Manager for Enrichment Facilities, Oak Ridge Operations Office, DOE, and the Director, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, NRC.

VI. Effective Date and Modification

This MOU shall become effective upon signing by the DOE Assistant Manager for Enrichment Facilities, Oak Ridge Operations, and the Director, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, NRC, and will be subject to periodic reviews and may be amended or modified upon written agreement by the parties. This MOU may be terminated by mutual agreement or by written notice of either party submitted six months in advance of termination.

VII. Separability

If any provision(s) of this MOU, or the application of any provision(s) to any person or circumstances, is held invalid, the remainder of this MOU and the application of such provision(s) to other persons or circumstances shall not be affected.

For the Nuclear Regulatory Commission.

Dated: October 27, 1997.

Elizabeth Q. Ten Eyck,

Director Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission.

For the Department of Energy.

Dated: October 28, 1997.

Joseph W. Parks,

Assistant Manager for Enrichment Facilities, Oak Ridge Operations Office, Department of Energy.

[FR Doc. 97–29244 Filed 11–4–97; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

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 October 10, 1997, through October 24, 1997. The last biweekly notice was published on October 22, 1997 (62 FR 54866).

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

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

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

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of 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, MD 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 December 5, 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: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

Baltimore Gas and Electric Company, Docket No. 50–317, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, MD

Date of amendment request: October 2, 1997.

Description of amendment request: The amendment request would change the Technical Specifications to identify a proposed upgrade of the electrical capacity of the No. 1B emergency diesel generator.

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

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

The Engineered Safety Features (ESF) electrical system provides a reliable source of electrical power to the 4.16 kV ESF busses to operate the necessary accident mitigation equipment, should offsite power be lost. The proposed change to the Technical Specifications was prompted by the upgrade of the electrical and mechanical capacity of the No. 1B Fairbanks Morse Emergency Diesel Generator (EDG). The increased electrical capacity of the No. 1B Fairbanks Morse EDG will give the operators greater flexibility in the choice of discretionary loads for the mitigation of accidents. This modification necessitates changes to the Technical Specifications.

The ESF electrical system, including the four EDGs, is used to mitigate the consequences of an accident. The modification to upgrade the capacity of No. 1B EDG will increase the electrical output of the EDG, but will not change the configuration of the ESF electrical system or any support systems such that the EDGs would become an accident initiator. Therefore, the proposed change would not increase the probability of an accident previously evaluated.

The proposed Technical Specifications will continue to demonstrate the reliability and capability of the upgraded No. 1B EDG to perform its accident mitigation function. The proposed changes to the surveillance requirements do not alter the intent or performance of the surveillance. Only the electrical loadings changed, reflecting the change in the EDG's electrical capacity. Implementation of the proposed Technical Specifications will not reduce the ability of No. 1B EDG to perform its safety functions. Any auxiliary systems that required modification or analysis to support the upgraded ratings of the 1B Fairbanks Morse EDG have been determined not to adversely impact operation of any other plant systems necessary to mitigate the consequences of an accident. Therefore, the proposed change would not increase the consequences of an accident previously evaluated.

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

2. Would not create the possibility of a new or different type of accident from any accident previously evaluated.

The proposed change increases the electrical loading for surveillance requirements to reflect the upgrade to the electrical capacity of the No. 1B Fairbanks Morse EDG. This change does not add any new equipment, modify any interfaces with any existing equipment, change the equipment's function, or the method of operating the equipment to be modified. The system will continue to operate in the same manner as before the capacity upgrades were implemented. The modified No. 1B EDG will continue to function as an accident mitigator, and will not become an initiator of any accident.

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

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

The safety function of the EDG is to provide a reliable source of electrical power to the ESF electrical system sufficient to power the necessary accident mitigation equipment, should offsite power be lost. This safety function is demonstrated by performing the required surveillance tests. The proposed changes do not alter the intent or method of performance of any of the surveillance tests.

The proposed change to the Technical Specifications was prompted by the upgrade of the electrical and mechanical capacity of the No. 1B Fairbanks Morse EDG. The higher electrical capacity will result in an increase in the margin between No. 1B EDG's electrical capacities and the electrical power required to operate safety-related equipment required for safe shutdown or accident mitigation. The increased electrical capacity results in the need to increase the electrical loadings used in the surveillance tests. The changes in the surveillance tests will continue to ensure that the EDG is tested appropriately and will continue to perform its safety function. In addition, it should be noted that upgrades on identical Fairbanks Morse EDGs have already been performed on Unit 2 and have resulted in identical changes to the Unit 2 Technical Specifications. Because of the increased electrical margin afforded by the upgraded EDG, these modifications may be considered an increase in the margin of safety.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Calvert County Library, Prince Frederick, MD 20678.

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

NRC Project Director: S. Singh Bajwa, Director.

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, IL; Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, IL

Date of amendment request: September 8, 1997.

Description of amendment request: The proposed amendment would revise Byron and Braidwood Technical Specification (TS) 4.5.2.b and associated bases as they relate to the requirement to vent the Emergency Core Cooling System (ECCS) pump casings and discharge piping high points outside containment. The change will revise the Unit 1 requirement for ultrasonic examinations every 31 days to also include ultrasonic examination of the piping at the 1CV206 valve for Byron (1CV207 valve for Braidwood) if the 1B Chemical and Volume Control (CV) pump is idle. These changes are required to align the surveillance requirements for Unit 1 with those of Unit 2. In addition, the condition that the Unit 1 requirements will be applicable only until the end of the current cycle is deleted consistent with the Unit 2 requirements. With these changes there will no longer be the need to maintain separate pages for Unit 1 and Unit 2 requirements.

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

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

The proposed changes will align the surveillance requirements for both Units 1 and 2 with the installed system design and normal operating conditions. No increase in the probability of an accident will occur as a result of this change. The conduct of surveillances required by the Technical Specifications is not postulated to initiate an accident. The level of surveillance performed to date has provided confidence that the objective of the current surveillance requirement has been met. As such, the proposed change does not result in a significant increase in the probability of occurrence of a previously analyzed accident.

The consequences of a previously analyzed accident are not increased. Operating experience has shown that the level of surveillance performed to date is sufficient to provide confidence that no significant voiding has occurred in the affected piping. Ultrasonic examinations have confirmed the water solid condition of the piping. Although voiding is not expected, evaluation of postulated voided conditions confirm that unacceptable dynamic loading would not occur, and, therefore, the integrity of the ECCS piping is not compromised. Thus, the ECCS will be capable of performing its design function of cooling the reactor core and providing shutdown capability following initiation of the certain accidents. This will ensure that the consequences of a previously analyzed accident are not significantly increased.

Therefore, these proposed revisions do not result in a significant increase in the probability or consequences of an accident previously analyzed.

