inspection and copying at the Commission's Public Document Room, located at 2120 L Street NW., Washington DC. 20555 and NRC's Local Public Document Room located at the Apollo Memorial Library, 219 North Pennsylvania Avenue, Apollo, PA 15613.

For further information, contact Dominick Orlando, US NRC, Mailstop T–8F37, Washington, DC 20555–001, telephone (301) 415–6947.

Dated at Rockville, Maryland, this 24th day of June, 1997.

For the Nuclear Regulatory Commission.

John W.N. Hickey,

Chief, Low-Level Waste and Decommissioning Projects Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 97–17296 Filed 7–1–97; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Monday, June 30, 1997.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public.

MATTERS TO BE CONSIDERED:

Monday, June 30

9:00 a.m. Affirmation Session (Public Meeting) A: Louisiana Energy Services Petitions for Review of LBP–97–8 (May 1, 1997)

The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292.

CONTACT PERSON FOR MORE INFORMATION: Bill Hill (301) 415–1661.

* * * * *

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

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

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn. Operations Branch, Washington, D.C. 20555 (301–415–1661).

In addition, distribution of this meeting notice over the internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: June 27, 1997.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 97–17461 Filed 6–30–97; 10:49 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from June 9, 1997, through June 20, 1997. The last biweekly notice was published on June 18, 1997 (62 FR 33117).

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 Review and Directives Branch, Division of Freedom of Information and Publications 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 August 1, 1997, 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: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

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

Date of amendment request: May 6, 1997.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.7.5, ''Ultimate Heat Sink,'' and the associated bases to support steam generator replacement and to incorporate recent Ultimate Heat Sink (UHS) design evaluations. The replacement steam generators have a larger primary side volume which results in a larger mass/energy release to the containment in the event of a loss-of-coolant accident (LOCA), and a corresponding increase in the heat load to the UHS.

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.

TS 3/4.7.5 establishes the operating requirements for the UHS. Operation of the UHS within its design basis ensures the following: (1) Sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions and (2) adequate inventory is available to provide a 30-day cooling water supply to safety related equipment. Design analyses supporting the proposed TS changes provide full qualification of the UHS.

A loss of off site power (LOOP) coincident with a loss of coolant accident (LOCA), designated a LOOP/LOCA, on one unit, in conjunction with the non-accident unit proceeding to an orderly shutdown and cooldown from maximum power using normal operating procedures, remains the limiting design basis event for the UHS basin temperature.

The proposed changes to the UHS Limiting Condition for Operation for basin temperature and the number of fans running do not, in themselves, factor into any initiating event for Updated Final Safety Analysis Report (UFSAR) Chapter 15 accidents and, consequently, do not increase the probability of occurrence for these previously evaluated accidents.

The UHS plays a vital role in mitigating the consequences of any accident or transient. The proposed changes will ensure that the minimum conditions necessary for the UHS to perform its design functions will always be met. Engineering calculations demonstrate that the SX [essential service water] pump discharge design temperature limit of 100°F, which was assumed as an initial input for the accident analyses, is preserved. Consequently, the proposed changes to the number of cooling tower fans required to be running in high speed relative to the SX pump discharge temperature do not increase the consequences of any accident previously evaluated.

The two unit plant trip from full power with the loss of normal auxiliary feedwater (AF) supply source has been shown to be more limiting than the LOOP/LOCA scenario for UHS makeup and volume considerations.

The proposed changes to the UHS LCO for minimum basin water level do not, in themselves, factor into any initiating event for the UFSAR Chapter 15 accidents and, consequently, do not increase the probability of occurrence for these previously evaluated accidents.

The proposed changes to increase the minimum basin water levels ensure there is a sufficient volume of water in the UHS basin at all times. With these proposed changes, the UHS will perform its design function for the required 30 days, and the consequences of any accident previously evaluated are not increased.

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

The supporting analyses for the revised TS 3/4.7.5 do not involve a new or different kind of accident from any accident previously evaluated. The proposed limits on SX basin minimum water level, maximum basin temperature, and the number of fans operating are within the design capabilities of the UHS, and ensure that the UHS will always be in a condition to perform its design function in the event of an accident or transient. New and revised analyses which support the requested TS changes ensure the full qualification of the UHS. The UHS will not be operated in a different manner such that the possibility of a new or different kind of accident would be created. Consequently, these changes do not create the possibility of a new or different kind of accident from those previously evaluated.

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

The proposed limits on SX basin minimum water level and maximum temperature are based on the results of new and revised design analyses which ensure that the margin of safety is not reduced. Required operator actions with appropriate times are incorporated into the analyses. The new limits on temperature and volume will ensure that, under the most limiting accident or transient scenario, cooling water from the basin will meet the accident analyses SX design temperature limit of 100 degrees Fahrenheit and will ensure that adequate inventory is available to provide a 30-day cooling water supply to safety related equipment. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010.

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: June 12, 1997.

Description of amendment request: The proposed license amendment request would change the licensee's name from ''Duke Power Company'' to ''Duke Energy Corporation'' in the facility operating licenses for the Catawba, McGuire, and Oconee nuclear stations as a result of a corporate merger of Duke Power Company with PanEnergy Corporation.

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

No. These LARs (license amendment requests) involve an administrative change only. The Oconee, McGuire, and Catawba FOLs (Facility Operating Licenses) are being changed to reference the new corporate name of the licensee. No actual plant equipment or accident analyses will be affected by the proposed changes. Therefore, these LARs will have no impact on the possibility of any type of accident: new, different, or previously evaluated.

(2) Will the change create the possibility of a new or different kind of accident from any accident previously evaluated?

No. These LARs involve an administrative change only. The Oconee, McGuire, and Catawba FOLs are being changed to reference the new corporate name of the licensee. No actual plant equipment or accident analyses will be affected by the proposed changes and no failure modes not bounded by previously evaluated accidents will be created. Therefore, these LARs will have no impact on the possibility of any type of accident: new, different, or previously evaluated.

(3) Will the change involve a significant reduction in a margin of safety?

No. Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. These LARs involve an administrative change only. The Oconee, McGuire, and Catawba FOLs are being changed to reference the new corporate name of the licensee.

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: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730.

Attorney for licensee: Mr. Paul R. Newton, Legal Department (PB05E), Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: June 12, 1997

Description of amendment request: The proposed license amendment request would change the licensee's name from ''Duke Power Company'' to ''Duke Energy Corporation'' in the facility operating licenses for the Catawba, McGuire, and Oconee nuclear stations as a result of a corporate merger of Duke Power Company with PanEnergy Corporation.

