FR 17237).

The September 21, 1998, supplement provided clarifying information that did not change the initial no significant hazards determination or alter the scope of the proposed action.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 23, 1999.

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 3rd day of March 1999.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 99–5751 Filed 3–9–99; 8:45 am]

BILLING CODE 7590–01–P

RAILROAD RETIREMENT BOARD

Proposed Collection; Comment Request

SUMMARY: In accordance with the requirement of Section 3506(c)(2)(A) of the Paperwork Reduction Act of 1995 which provides opportunity for public comment on new or revised data collections, the Railroad Retirement Board (RRB) will publish periodic summaries of proposed data collections.

Comments are Invited on

(a) Whether the proposed information collection is necessary for the proper performance of the functions of the agency, including whether the information has practical utility; (b) the accuracy of the RRB's estimate of the burden of the collection of the information; (c) ways to enhance the quality, utility, and clarity of the information to be collected; and (d) ways to minimize the burden related to the collection of information on respondents, including the use of automated collection techniques or other forms of information technology.

Title and Purpose of Information Collection

Evidence for application of Overall Minimum: OMB 3220–0083 Under Section 3(f)(3) of the Railroad Retirement Act (RRA), the total monthly benefits payable to a railroad employee and his/her family are guaranteed to be no less than the amount which would be payable if the employee's railroad service has been covered by the Social Security Administration. The Social Security Overall Minimum Guarantee is prescribed in 20 CFR 229. To administer this provision, the Railroad Retirement Board (RRB) requires information about a retired employee's spouse and child(ren) who would not be eligible for benefits under the RRA but would be eligible for benefits under the Social Security Act if the employee's railroad service had been covered by that Act. The RRB obtains the required information by the use of forms G–319 (Statement Regarding Family and Earnings for Special Guaranty Computation) and G–320 (Statement by Employee Annuitant Regarding Student Age 18–19). One form is completed by each respondent. Form G–319 is being revised to request information regarding a student's earnings and entitlement to other benefits. Reformatting, editorial and cosmetic revisions are also being proposed to Form G–319. Reformatting, and editorial revisions (including the deletion of information items requested on the proposed G–319) are proposed to Form G–320.

Estimate of Annual Respondent Burden

The estimated annual respondent burden is as follows:

Form#(s)	Annual responses	Time (Min)	Burden (Hrs)
G–319			
Employee Completed:			
With assistance	95	26	41
Without assistance	5	55	5
Spouse completed:			
With assistance	95	30	48
Without assistance	5	60	5
G–320			
With assistance	86	10	14
Without assistance	4	26	2
Total	290	....	115

FOR FURTHER INFORMATION CONTACT: To request more information or to obtain a copy of the information collection justification, forms, and/or supporting material, please call the RRB Clearance Officer at (312) 751–3363. Comments regarding the information collection should be addressed to Ronald J. Hodapp, Railroad Retirement Baord, 844 N. Rush Street, Chicago, Illinois 60611–2092. Written comments should be received within 60 days of this notice.

Chuck Mierzwa,

Clearance Officer.

[FR Doc. 99–5841 Filed 3–9–99; 8:45 am]

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