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

The proposed changes do not create the possibility of a new or different kind of accident. ComEd has evaluated the piping configuration for the ECCS discharge piping of the ECCS subsystems. A specific engineering evaluation of both a voided 2-inch and 8-inch RH [Residual Heat Removal] line was performed. This evaluation concluded that the piping can withstand the dynamic loads caused by the maximum credible air void. Due to the higher-pressure rating and smaller size of the SI [Safety Injection] and CV discharge piping, this evaluation is considered bounding for the ECCS subsystems. The results of the evaluation were submitted for staff review in a letter dated March 12, 1990, in support of Amendments 47 and 36 to the Operating Licenses for Byron and Braidwood, respectively. The proposed changes will not result in new failure modes because no new equipment is installed, and installed equipment is not operated in a new or different manner. Manual venting operations have been performed as permitted by system operation and piping configuration. This venting surveillance does not apply to subsystems in communication with operating systems because the flows and/or pressures prevalent in these systems are sufficient to provide confidence that water hammer which could occur from voiding would not result in unacceptable dynamic loads from water hammer will not occur. Accordingly, this change will not create the possibility of a new or different kind of accident.

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

The margin of safety is not significantly reduced because the proposed change will provide sufficient assurance that excessive voiding will not occur. This will assure proper system functioning. Venting of the idle subsystems, in conjunction with the operating conditions of the subsystems in operation, provides confidence that voiding is not present. This has been confirmed by the performance of ultrasonic examinations of the piping of interest. This meets the objective of the surveillance requirement and thus preserves the margin of safety.

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

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, IL 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, IL 60481.

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: January 18, 1996, as revised October 1, 1997.

Description of amendment request: The original proposed amendment (January 18, 1996) would have deleted the requirement in Section 6.5.6 of the Technical Specifications (TS) to perform inservice inspections of the primary coolant pump (PCP) flywheels. The October 1, 1997, submittal would revise Section 6.5.6 of the TS to lengthen the flywheel inspection period to 10 years rather than delete it entirely. The note added by Amendment 175 for the deletion of the inspection at the end of Cycle 12 would also be deleted. The original submittal was previously noticed in the **Federal Register** on September 11, 1996 (61 FR 47976).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of no significant hazards consideration in its original submittal. In its revised submittal the licensee stated that the conclusions reached in the original no significant hazards consideration determination were still valid because the revised submittal just reduces the frequency of the test as opposed to deleting it. The original no significant hazards consideration discussion is presented below:

The following evaluation supports the finding that operation of the facility in accordance with the proposed change to the Technical Specifications would not:

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

The proposed change to the Technical Specifications would delete the requirement to perform non-destructive examination of the upper flywheel on the PCPs. The fracture mechanics analyses conducted to support the change show that a preexisting crack sized just below detection level will not grow to the flaw size necessary to result in flywheel failure within the life of the plant. This analysis conservatively assumes minimum material properties, maximum flywheel accident speed, location of the flaw in the highest stress area and a number of startup/shutdown cycles eight times greater than expected. Since an existing flaw in the flywheel will not grow to the allowable flaw size under normal operating conditions or to the critical flaw size under LOCA [loss-of-coolant accident] conditions over the life of the plant, elimination of inservice inspection for such cracks during the plant's life will not involve a significant increase in the probability of an accident previously considered.

The proposed changes do not increase the amount of radioactive material available for release or modify any systems used for mitigation of such releases during accident conditions. Therefore, operation of the facility in accordance with the proposed change to the Technical Specifications would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to the Technical Specifications would not change the design, configuration, or method of operation of the plant and therefore, operation of the facility in accordance with the proposed change to the Technical Specifications would not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change to the Technical Specifications would not result in a significant reduction in the margin of safety. Significant conservatisms have been used for calculating the allowable flaw size, critical flaw size and crack growth rate in the PCP flywheels. These include minimum material properties, maximum flywheel accident speed, location of the postulated flaw in highest stress area and a number of startup/shutdown cycles eight times greater than expected. Since an existing flaw in the flywheel will not grow to the maximum allowable flaw size under normal operating conditions or to the critical flaw size under LOCA conditions over the life of the plant, elimination of inservice inspections for such cracks during the plant's life will 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. In addition, the staff agrees that this analysis bounds the conditions in the revised submittal. The editorial change to delete an obsolete note has no effect on plant operation or safety and also satisfies the three standards of 10 CFR 50.92(c). Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: Judd L. Bacon, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, MI 49201.

NRC Project Director: John N. Hannon.

GPU Nuclear Corporation, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, NJ

Date of amendment request: October 10, 1997.

Description of amendment request: The proposed change (TSCR 253) would reflect the registered trade name of ''GPU Nuclear'' in the operating license for the Oyster Creek Nuclear Generating Station (OCNGS) and change the legal name of the operator of OCNGS from GPU Nuclear Corporation to GPU Nuclear, Inc. In addition, two minor editorial corrections are included.

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

Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The proposed amendment adds to the license and the technical specifications the trade name of the

Owner of Oyster Creek. The change in the legal name of the operator of Oyster Creek is an administrative change made to reflect the name changes made throughout the GPU family of companies. The name change has no impact on plant design or operation.

Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated because no new failure modes are created by the proposed changes. The use of a trade name for the Owner of Oyster Creek and the change in the legal name of the operator of Oyster Creek has no impact on plant design or operation. Thus, there is no creation of the possibility of a new or different kind of accident from those previously evaluated.

Operation of the facility in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. The proposed amendment does not change any operating limits for reactor operation.

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. In addition, the staff has reviewed the licensee's proposed editorial changes and determined that they do not effect the conclusions of the analysis. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

Attorney for licensee: Ernest L. Blake, Jr., Esquire. Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Ronald B. Eaton, Acting Director.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 1, Oswego County, NY

Date of amendment request: October 21, 1997. This notice supersedes a previous notice, (62 FR 30625), published June 4, 1997, which was based upon the licensee's application for amendment dated May 16, 1997. The licensee's application dated October 21, 1997, supersedes the May 16, 1997, submittal in its entirety.

Description of amendment request: The proposed amendment would change the administrative section of the Technical Specifications (TS) regarding the Operations organization. Specifically, TS 6.2.2i currently states that ''The Manager Operations, Station Shift Supervisor Nuclear and Assistant Station Shift Supervisor Nuclear shall hold senior reactor operator licenses.'' This would be changed to state ''As a minimum, either the Manager Operations or the General Supervisor Operations shall hold a senior reactor operator license. The Station Shift Supervisor Nuclear and Assistant Station Shift Supervisor Nuclear shall hold senior reactor operator licenses.'' In addition TS 6.3.1 would be revised to indicate an additional exception to the operating staff's qualification requirements set forth in American National Standard Institute (ANSI) N18.1–1971, ''Selection and Training of Nuclear Power Plant Personnel.'' Specifically, this change would require that the Manager Operation, in lieu of meeting the senior reactor operator (SRO) requirements of ANSI N18.1–1971, shall (1) hold an SRO license at the time of appointment, or (2) have held an SRO license at Nine Mile Point Nuclear Station Unit 1 or a similar unit, or (3) have been certified for equivalent SRO knowledge.