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

No. These LARs (license amendment requests) involve an administrative change only. The Oconee, McGuire, and Catawba FOLs (Facility Operating Licenses) are being changed to reference the new corporate name of the licensee. No actual plant equipment or accident analyses will be affected by the proposed changes. Therefore, these LARs will have no impact on the possibility of any type of accident: new, different, or previously evaluated.

(2) Will the change create the possibility of a new or different kind of accident from any accident previously evaluated?

No. These LARs involve an administrative change only. The Oconee, McGuire, and Catawba FOLs are being changed to reference the new corporate name of the licensee. No actual plant equipment or accident analyses will be affected by the proposed changes and no failure modes not bounded by previously evaluated accidents will be created. Therefore, these LARs will have no impact on the possibility of any type of accident: new, different, or previously evaluated.

(3) Will the change involve a significant reduction in a margin of safety?

No. Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. These LARs involve an administrative change only.

The Oconee, McGuire, and Catawba FOLs are being changed to reference the new corporate name of the licensee.

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: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, North Carolina 28223–0001.

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: June 12, 1997.

Description of amendment request: The proposed license amendment request would change the licensee's name from ''Duke Power Company'' to ''Duke Energy Corporation'' in the facility operating licenses for the Catawba, McGuire, and Oconee nuclear stations as a result of a corporate merger of Duke Power Company with PanEnergy Corporation.

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

No. These LARs (license amendment requests) involve an administrative change only. The Oconee, McGuire, and Catawba FOLs (Facility Operating Licenses) are being changed to reference the new corporate name of the licensee. No actual plant equipment or accident analyses will be affected by the proposed changes. Therefore, these LARs will have no impact on the possibility of any type of accident: new, different, or previously evaluated.

(2) Will the change create the possibility of a new or different kind of accident from any accident previously evaluated?

No. These LARs involve an administrative change only. The Oconee, McGuire, and Catawba FOLs are being changed to reference the new corporate name of the licensee. No actual plant equipment or accident analyses will be affected by the proposed changes and no failure modes not bounded by previously evaluated accidents will be created. Therefore, these LARs will have no impact on the possibility of any type of accident: new, different, or previously evaluated.

(3) Will the change involve a significant reduction in a margin of safety?

No. Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. These LARs involve an administrative change only. The Oconee, McGuire, and Catawba FOLs are being changed to reference the new corporate name of the licensee.

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: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691.

Attorney for licensee: J. Michael McGarry III, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20036.

NRC Project Director: Herbert N. Berkow.

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: May 30, 1997.

Description of amendment request: Technical Specification (TS) Surveillances 4.5.2.f and 4.6.2.2.b require the periodic flow testing of the recirculation spray system pumps. The proposed amendment would change the surveillances by replacing the pump differential acceptance criteria with a pump acceptance curve.

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 10CFR50.92 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 10CFR50.92(c) are not satisfied. The proposed revision does not involve [an] SHC because the revision would not:

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

The proposed changes to Technical Specification Surveillances 4.5.2.f and 4.6.2.2.b will modify the surveillance acceptance criteria to require that each Recirculation Spray System (RSS) pump develop a differential pressure greater than or equal to the pump performance curve contained on Figure 3.5–1 when tested according to the requirements of Specification 4.0.5. Because it is undesirable to test the pumps on recirculation flow to the RWST [reactor water storage tank], pump testing will now be performed at lower flows than previously performed. Consistent with Specification 4.0.5, one point on Figure 3.5–1 will be used to meet the proposed surveillance acceptance criteria. Periodically comparing the reference differential pressure developed at this reduced flow detects trends that might be indicative of pump degradation. The proposed changes are consistent with RSS pump design criteria and performing surveillance testing does not significantly increase the probability of an accident previously evaluated.

The proposed changes to modify the surveillance acceptance criteria to require that each RSS pump develop a differential pressure greater than or equal to the pump performance curve provides the necessary assurance that the pumps will function as required in previous evaluations and does not significantly increase the consequence of an accident previously evaluated.

Therefore, the proposed changes do 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.

The proposed changes to the surveillance acceptance criteria of the RSS pumps does not change the operation of the Recirculation Spray System or any of its components during normal or accident evaluations.

Therefore, the 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.

The proposed changes will change the surveillance requirements needed to demonstrate operability for each of the RSS pumps. Technical Specification Surveillances 4.5.2.f and 4.6.2.2.b will now require that each pump meet its acceptance criteria in accordance with Figure 3.5–1 when tested according to the requirements of Specification 4.0.5. Figure 3.5–1 will be inserted into the Technical Specifications.

The new acceptance criteria for the RSS Technical Specification surveillance is above the accident analysis curve and is more restrictive than the current inservice inspection curve in the accident analysis region. The proposed TS curve has been degraded in accordance with the recommendations of ASME XI (American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI) for the full range of flow and will be used to meet the TS requirements.

Therefore, 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, CT 06141–0270.

NRC Deputy Director: Phillip F. McKee.

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: June 13, 1997.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) Surveillance Requirement 4.4.1.3.3 to be consistent with the requirements of TS 3.4.1.3. Specifically, the change would bring TS Surveillance 4.4.1.3.3 into agreement with TS 3.4.1.3 that would require at least two reactor coolant system loops to be operable and in operation when the reactor trip system breakers are closed during Mode 4.

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 10CFR50.92 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 10CFR50.92(c) are not satisfied. The proposed revision does not involve (an) SHC because the revision would not:

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

The proposed change to Technical Specification Surveillance 4.4.1.3.3 is being made to bring Technical Specification Surveillance 4.4.1.3.3 into agreement with Technical Specification 3.4.1.3 that requires at least two reactor coolant system loops to be operable and in operation when the reactor trip system breakers are closed during Mode 4. This requirement was incorporated into Technical Specification 3.4.1.3 in Amendment 7. This change to the surveillance does not alter the design, operation, maintenance or testing of the associated systems as previously analyzed.

Therefore, 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 change does not introduce any new failure modes or malfunctions, since the changes only bring Surveillance 4.4.1.3.3 in agreement with Technical Specification 3.4.1.3. Additionally, the proposed change does not alter the operation of the reactor coolant system during normal or accident conditions.

Therefore, the 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.

The proposed change to Technical Specification Surveillance 4.4.1.3.3 will reword the surveillance to ensure compliance with Technical Specification 3.4.1.3. Technical Specification 3.4.1.3 was changed in Amendment No. 7 to address the closure of the Reactor Trip System breakers in Mode 4. As written, Technical Specification Surveillance 4.4.1.3.3 does not adequately ensure compliance with Technical Specification 3.4.1.3. This proposed change is necessary to bring Surveillance 4.4.1.3.3 in agreement with Technical Specification 3.4.1.3 as it was amended.

Therefore, 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, CT 06141–0270.

NRC Deputy Director: Phillip F. McKee.

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

Date of amendment requests: May 7, 1997, as supplemented May 30, 1997.

Description of amendment requests: The proposed amendments would remove from the Technical Specifications certain limitations on crane operations in the spent fuel pool enclosure relating to spent fuel pool special ventilation system operability.

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

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

Operation of the Prairie Island plant in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes do not involve a physical modification to the plant.

The spent fuel pool special ventilation system is provided to mitigate the consequences of a design basis fuel handling accident which involves dropping a spent fuel assembly directly onto a stored spent fuel assembly. Spent fuel pool special ventilation system performance and environmental consequences were based on the conservative assumption that all fuel rods in one fuel assembly fail. However, evaluation of the mechanical performance of spent fuel stored in the spent fuel racks demonstrated that no fuel rods fail.

The proposed changes will continue to require the spent fuel pool special ventilation system to be operable to mitigate the consequences of a fuel handling accident in accordance with its original design intent. Spent fuel pool special ventilation system operability is not required in conjunction with crane operations. Heavy loads in the spent fuel pool enclosure are handled (1) by single-failure-proof cranes with rigging and plant procedures which implement Prairie Island commitments to NUREG–0612 [''Control of Heavy Loads at Nuclear Power Plants''] or (2) over spent fuel pool protective covers as described in the Prairie Island USAR [updated safety analysis report]. In accordance with the requirements of NUREG–0612, use of a single-failure-proof crane with rigging and procedures which implement the requirements of NUREG–0612 assures that the potential for a load drop is extremely small and the effects of heavy load drops are not considered. Spent fuel pool covers prevent dropped loads from falling into the spent fuel pool. Thus, there are no radiological releases resulting from handling heavy loads in the spent fuel pool enclosure for which spent fuel pool special ventilation system operability would be required. Therefore, these changes do not involve a significant increase in the probability or consequences of the fuel handling accident previously evaluated.

2. The proposed amendment(s) will not create the possibility of a new or different kind of accident from any accident previously analyzed.

The proposed Technical Specification changes continue to require the spent fuel pool special ventilation system to be operable during handling of irradiated fuel as originally designed. Heavy loads in the spent fuel pool enclosure are handled by means which assure that the potential for a dropped load is extremely small (through use of single-failure-proof cranes with rigging and plant procedures which implement Prairie Island commitments to NUREG–0612) or prevent dropped loads from falling into the spent fuel pool (through use of spent fuel pool protective covers as described in the USAR). Thus, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes, in themselves, do not introduce a new mode of plant operation, surveillance requirement or involve a physical modification to the plant.

The proposed changes do not alter the design, function, or operation of any plant components and therefore, no new accident scenarios are created. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated would not be created by these amendments.

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

The proposed amendment(s) will continue to require the spent fuel pool special ventilation system to operate following a fuel handling accident as originally designed. Heavy load crane operations in the spent fuel pool enclosure are handled (1) by single-failure-proof cranes with rigging and plant procedures which implement Prairie Island commitments to NUREG–0612; or (2) over spent fuel pool protective covers as described in the Prairie Island USAR. Provision of single-failure-proof equipment and compliance with the other requirements of NUREG–0612 provides an equivalent margin of safety to that which would be demonstrated by analysis of the radiological effects of dropped loads. Use of protective covers has been previously reviewed and approved by the NRC. Therefore, th[ese] proposed amendment(s) (do) not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment 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: John N. Hannon.

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Dockets Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: May 9, 1997.

Description of amendment request: The proposed change revises the Peach Bottom Atomic Power Station, Units 2 and 3 technical specifications to extend the interval for replacing the primary containment purge and exhaust valve inflatable seals.

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 TS (technical specification) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Revising SR [surveillance requirement] 3.6.1.3.16 to replace the inflatable seals for the Primary Containment purge and exhaust valves from every 48 months to every 96 months will not involve a significant increase in the probability or consequences of an accident previously evaluated. The valves will continue to be leak tight throughout the lifetime of the plant. This change will not result in increased onsite or offsite radiological dose. This change will result in reduced occupational dose exposure.

This submittal does not propose any change to the existing requirements contained in the PBAPS [Peach Bottom Atomic Power Station] Technical Specifications for leak testing of the Primary Containment purge and exhaust valves per 10 CFR 50, Appendix J, ''Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors.'' This continued testing will assure the leak tightness of the purge and exhaust valves.

The T-ring materials (Ethylene Propylene) has been found to withstand normal and accident thermal exposures for the design life of the plant based on thermal aging analysis. The elastomer seat material will provide acceptable seat tightness when exposed to a total integrated radiation dose of 10E7 rads based on information provided by EPRI [Electric Power Research Institute] in technical report NP–2129, entitled ''Radiation Effects on Organic Material in Nuclear Plants.'' The radiation dose of 10E7 rads bounds the design basis accident dose to which these valves would be exposed. The radiation dose these valves are exposed to during normal operation is insignificant as compared to the accident dose. Based on this, radiation effects from the additional exposure resulting from the extended replacement frequency will not adversely impact the T-ring seat material.

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

Revising SR 3.6.1.3.16 to replace the inflatable seals for the Primary Containment purge and exhaust valves from every 48 months to every 96 months does not create the possibility of a new or different kind of accident from any accident previously evaluated. This change does not involve any physical changes to a plant structure, system, or component (SSC) which could act as an accident initiator. The design, function, and reliability of the Primary Containment purge and exhaust valves are also not impacted by this change. This activity does not adversely influence any equipment, which is required to be maintained operable for the prevention or mitigation of accidents or transients. Furthermore, implementation of the proposed changes will not adversely affect the manner in which plant SSC are operated.

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

No margins of safety are reduced as a result of the proposed TS changes. The proposed changes do not alter the intended operation of plant structures, systems, or components utilized in the mitigation of accidents or transients. The operating experience of these valves and the testing performed in accordance with 10 CFR 50, Appendix J provides a high level of confidence in the ability of these valves to perform their intended safety function with respect to valve leak tightness.

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: Government Publications Section, State Library of Pennsylvania, (Regional Depository) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

Attorney for Licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General

Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, PA 19101.

NRC Project Director: John F. Stolz.

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Dockets Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: May 23, 1997.

Description of amendment request: The proposed change revises the Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3 Technical Specifications (TS) to exclude the measured Main Steam Isolation Valves (MSIVs) leakage from the total Type B and C local leak rate test (LLRT) results.