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

1. The operation of Nine Mile Point Unit 1 [NMP1], in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The addition of the position of GSO and the requirement for either the GSO or the Manager Operations to have an SRO license is a restructuring of the Operations department. The proposed changes are administrative changes that provide additional Operations management oversight capabilities. Additional restrictions placed on the Manager Operations minimum qualification requirements for experience and SRO level knowledge for the resulting organization meet the intent of ANSI N18.1–1971 and SRP [Standard Review Plan, NUREG–0800] 13.1.1–13.1.3. No physical modification of the plant is involved and no changes to the methods in which plant systems are operated are required.

None of the precursors of previously evaluated accidents are affected, and no new failure modes are introduced. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The addition of the position of GSO and the requirement for either the GSO or the Manager Operations to have an SRO license is a restructuring of the Operations department. The proposed changes are administrative changes that provide additional Operations management oversight capabilities. Additional restrictions placed on the Manager Operations minimum qualification requirements for experience and SRO level knowledge ensure the resulting organization meets the intent of ANSI N18.1–1971 and SRP 13.1.1–13.1.3. No physical modification of the plant is involved and no changes to the methods in which plant systems are operated are required. As such, the change does not introduce any new failure modes or conditions that may create a new or different accident. Therefore, this change does not itself create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The addition of the position of GSO and the requirement for either the GSO or the Manager Operations to have an SRO license is a restructuring of the Operations department. The proposed changes are administrative changes that provide additional Operations management oversight capabilities. Additional restrictions placed on the Manager Operations minimum qualification requirements for experience and SRO level knowledge ensure the resulting organization meets the intent of ANSI N18.1–1971 and SRP 13.1.1–13.1.3. No physical modification of the plant is involved and no changes to the methods in which plant systems are operated are required. As such, this change does not in itself adversely affect any physical barrier to the release of radiation to plant personnel or to the public. Therefore, the change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 400 L Street, NW., Washington, DC 20005–3502.

NRC Project Director: S. Singh Bajwa.

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

Date of amendment request: October 7, 1997.

Description of amendment request: Technical Specifications 4.6.1.1, 3/4.6.1.2, and 3/4.6.1.3 require the testing of the containment to verify leakage limits at a specified test pressure. The proposed amendment would (1) modify the list of valves that can be opened in Modes 1 through 4, (2) remove a footnote on Type A testing, and (3) make editorial changes to the Technical Specifications and associated Bases sections.

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

NNECO has reviewed the proposed revision in accordance with 10 CFR 50.92 and has concluded that the revision does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not satisfied. The proposed revision does not involve [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 4.6.1.1 deletes valves from the list of containment isolation valves that may be opened under administrative control. Deleting the valves, which means that they are not allowed to be opened under the Limiting Condition of Operation, [cannot] cause an accident. The valves being added in the steam lines to the steam-driven auxiliary feedwater pump can be used to heat the steam lines prior to testing the steam-driven auxiliary feed water pump. Heating the steam lines prior to testing the steam-driven auxiliary feedwater pump does not increase the likelihood of a steam line break.

The administrative change of replacing the ''-'' with an ''*'' in the valve designation can neither cause [an] accident nor affect the consequences of any accident.

The addition of the RHR [residual heat removal] system containment isolation valves reflects the fact that these valves can be opened during Mode 4 to allow plant heatup and cooldown. Plant heatup and cooldown, in accordance with normal plant operation and the Technical Specifications, does not increase the likelihood of the above accidents.

The administrative controls include the appropriate considerations that containment integrity will be established, when required. By establishing containment integrity, the assumptions in the design basis analyses are assured. This means that for LOCA [loss-of-coolant accident], steam line break and feed line break accidents inside containment, there is no effect on their consequences.

Valves in the steam lines to the steam-driven auxiliary feedwater pump are being added to the list of valves allowed to be opened under administrative control. This means that these could be open at the initiation of an accident. The administrative controls under which these valves are opened provides assurance that containment integrity will be established, when required. Similarly, for an SGTR [steam generator tube rupture], Locked Rotor or Control Rod Ejection event, the administrative controls provides assurance that these valves will be closed and, therefore, allowing them to be opened will not adversely impact the consequences of these events. If failure to close is postulated as a single failure for these events, the results would be bounded by the analyses described in the FSAR [final safety analysis report]. For example, the Locked Rotor accident assumes a stuck open steam generator power-operated pressure relief valve (SG PORV). The steam released by the assumed single failure of the SG PORV, for the twenty minutes until the valve is isolated, would exceed the expected releases as a result of failure to close valve 3MSS*V885, 3MSS*V886, or 3MSS*V887, which are in ¼ inch lines. Therefore, allowing these valves to be opened under administrative control does not effect the consequences of the previously evaluated accidents.

The FSAR, Section 15.1.5, provides the assumptions on steam releases for the consequences of the steam line break accident. The steam generator with the broken steam line is assumed to be open to the atmosphere for the duration of the event and, therefore, these valves being open would not impact that assumption. For the unaffected steam generators, steam is assumed released to the atmosphere to remove decay heat. These valves are in ¼ inch lines which means that any steam released via this path would only be a small fraction of decay heat and will not adversely affect control of decay heat removal. Therefore, whether these valves are open or not will not affect the consequences of a steam line break outside containment.

Allowing the RHR system containment isolation valves to be open, under administrative control in Mode 4, does not change the way the system is operated. This proposed change to the footnote does not change the operators response to an accident in Mode 4. Therefore, the addition of these valves does not affect the consequences of the previously evaluated accidents.

The proposed change to Technical Specification Surveillance 4.6.1.2.a will delete footnote ''*'' which referred to an exemption granted by the NRC to permit the Type A test to be delayed until RFO6 [refueling outage 6]. However, the current extended shutdown has significantly delayed RFO6 and NNECO intends to perform the Type A test during this midcycle shutdown. The deletion of the footnote does not alter the operation of any system or the containment or containment airlocks, as assumed for accident analyses.

Additionally, Technical Specifications 4.6.1.1, 3/4.6.1.2, and 3/4.6.1.3, and Bases Sections 3/4.6.1.1, 3/4.6.1.2, and 3/4.6.1.3 are reworded to provide clarity and consistency. These proposed changes do not alter the operation of any system or the containment or containment airlocks during accident analyses.

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.

The proposed changes to Technical Specifications 4.6.1.1, 3/4.6.1.2, and 3/4.6.1.3 and Bases Sections 3/4.6.1.1, 3/4.6.1.2, and 3/4.6.1.3 do not alter the operation of any system or the containment or containment airlocks, during normal operation or as assumed in accident analyses.

Deleting containment isolation valves from the list of those that are allowed to be opened under administrative control can not modify plant response to an accident. Adding administrative control when the RHR system containment isolation valves are opened in Mode 4 for normal plant cooldown and heatup can not create a new or different accident. Allowing valves to be opened to heat the steam lines to the steam-driven auxiliary feedwater pump prior to testing does not create the possibility of a new or different accident. The administrative change to the valve designation can not modify plant response.

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 to Technical Specifications 4.6.1.1, 3/4.6.1.2, and 3/4.6.1.3, and Bases Sections 3/4.6.1.1, 3/4.6.1.2, and 3/4.6.1.3 do not alter the design, maintenance or function of any system or the containment or the containment airlocks. Additionally, the proposed changes do not alter the testing of any system or the containment or containment airlocks, or alter any assumption used in the accident analyses.