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

Excluding the MSIV leakage from the total Type B and C LLRT results does not involve any change in the safety function or method of operation of any plant component, system, or structure. No new accident initiators or failure modes are created as a result of this change. Therefore, this change will not result in an increase in the probability of an accident previously evaluated.

The MSIV leakage release pathway is of significance only for the evaluation of the design basis LOCA (loss-of-coolant accident) as described in the PBAPS, Units 2 and 3 UFSAR (updated final safety analysis report). The doses effectively reflected in the PBAPS, Units 2 and 3 UFSAR reflect the impact of a 0.635% Primary Containment volume per day Primary to Secondary Containment leakage, plus a 0.145% Secondary Containment bypass leakage to the condenser. Since accident consequences already reflect both leakage release pathways, the consequences of the design basis LOCA are not increased.

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

The MSIV's provide the means for mitigating the radiological consequences of an accident. Revising Section 5.5.12 of the PBAPS, Units 2 and 3 TS to exclude the measured MSIVs leakage from the total Type B and C LLRT results has no effect on accident initiators which lead to a new or different kind of accident. This change will not involve any changes to plant systems, structures, or components which could act as new accident initiators. The design, function, and reliability of the MSIVs are also not impacted by this change. Therefore, this change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

No margins of safety are reduced as a result of this change to the TS. No safety limits will be changed as a result of this TS change. The MSIVs will continue to perform their intended safety function. The combined dose rates from the two release paths (i.e., Primary to Secondary Containment leakage and Secondary Containment bypass leakage) are unchanged as a result of this change, and are within the limits of 10 CFR 100, and in conformance with NUREG–0737 post-accident access requirements.

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: Government Publications Section, State Library of Pennsylvania, (Regional Depository) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

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

NRC Project Director: John F. Stolz

Tennessee Valley Authority, Docket No. 50–390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: March 27, May 28, and June 4, 1997.

Description of amendment request: The proposed amendment would revise the technical specifications (TSs) as follows:

Part 1—Boron Concentration Changes

The Cycle 2 core design for Watts Bar (WBN) will include a longer fuel cycle and more highly enriched fuel (from 3.1 percent to 3.7 percent). To accommodate this design, the refueling water storage tank (RWST) and accumulator boron concentrations will be increased to provide enough boron in the sump to meet the large break loss-of-coolant accident (LBLOCA) requirement for sump boron concentration. This requirement is that during a LBLOCA, the core will remain subcritical from boron provided by the emergency core cooling system (ECCS), which takes suction from the RWST and containment sump.

The increase in RWST (TS 3.5.4) and accumulator (TS 3.5.1) boron concentrations will be from a range of 2000–2100 ppm to 2500–2700 ppm and from 1900–2100 to a range of 2400–2700 ppm, respectively. Associated changes are proposed for TS Bases B 3.5.4.

Part 2—Safety Limits, Instrumentation, and Reactor Coolant System

Watts Bar has experienced hot leg temperature fluctuations, including random spikes, which decrease the operating margin to both the overtemperature delta temperature (OTDT) and overpower delta temperature (OPDT) reactor trip setpoints. These fluctuations have caused, in some cases, the plant to experience OT alarms during steady-state operation since the temperature fluctuations reduced the operating margin. To mitigate the temperature fluctuations and associated alarms, the OTDT and OPDT setpoints have been enhanced to increase the operating margin associated with these trip functions.

In addition, Watts Bar has decided to reduce the plant thermal design flow from 97,500 gpm per loop to 93,100 gpm per loop (total of 390,000 gpm) to accommodate 10 percent steam generator tube plugging and a 2 percent reduction in thermal design flow (RTDF).

Also, Watts Bar has decided to implement a tolerance of 0.6°F for the TS Surveillance for indicated differential temperature and 1 °F tolerance for the surveillance of T_{AVG} (identified as T prime and T double prime in the TSs). The use of this tolerance will help to determine whether the indicated DT and T^{AVG} should be left as is, or rescaled during the surveillance. These tolerances have been incorporated as biases into the uncertainty analysis for the affected protection system functions. These functions include the OTDT, OPDT and vessel DT equivalent to power (used in the steam generator low-low water level trip functions). As a result of implementing these biases into the protection system functions (and the changes to the OTDT/OPDT setpoints and reduced TDF), the Allowable Value in the TSs for the OTDT, OPDT and vessel DT equivalent to power functions have been modified.

The licensee's safety evaluation has been prepared to allow for plant operation during Cycle 2 with the revised OTDT and OPDT setpoints, the thermal design flow of 93,100 gpm and the tolerances for indicated differential temperature, T prime and T double prime. To obtain sufficient departure from nucleate boiling (DNB) margin for the OTDT/OPDT setpoint, reduced TDF and Cycle 2 design features, it was necessary to implement the RTDP. The

RTDP program changes the uncertainty treatment for core power, T_{AVG}, pressurizer pressure, and RCS flow. These uncertainties have been incorporated, where applicable, into the safety analyses addressed in the Safety Evaluation.

The following TSs will be changed to incorporate the OTDT/OPDT margin enhancement, thermal design flow of 93,100 gpm and tolerances for indicated differential temperature, T prime and T double prime.

The Reactor Core Safety Limits (TS Figure 2.1.1–1 of the licensee's application) have been modified to improve DNB margin. The Allowable Values for the Vessel DT Equivalent to Power input to Steam Generator Water Level Low-Low in the Reactor Trip System Instrumentation (Table 3.3.1–1, page 4) and Engineered Safety Feature Actuation System (ESFAS) Instrumentation (Table 3.3.2–1, page 4), have been changed to reflect the addition of a 0.6°F tolerance to the measurement of indicated differential temperature.

The revised reactor core safety limits lines allow for changes in the OTDT/OPDT reactor trip setpoints to improve operating margin. The allowable values for these functions in the Reactor Trip System Instrumentation (TS Table 3.3.1–1) have changed as a result of including tolerances for indicated differential temperature, T prime and T double prime in the uncertainty analysis. Several setpoint gains and time constants have been modified to enhance plant operation.

Regarding the RCS Pressure, Temperature and Flow DNB Limits (Section 3.4.1), the RCS average temperature limit has been revised to account for the change in uncertainty from implementing RTDP. The total RCS flow has been modified to account for the reduced thermal design flow from 97,500 gpm to 93,100 gpm. The total flow value in the Technical Specification includes an allowance for instrument uncertainty.

Associated changes have been made to the following TS Bases sections: Reactor Core Safety Limits (Section B 2.1.1); Nuclear Enthalpy Rise Hot Channel Factor (Section B 3.2.2); Reactor Trip System Functions OTDT, OPDT and Steam Generator Water Level Low-low (Vessel Delta T Equivalent to Power) (Section B 3.3.1); Reactor Trip System Functions—Reactor Coolant Flow—Low (Single Loop and Two Loops) (Section B 3.3.1); ESFAS Instrumentation (Section B 3.3.2); RCS Pressure, Temperature, and Flow DNB (Section B 3.4.1).

Part 3—Addition To Core Operating Limit Report Methodologies

The amendment would revise the Core Operating Limits Report (COLR) methodologies listed in TS 5.9.5.b to add the reference to the Westinghouse report WCAP–12610-P-A, ''Vantage + Fuel Assembly Reference Core Report.'' The report reflects use of fuel assemblies in Cycle 2 using ZIRLO fuel rod cladding.

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:

Part 1—Boron Concentration Changes

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92 (c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if 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;

The RWST and accumulator boron concentrations do not affect any initiating event for accidents currently evaluated in the FSAR [final safety analysis report]. The increased concentrations will not adversely affect the performance of any system or component which is placed in contact with the RWST or accumulator water. The integrity and operability of the stainless steel surfaces in the RWST, accumulator and affected NSSS [nuclear steam supply system] components/systems will be maintained. The decrease in solution pH is small and will not degrade the stainless steel. Also, the integrity of the Class 1E instrumentation and control equipment will be maintained since the lower sump pH, resulting from the increased boron concentrations, is still within the applicable equipment qualification [EQ] limits. These limits are set to preclude the possibility of chloride induced stress corrosion cracking and assure that there is no significant degradation of polymer materials. The design, material and construction standards of all components which are placed in contact with the RWST and accumulator water remain unaffected.

For the evaluations, the consequences of an accident previously evaluated in the FSAR will not be increased. There is no increase in the LOCA accident consequences. The changes in the concentrations increase the amount of boron in the sump during a LOCA. The increased boron in the sump is sufficient to maintain the core in a subcritical condition during a LOCA. Also, a revised hot leg switchover time has been calculated and will be implemented in the plant EOPs (emergency operating procedures). Thus, there will be no boron precipitation in the core during a LOCA.

Furthermore, there is no increase in consequences of the non-LOCA events. The concentration changes are a benefit to the SLB (steam line break) at full power analysis due to the reduction in power during the accident. The loss of normal feedwater event is not sensitive to changes in the RWST and accumulator boron concentrations. The concentration changes do not affect the inadvertent operation of ECCS analysis since the minimum DNBR (departure from nucleate boiling ratio) occurs at the event initiation, and the concentration changes do not affect the analysis trend.

Finally, the concentration changes are a benefit for the SLB M&E (mass and energy) release and SGTR (steam generator tube rupture) events since the increased boron increases the available shutdown margin for these events. In addition, the increase in RWST and accumulator boron concentrations and subsequent slight decrease in containment sump and a spray pH does not impact the LOCA dose evaluation since pH is not a function of radionuclide concentration. Therefore, the present analysis remains bounding. Also, the slight decrease in sump, core and spray fluid pH has been evaluated to not impact the corrosion rate (and subsequent generation of Hydrogen) of Aluminum and Zinc inside containment significantly that the present analysis does not remain bounding. Further, the decreased sump, core and spray fluid pH has been evaluated to not affect the amount of hydrogen generated from the radiolytic decomposition of the sump and core solution. In view of the preceding, it is concluded that the proposed change will not increase the consequences of an accident previously evaluated in the FSAR.

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

The changes to the RWST and accumulator concentrations do not cause the initiation of any accident nor create any new credible limiting single failure. The changes do not result in a condition where the design, material, and construction standards of the RWST and accumulators and other potentially affected NSSS components, that were applicable prior to the changes, are altered. * * * *

The changes do not invalidate any of the accident analyses results or conclusions. All of the safety analysis acceptance criteria continue to be met. The changes in the concentrations increase the amount of boron in the sump during a LOCA. The increased boron in the sump is sufficient to maintain the core in a subcritical condition during a LOCA. Also, a revised hot leg switchover time has been calculated and will be implemented in the plant EOPs. Thus, there will be no boron precipitation in the core during a LOCA.

Furthermore, there is no possibility of a different kind of non-LOCA event. The concentration changes are a benefit to the SLB at full power analysis due to the reduction in power increase during the accident. The loss of normal feedwater event is not sensitive to changes in the RWST and accumulator boron concentrations. The concentration changes do not affect the inadvertent operation at ECCS analysis since the minimum DNBR occurs at the event initiation, and the concentration changes do not affect the analysis trend.

Finally, the concentration changes are a benefit for the SLB M&E release and SGTR events since the increased boron increases the available shutdown margin for these events.

(3) or involve a significant reduction in a margin of safety.

The changes do not invalidate any of the non-LOCA safety analysis results or conclusions, and all of the non-LOCA safety analysis acceptance criteria continue to be met. The margin of safety associated with the licensing basis LBLOCA and SBLOCA (small-break loss-of-coolant accident) analyses is not reduced as a result of the proposed changes. Since adequate margin to the PCT (peak cladding temperature) limit of 2200°F has been maintained, no degradation in the margin of safety to the design failure point (fuel melt) has been calculated. The licensing basis containment and steam line break mass and energy releases remain bounding, and the SGTR event acceptance criteria continue to be met. Furthermore, the changes do not affect the safety related performance of the RWST, accumulator or related NSSS components.

Part 2—Safety Limits, Instrumentation, and Reactor Coolant System.

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92 (c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if 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;

The proposed changes do not result in a condition where the design, material, and construction standards, which were applicable prior to the changes, are altered. The revised OTDT and OPDT setpoints do not require any hardware changes and are used for accident mitigation. Thus, the setpoint changes do not increase the probability of the accident.

All of the affected NSSS systems and components have been evaluated with the TDF (thermal design flow) of 93,100 gpm. The primary loop components (reactor vessel, reactor internals, CRDMs (control rod drive mechanism), loop piping and supports, reactor coolant pump, steam generator, and pressurizer) meet the applicable structural limits with the revised TDF of 93,100 gpm and will continue to perform their design functions. The RCCA (rod cluster control assembly) drop time remains unaffected and the current design core bypass flow remains valid. No additional steam generator tubes need to be plugged to mitigate the potential for U-Bend fatigue. Also, all of the NSSS systems will still perform their intended design functions. The pressurizer spray flow remains above the design value and the pressurizer relief system remains unaffected since the TDF is lower than the current design flow and the required pressure drop is lower. The design of the auxiliary system components remains bounding for the revised TDF and the corresponding changes to the NSSS thermal hydraulic parameters. In addition, all of the NSSS/BOP (nuclear steam supply system/balance of plant) interface systems will perform their intended design functions. The steam generator safety valves will provide adequate relief capacity to maintain the steam generator within applicable design limits. The ADVs [atmospheric dump valves] will still relieve 20 percent of the maximum full load steam flow. The steam dump system will still relieve 40 percent of the maximum full load steam flow.