The considerations associated with administrative control are being added to the bases of the technical specification. These considerations are identical to those provided in GL 91–08 [Generic Letter 91–08]. This means that the changes will maintain the margin of safety. The valves that are allowed to be open in the steam lines to the steam-driven auxiliary feedwater [pump] do not impact the accident analyses and therefore do not reduce the margin of safety. The addition of the RHR system containment isolation valves reflects the fact that these valves are opened for heatup and cooldown in Mode 4. The change adds the requirements of administrative controls to these RHR system valves in Mode 4, but does not modify the use of these valves. The administrative change to the valve designation can not affect the margin of safety.

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, CT, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT.

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

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, CT

Date of amendment request: October 15, 1997.

Description of amendment request: Technical Specification Surveillances 4.1.2.3.1, 4.1.2.4.1, 4.5.2, 4.6.2.1, and 4.6.2.2 require the recirculation spray, quench spray, residual heat removal, centrifugal charging, and safety injection pumps to be tested on a periodic basis and after modifications that alter subsystem flow characteristics. The proposed changes to these surveillances would include replacing the specific surveillance pump pressure with a statement that the test be conducted in accordance with Specification 4.0.5, Inservice Testing Program. The proposed changes would also include a decrease in the required individual safety injection and centrifugal charging pump injection line flow rates, an increase in the allowed individual safety injection pump runout flow rate, and editorial changes to the surveillances.

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

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

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

The Technical Specification changes transfer control of the pump developed head requirements for the Centrifugal Charging, Safety Injection, Quench Spray, Residual Heat Removal, and Recirculation Spray pumps from the Technical Specifications to the Inservice Test program. The acceptance criteria will still assure that the safety analysis assumptions are valid. The Technical Specification changes reduce the minimum flow requirements for the Charging and Safety Injection pumps and increase the maximum allowed flow for the Safety Injection pumps. Modifying the surveillance requirements [cannot] cause an accident and, therefore, [cannot] increase the probability of an accident. The revised minimum required flows are consistent with the flows used in the accident analyses and, therefore, the change [cannot] increase the consequences of any accident. The safety injection pumps are disabled such that they [cannot] be a source of mass addition to the RCS [reactor coolant system] whenever the cold overpressure system is required to be operable. Therefore, the increase in the allowed maximum safety injection pump flow has no effect on the cold overpressure accident analysis.

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.

The proposed changes transfer control of the pump developed head requirements from the Technical Specifications to the Inservice Test program and modify the required flow surveillance values. The surveillance values that are used in the Inservice Test program and the Technical Specification are consistent with the accident analysis. The increase in the allowed maximum safety injection pump flow does not impact the cold overpressure accident analysis. The changes do not involve any changes to the way that the pumps are operated. The pumps will be used post-accident the same way as they are used prior to the change.

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 control of the pump developed head acceptance criteria is being transferred from the Technical Specification to the Inservice Test program. The acceptance criteria, at a minimum, will assure that the design basis analyses are valid. The minimum pump flow surveillance requirements in Specification 4.5.2.h are consistent with the assumptions of the accident analysis. The maximum allowed Safety Injection flow does not exceed the vendor recommendation for maximum continuous runout flow. The NPSH [net positive suction head] available to the pumps during both the injection and recirculation phases post-accident exceeds the NPSH required at the higher allowed flow. Also, the safety injection pumps are disabled so that they [cannot] be an injection source when the cold overpressure system is required to be operable which means that the increase in maximum flow does not affect the cold overpressure accident analysis. Restricting orifices are being installed in the injection lines from the safety injection and charging pumps to the Reactor Coolant System as required. The restricting orifices and the changes to the required flows will allow for resetting the throttle position of the existing throttle valves. The sizing of the restricting orifices and the associated re-throttling of the throttle valves will be in accordance with Regulatory Guide 1.82. The proposed changes allow for the setting of the throttle valve positions so that the openings will be larger than the sump screen mesh opening size while assuring that the design basis flow values are valid.

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, CT, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT.

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.

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

Date of amendment request: July 25, 1997.

Description of amendment request: The proposed amendment request would revise the Technical Specifications (TS) to implement 10 CFR Part 50 Appendix J, Option B by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leakage-Test Program,'' with certain exceptions detailed in the licensee's application. This revision supersedes the staff's description of amendment request that was published on October 8, 1997 (62 FR 52586).

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

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

The proposed change implements Option B of 10 CFR Part 50 Appendix J on performance-based containment leakage testing. The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any parameters or conditions that contribute to the initiation of any accidents previously evaluated. The proposed change potentially affects the leak-tight integrity of the containment structure designed to mitigate the consequences of a Loss-of-Coolant Accident (LOCA). The function of the containment is to maintain functional integrity during and following the peak transient pressures and temperatures and limit fission product leakage following the design basis LOCA. Because the proposed change does not alter the plant design, only the frequency of measuring Type A, B, and C leakage, the proposed change does not directly result in an increase in containment leakage.

Test intervals will be established based on the performance history of components being tested. The frequency of monitoring the relatively few containment isolation valves and/or containment penetrations subject to above normal leakage will not decrease by implementing Option B of Appendix J. A performance based program will identify those valves and penetrations which must continue to be tested each refueling outage.

The risk resulting from the proposed changes is characterized as follows, based primarily on the results contained in NUREG–1493 ''Performance-Based Containment Leakage Test Program,'' the principal Technical Support Document used by the NRC as the basis for the Appendix J Final Rule:

Type A Testing

NUREG–1493 found that the effect of containment leakage on overall accident risk is minimal since risk is dominated by accident sequences that result in failure or bypass of the containment. Industry wide, Integrated Leak Rate Tests (ILRTs) have only found a small fraction of the leaks that exceed current acceptance criteria. Only three percent of all leaks are detectable only by ILRTs, and therefore, by extending the Type A testing intervals, only three percent of all leaks have a potential for remaining undetected for longer periods of time. In addition, when leakage has been detected by ILRTs, the leakage rate has been only marginally above existing requirements. The Fort Calhoun Station Unit No. 1 Type A testing confirms the industry-wide experience that a majority of the leakage experienced during Type A testing is through components tested by Type B and C tests.

NUREG–1493 found that these observations, together with the insensitivity of reactor accident risk to the containment leakage rate, show that increasing the Type A leakage test intervals would have a minimal impact on public risk.

Type B and C Testing

NUREG–1493 found that while Type B and C tests can identify the vast majority (greater than 95 percent) of all potential leakage paths, performance-based alternatives to current local leakage-testing requirements are feasible without significant risk impacts. The risk model used in NUREG–1493 suggests that the number of components tested would be reduced by about 60 percent with less than a three-fold increase in the incremental risk due to containment leakage. Since, under existing requirements, leakage contributes less than 0.1 percent of overall accident risk, the overall impact is very small. In addition, the NRC's Final Regulatory Impact Analysis concluded that while the extended testing intervals for Type B and C tests led to minor increases in potential offsite dose consequences, the beneficial expected decrease in onsite worker dose received during ILRT and local leak rate testing exceeds (by at least an order of magnitude) the potential off-site dose consequences.