All of the applicable acceptance criteria for the accidents described in the FSAR continue to be met. The LBLOCA analysis currently uses a TDF of 93,100 gpm. Thus, no adjustments are required for the LBLOCA input parameters to accommodate the TDF of 93,100 gpm. The SBLOCA has been performed with the TDF of 93,100 gpm, and the corresponding PCT is well below the 2200°F limit. The post LOCA boron concentration and the hot leg switchover time are unaffected. The revised thermal design procedure has been implemented to obtain sufficient DNB margin to account for the TDF of 93,100 gpm, the new OTDT/OPDT setpoints and the Cycle 2 design features. All of the non-LOCA analyses have been re-analyzed or re-evaluated and all of the applicable acceptance criteria continue to be met.

The SLB radiological doses are unaffected and are still within the existing licensing basis limits. The margin to overfill during the SGTR event has been improved and the offsite doses during an SGTR have been re-calculated and shown to be well within the 10CFR100 guidelines. The plant control systems will still provide adequate response for the Condition 1 transients without causing a reactor trip on OTDT and OPDT.

Finally, the changes in the tolerances for indicated differential temperature, T prime and T double prime do not require any hardware modifications and only require changes to the Technical Specification Allowable Values for the OPDT and OTDT setpoints and for the vessel DT equivalent to power functions. Thus, there is no increase in the probability of an accident since the appropriate Allowable Values have been modified to determine channel operability for these functions.

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

The proposed changes do not cause the initiation of any accident nor create any new limiting single failures. The OTDT and OPDT protection functions are used for accident mitigation and do not initiate any accidents. Also, the affected systems and components will still perform their intended design functions.

* * *

The proposed changes do not create any new failure modes for safety related equipment. The changes do not result in any original design specification, such as seismic requirements, electrical separation requirements or equipment qualification being altered. The OTDT and OPDT setpoint changes do not require any hardware modifications and only require adjustments to the setpoint values. The setpoints are modeled in accident analyses which are used to demonstrate equipment and structural qualification during a SLB. With the setpoint changes and the TDF of 93,100 gpm, the current SLB break M&E releases inside containment remain bounding and thus there is no effect on the qualification of the equipment inside containment during a SLB. The SLB M&E releases outside containment have been re-calculated. The analysis of the impacts on equipment qualification outside containment has been completed by generating new temperature profiles. The application addresses and provides for continued qualification of equipment through the normal EQ program.

Also, with the reduced TDF of 93,100 gpm, the current LOCA M&E releases are still bounding, and thus there is no effect on the qualification of equipment inside containment during a LOCA. The OTDT and OPDT functions are not modeled in the LOCA analyses. Furthermore, all of the applicable compartments and subcompartments will maintain their integrity during the LOCA and the SLB since the mass and energy releases for these compartments and subcompartments remain unaffected.

In addition, the LOCA hydraulic forcing functions remain bounding for the TDF of 93,100 gpm. Thus, the applicable NSSS systems and components will still perform their structural functions during a LOCA.

Finally, the changes in the tolerances for DT_o, T prime and T double prime do not require any hardware modifications and only require changes to the Technical Specification Allowable Values for the OPDT and OTDT setpoints and for the vessel DT equivalent to power functions. Thus, there is no increase in the probability of an accident different than any previously evaluated since the appropriate Allowable Values have been modified to determine channel operability for these functions.

(3) or involve a significant reduction in a margin of safety.

The margin of safety for the applicable safety analyses has not been reduced. The OPDT and OTDT setpoints have been incorporated into the affected safety analyses and all safety analysis criteria continue to be met. All of the applicable DNB limits continue to be met for the non-LOCA analyses. The LBLOCA input parameters do not require adjustment for the TDF of 93,100 gpm. The SBLOCA has been re-analyzed for the TDF of 93,100 gpm, and the SBLOCA PCT is well below the 2200°F limit. The affected NSSS systems and components will still meet the applicable design limits and perform their intended safety functions with the TDF of 93,100 gpm. Also, the SLB and LOCA M&E releases are still within the applicable equipment qualification limits. The SGTR doses remain within the applicable 10 CFR 100 limits, and the steam generator margin to overfill is maintained.

Summary—Parts I and II. Based on the above, TVA has determined that operation of

Watts Bar in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Therefore, operation of Watts Bar in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92.

Part 3—Addition to Core Operating Limit Report Methodologies

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

The use of $ZIRLO_{TM}$ is already permitted by TS section 4.2.1. Accordingly, the addition of the NRC approved Westinghouse COLR methodology reference is administrative in nature. Therefore, there is no increase in the probability or consequences of an accident previously evaluated.

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

Since the use of $ZIRLO_{TM}$ is already permitted by TS section 4.2.1, the addition of the NRC approved Westinghouse COLR methodology reference is administrative in nature. Accordingly, no new or different kind of accident has been created from those previously evaluated.

(3) or involve a significant reduction in a margin of safety.

The use of $ZIRLO_{TM}$ is already permitted by TS section 4.2.1. The addition of the NRC approved Westinghouse COLR methodology reference is administrative in nature. Therefore, there is no significant reduction in the margin of safety.

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

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Project Director: Frederick J. Hebdon.

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

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

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

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

Date of application for amendment: May 2, 1997.

Brief description of amendment request: The proposed amendment would change the main steam isolation valve (MSIV) closure time assumption used in the main steam line break accident analysis and referenced in the Basis for Technical Specification 4.7.

Date of individual notice in **Federal Register**: May 15, 1997 (62 FR 26829).

Expiration date of individual notice: June 16, 1997.

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

Wisconsin Electric Power Company, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: September 30, 1996, as supplemented November 26, and December 12, 1996, February 13, March 5, April 2, April 16, May 9, and June 3, 1997 (TSCR 192).

Description of amendment request: The proposed amendments would change Technical Specification requirements related to the service water system, component cooling water system, containment cooling and iodine removal systems, auxiliary electrical systems, and the control room emergency filtration system. The supplemental applications dated April 2, April 16, May 9, and June 3, 1997, would eliminate separate requirements for the component cooling water system for single-unit and two-unit operation, revise the acceptance criteria for laboratory testing of the control room emergency filtration system charcoal adsorber banks from 90 percent to 99 percent, and supplement additional information on the basis for acceptability of equipment qualification analyses and dose assessments resulting from a loss-of-coolant accident. The June 3, 1997, submittal requested the proposed amendments be handled on an exigent basis based on the current schedule which indicates that Unit 2 restart is scheduled for June 25, 1997, and Unit 1 restart is scheduled for July 1, 1997, and failure of the issuance of the amendments by these dates would result in prevention of Point Beach's resumption of operation.

Date of individual notice in the **Federal Register**: June 10, 1997 (62 FR 31636).

Expiration date of individual notice: July 10, 1997.

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

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: June 20, 1996, as supplemented by letters dated December 30, 1996, and March 5, 1997.

Brief description of amendments: The amendments would change the Technical Specifications (TS) by incorporating NRC-approved thermal limit licensing methodology in the list of approved methodologies used in establishing the fuel cycle-specific thermal limits. In addition, the proposed amendment will change the TS to reflect the use of Siemens Power Corporation (SPC) ATRIUM–9B fuel for all operating Modes at Dresden, Unit 3. The proposed amendment would also correct minor editorial items in the TS.

Date of issuance: June 12, 1997.

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

Amendment Nos.: 160 and 155.

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

Date of initial notice in **Federal Register:** April 9, 1997 (62 FR 17227). The Commission's related evaluation of the amendments is cotained in a Safety Evaluation dated June 12, 1997.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 12, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.Consolidated Edison Company of New York, Docket No. 50–247, Indian PointNuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: March 31, 1997.Brief description of amendment: The amendment revises Technical Specifications (TSs) to remove the reference of Valve 863 from TS Table 3.6–1. This revision would allow for the installation of a proposed modification for automatic closure of Valve 863 upon receipt of a Phase A containment Isolation signal.

Date of issuance: June 19, 1997.

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

Amendment No.: 193.

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

Date of initial notice in **Federal Register**: May 15, 1997 (62 FR 26823)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 19, 1997.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: March 27, 1997, as supplemented by letter dated May 6, 1997.

Brief description of amendment: The amendment changes the Technical Specification 3/4.5.2, ''ECCS Subsystems—Modes 1, 2, and 3.'' The proposed changes add a surveillance requirement to verify the Emergency Core Cooling System (ECCS) piping is full of water at least once per 31 days, and clarifies wording of surveillance requirement 4.5.2.j. The amendment also revises the TS Bases 3/4.5.2 and 3/4.5.3 to reflect surveillance requirement.

Date of issuance: June 11, 1997.

Effective date: June 11, 1997, to be implemented within 60 days.

Amendment No.: 130.

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

Date of initial notice in **Federal Register**: April 9, 1997 (62 FR 17234). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 11, 1997.

No significant hazards consideration comments received: No.

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

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida.

Date of application for amendment: December 20, 1996, and supplemented February 13, and April 17, 1997.

Brief description of amendment: This amendment modifies the Technical Specifications (TS) to delete a footnote associated with TS 2.1.1, ''Reactor Core Safety Limits'' which requires reactor thermal power to be limited to 90% of 2700 Megawatts thermal for Cycle 14 operation beyond 7000 Effective Full Power Hours.

Date of Issuance: May 16, 1997.

Effective Date: May 16, 1997.

Amendment No.: 151.

Facility Operating License No. DPR–67: Amendment revised the TS.

Date of initial notice in **Federal Register**: January 15, 1997 (62 FR 2190).

The February 13, and April 17, 1997, letters provided clarifying information that did not change the scope of the December 20, 1996, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 16, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

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

Date of application for amendments: July 28, 1995, as revised February 21, 1997.

Brief description of amendments: The amendments revise the Technical Specifications for the Prairie Island Nuclear Generating Plant to allow credit for soluble boron in spent fuel criticality analyses. The request is based on the NRC approval of the Westinghouse Owners Group generic methodology for crediting soluble boron given in Topical Report WCAP–14416–NP–A, ''Westinghouse Spent Fuel Rack Criticality Analysis Methodology,'' Revision 1, November 1996.

Date of issuance: June 12, 1997.

Effective date: June 12, 1997, with full implementation within 30 days.

Amendment Nos.: 129 and 121.

Facility Operating License Nos. DPR–42 and DPR–60. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 26, 1997 (62 FR 14464).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 12, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

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: January 13, 1997, as supplemented March 24, 1997, May 13, 1997, and May 23, 1997.

Brief description of amendment: The amendment revises Technical Specifications Requirements for containment leakage testing to add several containment isolation valves and to implement the requirements of 10 CFR Part 50, Appendix J, Option B for performance-based primary reactor containment leakage testing.

Date of issuance: June 17, 1997.

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

Amendment No.: 174.

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

Date of initial notice in **Federal Register**: March 19, 1997 (62 FR 13173).

The March 24, May 13, and May 23, 1997, supplemental letters provided clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 17, 1997.

No significant hazards consideration comments received: No.

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

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 application for amendments: January 31, 1997.

Brief description of amendments: The amendments revise Technical Specification 3/4.6.1.5, and its associated Bases section, to ensure that a representative average containment air temperature is measured.

Date of issuance: June 13, 1996.

Effective date: Both units, as of the date of issuance, to be implemented within 60 days.

Amendment Nos. 195 and 178.

Facility Operating License Nos. DPR–70 and DPR–75. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 12, 1997 (62 FR 11497).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 13, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, NJ 08079.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: August 22, 1996, as supplemented March 28, 1997.

Brief description of amendments: Revise Technical Specifications (TS) 3.6.5 and associated Bases to lower the minimum TS ice basket weight. Also extend the chemical analysis surveillance interval for the ice condenser ice bed from 12 months to 18 months.

Date of issuance: June 10, 1997.

Effective date: June 10, 1997.

Amendment Nos.: 224, 215.

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

Date of initial notice in **Federal Register**: April 23, 1997 (62 FR 19835).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 10, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

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

Date of application for amendment: April 22, 1997, as supplemented on May 15, and June 2, 1997. The April 22, 1997, submittal superseded a previous submittal on this subject dated September 6, 1996 (61 FR 53769), as supplemented on October 30, October 31, November 7, November 15, and November 27, 1996, and January 23 and January 29, 1997.

Brief description of amendment: The amendment revises TS Section 4.2.b, ''Steam Generator Tubes,'' and its associated Basis, by allowing a laser-welded repair of Westinghouse hybrid expansion joint (HEJ) sleeved steam generator tubes.

Date of issuance: June 7, 1997.