Therefore, the proposed change will not result in a significant increase in the probability or consequences of any accident previously evaluated.

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

There will be no physical alterations to the plant configuration, changes to setpoint values, or changes to the implementation of setpoints or limits as a result of this proposed change. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents.

This change involves the reduction of Type A, B, and C test frequency. Except for the method of defining the test frequency, the methods for performing the actual tests are not changed. No new accident modes are created by extending the testing intervals. No safety-related equipment or safety functions are altered as a result of this change. Extending the test frequency has no influence on, nor does it contribute to, the possibility of a new or different kind of accident or malfunction from those previously analyzed. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change only affects the frequency of Type A, B, and C testing. Except for the method of defining the test frequency, the methods for performing the actual tests are not changed.

The frequency of monitoring the relatively few containment isolation valves and/or containment penetrations subject to above normal leakage will not decrease by implementing Option B of Appendix J. A performance based program will identify those valves and penetrations which must continue to be tested each refueling outage. NUREG–1493 has determined that, under several different accident scenarios, the increased risk of radioactivity release from containment is negligible with the implementation of these proposed changes.

The margin of safety that has the potential of being impacted by the proposed change involves the offsite dose consequences of postulated accidents which are directly related to containment leakage rate. The containment isolation system is designed to limit leakage to La, which is stated in the Fort Calhoun Station Unit No. 1 Technical Specifications to be 0.1 percent by weight of the containment air per 24 hours at 60 psig.

The limitation on containment leakage rate is designed to ensure that total leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure. The margin to safety for the offsite dose consequences of postulated accidents directly related to the containment leakage rate is maintained by meeting the 1.0 La acceptance criteria. The La value is not being modified by this proposed change.

Except for the method of defining the test frequency, no change in the method of testing is being proposed. The Type B and C tests will continue to be done at 60 psig or greater. Other programs are in place to ensure that proper maintenance and repairs are performed during the service life of the primary containment and systems and components penetrating the primary containment.

Therefore, the proposed change will not result in a significant reduction in a margin of safety.

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

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

Attorney for licensee: Perry D. Robinson, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: William H. Bateman.

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

Date of amendment request: September 3, 1997.

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) to revise the number of hours operating personnel can work in a normal shift. The proposed amendment also contains some administrative changes to the TSs.

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

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

A. Establishing operating personnel work hours at, ''an 8 to 12 hour day, nominal 40 hour week,'' allows normal plant operations to be managed more effectively and does not adversely effect performance of operating personnel. Overtime remains controlled by site administrative procedures in accordance with NRC Policy Statement on working hours (Generic Letter 82–12). If 8 hour shifts are maintained in part or whole, then acceptable levels of performance from operating personnel is assured through effective control of shift turnovers and plant activities. No physical plant modifications are involved and none of the precursors of previously evaluated accidents are affected. Therefore, this change will not involve a significant increase in the probability or consequence of an accident previously evaluated.

B. Editorial changes clarify section 6.2.2.g without changing the intent or meaning. The proposed change meets the intent of the NRC Policy Statement on working hours (Generic Letter 82–12).

C. Changes to sections 3.10.6.1.a and 3.10.9 do not change the intent or meaning of the technical specification sections. Clarification to the table notation in section 4.1 related to the definition of shift checks to monitor plant conditions will continue as intended but are allowed to increase up to at least once per 12 hours. This increase is consistent with standard industry practice as represented by the Standard Technical Specifications (STS), Reference 1.

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

A. Establishing operating personnel work hours at, ''an 8 to 12 hour day, nominal 40 hour week,'' allows normal plant operations to be managed more effectively and does not adversely effect performance of operating personnel. If 8 hour shifts are maintained in part or whole, then acceptable levels of performance from operating personnel is assured through effective control of shift turnovers and plant activities. Overtime remains controlled by site administrative procedures in accordance with the NRC Policy Statement on working hours (Generic Letter 82–12). No physical modification of the plant is involved. As such, the change does not introduce any new failure modes or conditions that may create a new or different accident. Therefore, operation in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

B. Editorial changes clarify section 6.2.2.g without changing the intent or meaning. The proposed change meets the intent of the NRC Policy Statement on working hours (Generic Letter 82–12).

C. Changes to sections 3.10.6.1.a and 3.10.9 do not change the intent or meaning of the technical specification sections. Clarification to the table notation in section 4.1 related to the definition of shift checks to monitor plant conditions will continue as intended but are allowed to increase up to at least once per 12 hours. This increase is consistent with standard industry practice as represented by the Standard Technical Specifications (STS), Reference 1.

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

A. Establishing operating personnel work hours at, ''an 8 to 12 hour day, nominal 40 hour week,'' allows normal plant operations to be managed more effectively and does not adversely effect performance of operating personnel. If 8 hour shifts are maintained in part or whole, then acceptable levels of performance from operating personnel is assured through effective control of shift turnovers and plant activities. Overtime remains controlled by site administrative procedures in accordance with the NRC Policy Statement on working hours (Generic Letter 82–12) and is consistent with the Standard Technical Specifications. The proposed change involves no physical modification of the plant, or alterations to any accident or transient analysis. There is no Basis to section 6 of the Technical Specifications, and the changes are administrative in nature. Therefore, the change does not involve any significant reduction in a margin of safety.

B. Editorial changes clarify section 6.2.2.g without changing the intent or meaning. The proposed change meets the intent of the NRC Policy Statement on working hours (Generic Letter 82–12).

C. Changes to sections 3.10.6.1.a and 3.10.9 do not change the intent or meaning of the technical specification sections. Clarification to the table notation in section 4.1 related to the definition of shift checks to monitor plant conditions will continue as intended but are allowed to increase up to at least once per 12 hours. This increase is consistent with standard industry practice as represented by the Standard Technical Specifications (STS), Reference 1.

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

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

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

NRC Project Director: S. Singh Bajwa, Director.

Rochester Gas and Electric Corporation, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, NY

Date of amendment request: September 29, 1997, as supplemented October 8, 1997. The September 29 application and October 8, 1997, supplement supersede the September 13, 1996, application and its April 24, 1997, supplement. This notice supersedes the notice published on October 9, 1996 (61 FR 197) in its entirety.

Description of amendment request: The proposed amendment would change the Ginna Station Technical Specifications (TSs) which would allow referencing of revision of the Ginna Station pressure and temperature limits report (PTLR) for the reactor coolant system (RCS) pressure and temperature (P/T) limits and low temperature overpressure protection (LTOP) limits. The proposed amendment would correct some typographical errors in the TSs.

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. Operation of Ginna Station 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 revise Administrative Controls Section 5.6.6.c to update the reference to the NRC's approval of the first use of the PTLR methodology, update the RCS P/T methodology to the final NRC approved version, allow use of ASME Code Case N–514 for LTOP enable temperature methodology, and to correct a typographical error. These changes complete implementation of Generic Letter 96–03 by referencing NRC approved methodology within the Administrative Controls. The updated RCS P/T methodology has been generically approved by the NRC while the use of ASME Code Case N–514 for LTOP enable temperature methodology was previously approved for use at Ginna Station by the NRC. As such, these changes are administrative in nature and do not impact initiators or analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed changes will not impose any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes will not reduce a margin of plant safety because the methodology have been shown to ensure that the P/T and LTOP limits in the PTLR continue to meet all necessary requirements for reactor vessel integrity. These changes are administrative in nature since the limits were previously relocated to the PTLR under a separate LAR [License Amendment Request]. As such, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room Location: Rochester Public Library, 115 South Avenue, Rochester, NY 14610.

Attorney for licensee: Nicholas S. Reynolds, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Project Director: S. Singh Bajwa, Director.

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

Date of amendment requests: December 22, 1995.

Description of amendment requests: The licensee proposes to delete the physical protection program reporting requirement from License Condition 2.G, and to clarify in License Condition 2.E that all the documents composing the physical protection program plans may not contain safeguards information.

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.

This proposed change is considered an administrative change. It has no impact on the probability or consequences of any of the accidents previously evaluated. This change revises license conditions for clarification and removes the burden of duplicate reporting requirements. This change does not affect the physical protection program as previously approved by the Nuclear Regulatory Commission (NRC). License Condition 2.E is being revised to clarify that the physical security, security force training and qualification, and safeguards contingency plans may or may not contain safeguards information. The security force training and qualification plan does not currently contain safeguards information.

A reporting requirement in License Condition 2.G is being revised to remove the reference to License Condition 2.E for the physical protection program. The reporting requirements for the physical protection program are located in the regulations, 10 CFR 73.71 and 10 CFR 73 part, Appendix G.

Therefore, the probability and consequences of an accident previously evaluated are not affected by these proposed changes.

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

This proposed change is considered an administrative change. It has no impact on equipment, systems, or structures such that a new or different kind of accident is created. This change revises license conditions to clarify that safeguards information may be located in the physical protection program plans and to remove duplicate and unnecessary reporting requirements for the physical protection program. There is no change associated with the implementation and maintenance of the physical protection program as previously approved by the NRC.

Therefore, the possibility of a new or different kind of accident from an accident previously evaluated is not created.

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

This proposed change is considered an administrative change only. It has no impact on the margin of safety associated with the physical protection program. This change revises license conditions to clarify the location of safeguards information in the physical protection program plans and remove duplicative and unnecessary reporting requirements for the physical protection program. The maintenance and implementation of the physical protection program is not affected by this change.

Therefore, there will not be a significant reduction in a margin of safety.

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

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

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

NRC Project Director: William H. Bateman.

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, OH

Date of amendment request: October 22, 1997.

Description of amendment request: The amendment would change the Perry Nuclear Power Plant design basis as described in the Updated Safety Analysis Report. The change will add a description of the temperature control valves and associated bypass lines around the Emergency Closed Cooling System heat exchangers. These features are designed to ensure operability of the Control Complex Chilled Water System under post-accident load conditions, without the need for compensatory actions.

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

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

The proposed amendment is requesting Nuclear Regulatory Commission (NRC) review and approval of changes to the Perry Nuclear Power Plant (PNPP) Updated Safety Analysis Report (USAR) to incorporate descriptions (in the form of text, tables and drawings) of a modification to the plant involving two temperature control valves and associated temperature elements, and piping segments that have been installed in the Emergency Closed Cooling Water (ECC) System. These valves, temperature elements, and piping segments were installed to increase the overall reliability of the ECC System and the other safety related plant systems that it serves, to help ensure that they perform their specified safety functions without reliance on manual throttling actions.

The probability of occurrence and the consequences of an accident previously evaluated in the USAR are not considered to be increased as a result of the temperature control valve modification.

Based on conformance with the original system design criteria, the fact that the ECC System is an accident mitigation system, and that this modification does not introduce any new initiators to a previously postulated accident, the addition of this temperature control function can not increase the probability of occurrence of an accident previously evaluated in the USAR. Accidents reviewed involve the Loss of Coolant Accident applications described in USAR Chapter 6 with their corresponding consequence postulations shown in USAR Chapter 15, accident and transient scenarios as described in USAR Chapter 15, flooding and rupture postulations as described in USAR Chapter 3, and fire protection analyses as described in USAR Chapter 9.

The modification has been designed, procured, and installed to the original design codes and standards. The modification also satisfies single failure criteria and does not adversely affect the mitigation function of the ECC System. Therefore, the ability to mitigate accidents previously evaluated in the USAR is maintained and the radiological consequences of such accidents remain unaffected.

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

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

The modification has been designed to satisfy the requirements of the original ECC System. A single failure of the new configuration will not result in more than the loss of one respective

ECC System loop as already analyzed. Analysis of flooding shows no scenario greater than the currently bounding event. Missile generation is not a concern since no mechanisms conducive to that potential have been introduced. From the electrical analysis perspective, analysis has shown no adverse effects on the Emergency Diesel Generator loadings or other system applications.

Based on the above discussions, the proposed change would not create the possibility of a new or different kind of accident than those previously evaluated.

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

This request does not involve a significant reduction in a margin of safety. The modification, including design, procurement, and installation, has been performed in accordance with the applicable codes, standards, and installation specifications. The modification does not change the heat removal capabilities or any previously designed parameters of the ECC System. Hence, the ECC System margin of safety with respect to safety classification, protection, redundancy, heat removal capability, and seismic classification remains unaffected.

The margins of safety contained in the Technical Specifications and the associated Bases also remain unaffected by this modification due to conformance with the applicable codes, standards, and installation specifications. Specifically, Technical Specification 3.7.10, ''Emergency Closed Cooling Water (ECCW) System'' and the description in the Bases remain unchanged and fully applicable. The following Technical Specifications also remain unaffected and applicable: 3.3.3.2, ''Remote Shutdown System''; 3.7.1, ''Emergency Service Water (ESW) System—Divisions 1 and 2''; 3.7.4, ''Control Room Heating, Ventilation, and Air Conditioning (HVAC) System''; and the Technical Specifications related to Sections 3.8 (Electrical Power Systems), 3.5 (Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System) and 3.6 (Containment Systems). On this basis, the margins of safety defined in the Technical Specifications remain unchanged.

Therefore, the changes associated with this license amendment request do not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Gail H. Marcus.

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.

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

Date of application for amendment: August 27, 1996, as supplemented December 18, 1996, January 17, February 18, March 27, April 4, April 25, April 29, May 30, June 2, June 13, June 18, August 4, August 8, September 10, October 2 (RNP RA/97–0216), October 2, (RNP RA/97–0207), October 13, and October 21, 1997.

Brief description of amendment: This amendment addresses a more restrictive change proposed by the licensee in minimum allowable containment pressure.

Date of publication of individual notice in **Federal Register**: October 7, 1997 (62 FR 52362).

Expiration date of individual notice: October 21, 1997.

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

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

Date of amendment request: September 24, 1997.

Brief description of amendment request: The proposed amendment would add a surveillance requirement in Section 3/4.5.1 to perform a monthly valve position verification for each of the four residual heat removal crosstie valves.