Effective date: June 7, 1997.

Amendment No.: 135.

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

Date of initial notice in **Federal Register**: May 7, 1997 (62 FR 24988).

The May 15, and June 2, 1997, submittals provided supplemental 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 June 7, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 24, 1997, as supplemented on May 15 and 28, and June 5, 1997.

Brief description of amendment: The amendment revises TS Section 4.2.b, ''Steam Generator Tubes,'' to allow repair of steam generator (SG) tubes with Combustion Engineering (CE) leak-tight sleeves in accordance with CE generic topical report CEN–629–P, Revision 2, ''Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak-Tight Sleeves.'' The TS are also revised to allow re-sleeving of tubes with existing sleeve joints in accordance with KNPP specific topical report CEN–632–P, ''Repair of Kewaunee Steam Generator Tubes Using a Re-Sleeving Technique.''

Date of issuance: June 7, 1997.

Effective date: June 7, 1997.

Amendment No.: 134.

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

Date of initial notice in **Federal Register**: May 7, 1997 (62 FR 24989).

The May 15 and 28, and June 5, 1997, submittals provided supplemental 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 June 7, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 28, 1997, as supplemented on May 19, 1997.

Brief description of amendment: The amendment establishes a new design basis flow rate for the auxiliary feedwater (AFW) pumps consistent with the assumptions used in the reanalysis of the limiting design basis event for the

AFW system. The Basis for TS 3.4.b, ''Auxiliary Feedwater System,'' has been revised to reflect the change in AFW flow and to clarify the requirements for the AFW cross-connect valves.

Date of issuance: June 7, 1997.

Effective date: June 7, 1997.

Amendment No.: 133.

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

Date of initial notice in **Federal Register**: May 7, 1997 (62 FR 24977).

The May 19, 1997, submittal 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 June 7, 1997.

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 25th day of June, 1997.

For the Nuclear Regulatory Commission.

Jack W. Roe,

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

[FR Doc. 97–17140 Filed 7–1–97; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–160]

Georgia Institute of Technology Research Reactor; Closing of Local Public Document Room

Notice is hereby given that the Nuclear Regulatory Commission (NRC) is closing the local public document room (LPDR) for records pertaining to the Georgia Institute of Technology (Georgia Tech) Research Reactor located at the Decatur Library, Decatur, Georgia, effective July 3, 1997.

This LPDR was established in April 1996 during the NRC's review of Georgia Tech's license renewal application. There is no longer a need for the LPDR since License R–97 was renewed for a 20-year term on May 30, 1997.

Dated at Rockville, Md., this 26th day of June 1997.

For the Nuclear Regulatory Commission.

Russell A Powell,

Chief, Freedom of Information/Local Public Document Room Branch, Office of Information Resources Management.

[FR Doc. 97–17297 Filed 7–1–97; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Categorizing and Transporting Low Specific Activity Materials and Surface Contaminated Objects: Availability

The Nuclear Regulatory Commission and the Department of Transportation have jointly prepared a draft report (designated NUREG–1608 and RSPA Advisory Guidance 97–005) entitled ''Categorizing and Transporting Low Specific Activity Materials and Surface Contaminated Objects.'' NRC is issuing the draft report for review and comment.

The primary purpose of this draft guidance is to assist shippers in preparing low specific activity materials (LSA) and surface contaminated objects (SCOs) for shipment in compliance with Federal regulations. The draft guidance is provided in question and answer format on the classification, characterization, packaging and transportation of LSA and SCOs, including the definition of LSA and SCOs, the determination of distribution on of activity in LSA material or on SCO surfaces, mixing LSA and SCOs in a package, radiation level measurements, and various other aspects of transporting LSA and SCOs.

NRC is particularly interested in comment regarding the use of its ''Final Branch Technical Position on Concentration Averaging and Encapsulation,'' January 17, 1995, in the draft guidance (see Questions 4.2.4, 5.1.4, and 5.2.3). Also, NRC is interested in comment on the utility to shippers of Appendix A.

Draft NUREG–1608 is available for inspection and copying for a fee at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington DC 20555–0001. A free single copy of Draft NUREG–1608, to the extent of the supply, may be requested by writing to Distribution Services, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

Submit comments on draft NUREG–1608 by (90 days after publication date). Mail comments to: Chief, Rules and Directives Branch, Mail Stop T–6 D59, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Comments may be hand-delivered to 11545 Rockville Pike, Maryland between 7:30 a.m. and 4:15 p.m. on Federal workdays.

Comments may also be submitted electronically, in either ASCII text or WordPerfect format (version 5.1 or later), by calling the NRC Electronic Bulletin Board on FEDWORLD. The bulletin board may be accessed using a personal computer, a modem, and one of the commonly available communications software packages, or directly via Internet.

If using a personal computer and modem, the NRC subsystem on FEDWORLD can be accessed directly by dialing the toll free number: 1–800–303–9672. Communication software parameters should be set as follows: parity to none, data bits to 8, and stop bits to I (N,8,I). Using ANSI terminal emulation, the NRC NUREG and Reg Guide Comments subsystem can then be accessed by selecting the ''NRC Rules Menu'' option from the ''NRC Main Menu.'' For further information about options available for NRC at FEDWORLD consult the ''Help/Information Center'' from the ''NRC Main Menu.'' Users will find the ''FEDWORLD Online User's Guides'' particularly helpful. Many NRC subsystems and databases also have a ''Help/Information Center'' option that is tailored to the particular subsystem.

The NRC subsystem on FEDWORLD can also be accessed by a direct dial phone number for the main FEDWORLD BBS: 703–321–3339; Telnet via Internet: fedworld.gov (192.239.92.3); File Transfer Protocol (FTP) via Internet: ftp.fedworld.gov (192.239.92.205); and World Wide Web using: http://www.fedworld.gov (this is the Uniform Resource Locator (URL)).

If using a method other than the toll free number to contact FEDWORLD, the NRC subsystem will be accessed from the main FEDWORLD menu by selecting the ''Regulatory, Government Administration and State Systems,'' then selecting ''Regulatory Information Mall.'' At that point, a menu will be displayed that has an option ''U.S. Nuclear Regulatory Commission'' that will take you to the NRC Online main menu. The NRC Online area can also be accessed directly by typing ''/go nrc'' at a FEDWORLD command line. If you access NRC from FEDWORLD's main menu, you may return to FEDWORLD by selecting the ''Return to FEDWORLD'' option from the NRC Online Main Menu. However, if you access NRC at FEDWORLD by using NRC's toll-free number, you will have full access to all NRC systems but you will not have access to the main FEDWORLD system.