Date of publication of individual notice in **Federal Register**: October 6, 1997 (62 FR 52162).

Expiration date of individual notice: November 5, 1997.

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

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

Date of amendment request: September 29, 1997.

Brief description of amendment request: The proposed amendment would change Technical Specification 3/4.11.1, ''Liquid Effluents—Concentration.'' The proposed change adds a requirement to perform weekly sampling and monthly and quarterly composite analyses of the Station Service Water System when the Reactor Auxiliaries Cooling System is contaminated.

Date of publication of individual notice in **Federal Register**: October 6, 1997 (62 FR 52161).

Expiration date of individual notice: November 5, 1997.

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

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

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

Date of amendment request: January 7, 1997, as supplemented on July 25, 1997, August 27, 1997, and September 15, 1997.

Brief description of amendment: The amendments correct an error involving the transposition of two of the reactor pressure vessel (RPV) pressure-temperature (P–T) limits curves between the Technical Specifications for the Brunswick Steam Electric Plant, Units 1 and 2 and update the hydrostatic pressure test limits curves for both units.

Date of issuance: October 7, 1997.

Effective date: October 7, 1997.

Amendment No.: 189 and 220.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register**: March 12, 1997 (62 FR 11485). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 7, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 27, 1996, as supplemented December 18, 1996, January 17, February 18, March 27, April 4, April 25, April 29, May 30, June 2, June 13, June 18, August 4, August 8, September 10, October 2 (RNP RA/97–0216), October 2, (RNP RA/97–0207), October 13, and October 21, 1997.

Brief description of amendment: This amendment addresses a more restrictive change proposed by the licensee in minimum allowable containment pressure.

Date of issuance: October 24, 1997.

Effective date: October 24, 1997.

Amendment No.: 176.

Facility Operating License No. DPR–23: Amendment revises the License and Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (62 FR 52362 dated October 7, 1997). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided for an opportunity to request a hearing by November 6, 1997, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of NSHC are contained in a Safety Evaluation dated October 24, 1997.

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

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

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, NC

Date of application for amendment: February 21, 1997.

Brief description of amendment: This amendment adds a specific time limit to Technical Specification Table 3.3–3 to place an inoperable refueling water storage tank level channel in a bypassed condition.

Date of issuance: September 30, 1997.

Effective date: September 30, 1997.

Amendment No: 74.

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

Date of initial notice in **Federal Register**: April 9, 1997 (62 FR 17225). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 30, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, NC 27605.

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

Date of application for amendments: March 5, 1997 as supplemented October 3, 1997.

Brief description of amendments: The amendments would revise the Technical Specifications by removing the main steamline radiation monitor reactor scram function and the main steamline tunnel radiation isolation function.

Date of issuance: October 24, 1997.

Effective date: Immediately, to be implemented within 60 days.

Amendment Nos.: 163, 158.

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

Date of initial notice in **Federal Register**: April 18, 1997 (62 FR 19141). The October 3, 1997, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 24, 1997.

No significant hazards consideration comments received: No.

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

Entergy Gulf States, Inc., Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, LA

Date of amendment request: August 29, 1996, supplemented August 29, 1996 (proprietary), September 5, and October 8, 1997.

Brief description of amendment: The amendment eliminates the Average Power Range Monitor (APRM) setpoint T-Factor setdown requirements and provides for reactivity anomaly calculation improvements. The request to decrease the local power range monitor (LPRM) calibration frequency will be handled by separate review and action.

Date of issuance: October 10, 1997.

Effective date: October 10, 1997.

Amendment No.: 100.

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

Date of initial notice in **Federal Register**: October 23, 1997 (61 FR 55032). The Licensee's letters dated August 29, 1996 (proprietary), September 5, and October 8, 1997, provided additional clarification and corrections to other TSs that would have erroneously referenced the TSs being eliminated and did not change the staff's initial no significant hazards determination.

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803.

GPU Nuclear Corporation, et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, PA

Date of application for amendment: July 30, 1997, as supplemented September 19, and September 24, 1997.

Brief description of amendment: The amendment reduces current technical specification leakage limit from the decay heat removal system from 6.0 gallons per hour (gph) to 0.6 gph.

Date of issuance: October 15, 1997.

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

Amendment No.: 205.

Facility Operating License No. DPR–50: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 27, 1997 (62 FR 45458). The September 19, and September 24, 1997, submittals did not affect the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 15, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Law/Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

GPU Nuclear Corporation, et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, PA

Date of application for amendment: August 12, 1997, as supplemented August 28, September 15, October 3, 9, and 10, 1997.

Brief description of amendment: The amendment changes the technical specifications surveillance requirements for once-through steam generator inservice inspection for Cycle 12 operation.

Date of issuance: October 16, 1997.

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

Amendment No.: 206.

Facility Operating License No. DPR–50: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 27, 1997 (62 FR 45458). The supplemental letters did not affect the initial no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Law/Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, TX, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, TX

Date of amendment request: August 14, 1997, as supplemented September 23, 1997. The supplement provided clarifying information within the scope of the amendment request and did not change the initial no significant hazards consideration determination.

Brief description of amendments: The amendments revise the allowed tolerance of the reactor coolant system volume provided in Technical Specification 5.4.2 to account for steam generator tube plugging.

Date of issuance: October 20, 1997.

Effective date: October 20, 1997.

Amendment Nos.: Unit 1—Amendment No. 92; Unit 2—Amendment No. 79.

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

Date of initial notice in **Federal Register**: August 26, 1997 (62 FR 45278). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 20, 1997.

No significant hazards consideration comments received: No.

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

Northeast Nuclear Energy Company, Docket No. 50–245, Millstone Nuclear Power Station, Unit 1, New London County, CT

Date of application for amendment: February 7, 1997, as supplemented April 3 and September 19, 1997.

Brief description of amendment: The amendment clarifies the requirement for calibration of instrument channels that use resistance temperature detectors or thermocouples.

Date of issuance: October 22, 1997.

Effective date: As of the date of issuance, to be implemented within 90 days.

Amendment No.: 102.

Facility Operating License No. DPR–21: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 9, 1997 (62 FR 17236). The April 3 and September 19, 1997, letters provided additional and clarifying information that did not change the scope of the February 7, 1997, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 22, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 19, 1997.

Brief description of amendment: Technical Specification Table 2.2–1 NOTES 1 and 3 define the values for the constants used in the Overtemperature Delta-T and Overpower Delta-T reactor trip system instrumentation setpoint calculators. The amendment makes changes to the NOTES as well as the associated Bases section.

Date of issuance: October 22, 1997.

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

Amendment No.: 152.

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

Date of initial notice in **Federal Register**: July 30, 1997 (62 FR 40852). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 22, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 6, 1996, as supplemented April 10 and October 1, 1997.

Brief description of amendments: The amendments revise Technical Specifications governing the cooling water system and are a partial response to the licensee's application. The changes improve plant operation based on operational experience with the vertical motor-driven cooling water pump. The changes also incorporate information gathered by the licensee during its self-assessment Service Water System Operational Performance Inspection (SWSOPI) completed in late 1995. The remainder of the licensee's application will be addressed in a separate licensing action.

Date of issuance: October 21, 1997.

Effective date: October 21, 1997, with full implementation within 90 days.

Amendment Nos.: 131 and 123.

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

Date of initial notice in **Federal Register**: January 29, 1997 (62 FR 4338) The April 10 and October 1, 1997, letters provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 21, 1997.

No significant hazards consideration comments received: No.

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

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50–278, Peach Bottom Atomic Power Station, Unit No. 3, York County, PA

Date of application for amendment: June 30, 1997, as supplemented by letter dated September 26, 1997.

Brief description of amendment: Revises the minimum critical power ratio (MCPR) safety limit in Section 2.1 of the Technical Specifications from 1.07 to 1.11 for two recirculation loops in operation. For a single loop in operation, the MCPR will change from 1.08 to 1.12. The new MCPR safety limits reflect the effect of the new General Electric—13 part length fuel design and other Peach Bottom core-specific parameters.

Date of issuance: October 9, 1997.

Effective date: As of the date of issuance, to be implemented prior to startup from Unit 3 refueling outage 3R11.

Amendment No.: 225.

Facility Operating License No. DPR–56: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 13, 1997 (62 FR 43373).

The supplemental letter provided clarifying information that did not change the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 9, 1997.

No significant hazards consideration comments received: No.

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.

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

Date of application for amendments: April 9, 1997.

Brief description of amendments: These amendments revise the TSs to clarify existing battery-specific gravity requirements, delete the requirement to correct specific gravity values based on electrolyte level, and allow the use of charging current measurements to verify the battery's state of charge.

Date of issuance: October 8, 1997.

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

Amendment Nos.: 123 and 88.

Facility Operating License Nos. NPF–39 and NPF–85: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 4, 1997 (62 FR 30643).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 8, 1997.

No significant hazards consideration comments received: No.

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

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

Date of amendments request: March 7, 1997.

Brief Description of amendments: The amendments change the Technical Specifications for both Farley units to allow operability testing for certain containment isolation valves during defueled status.

Date of issuance: October 17, 1997.

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

Amendment Nos.: Unit 1—130; Unit 2—123.

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

Date of initial notice in **Federal Register**: April 23, 1997 (62 FR 19834).

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

No significant hazards consideration comments received: No.

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

Southern Nuclear Operating Company, Inc., Docket No. 50–348, Joseph M. Farley Nuclear Plant, Unit 1, Houston County, AL

Date of amendment request: September 3, 1997.

Brief Description of amendment: The changes reduce the number of required incore detectors necessary for continued operation for the remainder of Cycle 15 only.

Date of issuance: October 23, 1997.

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

Amendment No.: 131.

Facility Operating License No. NPF–2: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: September 10, 1997 (62 FR 47695).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 23, 1997.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: June 20, 1997.

Brief description of amendment: Modify the Watts Bar Technical Specifications (TS) to incorporate the use of Code Case N–514 into the methodology for the Pressure-Temperature Limits Report.

Date of issuance: October 21, 1997.

Effective date: October 21, 1997.

Amendment No.: 9.

Facility Operating License No. NPF–90: Amendment revises the TS.

Date of initial notice in **Federal Register**: September 10, 1997 (62 FR 47700).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 21, 1997.

No significant hazards consideration comments received: None

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

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

Date of amendment request: July 3, 1997, as supplemented by letter dated August 20, 1997.

Brief description of amendment: The amendment revises Surveillance Requirements 4.3.1.2 and 4.3.2.2, and Technical Specifications 3/4.3.1 and 3/4.3.2, and associated Bases Sections B 3/4.3.1 and B 3/4.3.2 to eliminate periodic response time testing requirements for selected pressure and differential pressure sensors in the reactor trip system and engineered safety features actuation system instrumentation channels.

Date of issuance: October 20, 1997.

Effective date: October 20, 1997, to be implemented prior to restart from the ninth refueling outage currently scheduled to start on October 4, 1997.

Amendment No.: 113.

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

Date of initial notice in **Federal Register**: July 30, 1997 (62 FR 40862).

The August 20, 1997, supplemental letter provided additional clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 20, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, KS 66801 and Washburn University School of Law Library, Topeka, KS 66621.

Dated at Rockville, Maryland, this 29th day of October 1997.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 97–29138 Filed 11–4–97; 8:45 am]

BILLING CODE 7590–01–P

THE PRESIDENT'S COUNCIL ON SUSTAINABLE DEVELOPMENT

The Eighteenth Meeting of the President's Council on Sustainable Development (PCSD) in Atlanta, Georgia

Summary: The President's Council on Sustainable Development (PCSD), a Presidential Commission with representation from industry, government, environmental, and Native American organizations, will convene its eighteenth meeting in Atlanta, Georgia on Thursday, November 20, 1997.

Under its current charter, the Administration asked the Council to continue its work by continuing to forge consensus on policy, demonstrating implementation, getting the word out about sustainable development, and evaluating progress. The Council will advise the President in four specific areas: domestic implementation of policy options to reduce greenhouse gas emissions, next steps in building the new environmental management system of the 21st century, promoting multi-jurisdictional and community cooperation in metropolitan and rural areas, and policies that fosters the United States' leadership role in sustainable development internationally.

At the Council's last meeting in Tulsa, Oklahoma on September 22, 1997, members were briefed on the science impacts, technology impacts, and economics related to climate change. The Council also heard from Tulsa's community about ways in which the climate change issue affects their lives.

At this next meeting, the Council will receive input from a community forum on climate change, focus on technology options to reduce greenhouse gas emissions and hear from a series of experts in the field. Specifically, the discussion will address the following agenda items:

- Current sources of greenhouse gas emissions; and
- Technology opportunities in a variety of sectors within the United States economy to reduce greenhouse gas emissions.

Public comment period: The Council will seek public comment on potential Council activities to implement the Administration's directive.

Specifically, the Council is interested in hearing from the public on the following questions:

- How might climate change affect the quality of life in the Atlanta region?
- Are there local opportunities in Atlanta, Georgia and surrounding regions to reduce greenhouse gas emissions?
- What policy recommendations should the Council give to President Clinton to more quickly develop and deploy energy efficient technologies?

The Council's previous recommendations to the President may be found in two reports: Sustainable America: A New Consensus for Prosperity, Opportunity, and a Healthy Environment for the Future (March 1996) and Building on Consensus: A Progress Report on Sustainable America (January 1997). Copies of both reports can be ordered by calling 1–800–363–3732 or downloaded off the Internet at "http://www.whitehouse.gov/PCSD".

Dates/Times: Thursday, November 20, 1997 from 9:00 a.m. to 1:00 p.m.

Place: Georgia Public Broadcasting Building, 206 14th Street in the main floor television studio, Atlanta, Georgia, 30318. PH: 404–685–2253; FAX: 404–756–2417.

Status: Open to the public. Public comments are welcome and may be submitted orally on November 20 or in writing any time prior to or during the meeting. Please submit written comments prior to meeting to: PCSD, Public Comments, 730 Jackson Place, NW, Washington, D.C. 20503, or fax to: