Week of August 24—Tentative

Tuesday, August 25

10:00 a.m.—Briefing on 10 CFR Part 70—Proposed Rulemaking, ''Revised Requirements for the Domestic Licensing of Special Nuclear Material'' (Public Meeting) (Contact: Elizabeth Ten Eyck, 301–415–7212)

Wednesday, August 26

2:00 p.m.—Briefing on Status of Activities with CNWRA and HLW Program (Public Meeting) (Contact: Mike Bell, 301–415–7286)

3:30 p.m.—Affirmation Session (Public Meeting) (if needed)

Week of August 31

Wednesday, September 2

10:00 a.m.—Briefing on PRA Implementation Plan (Public Meeting) (Contact: Tom King, 301–415–5828)

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

Thursday, September 3

10:00 a.m. and 1:30 p.m.—All Employees Meetings (Public Meetings) on ''The Green'' Plaza Area between buildings at White Flint (Contact: Bill Hill—301–415–1661)

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

The NRC Commission Meeting Schedule can be found on the Internet at: http://www.nrc.gov/SECY/smj/schedule.htm

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, D.C. 20555 (301–415–1661). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: August 7, 1998.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 98–21667 Filed 8–7–98; 4:14 pm]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from July 20, 1998, through July 31, 1998. The last biweekly notice was published on July 29, 1998 (63 FR 40551).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administration Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By September 11, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

Baltimore Gas and Electric Company, Docket No. 50–318, Calvert Cliffs Nuclear Power Plant, Unit No. 2, Calvert County, Maryland

Date of amendment request: July 20, 1998.

Description of amendment request: Baltimore Gas and Electric Company (BGE) request a modification involving replacing the service water (SRW) heat exchangers with new plate and frame heat exchangers having increased thermal performance capability. A similar license amendment dated February 8, 1998, was granted to Operating License No. DPR–53—Calvert Cliffs Nuclear Power Plant, Unit 1.

The planned modification for Unit 2 is virtually identical to the one just completed for Unit 1 during the spring 1998 refueling outage. The only exception is the addition of an extra manual valve in the Unit 2 system to isolate the bypass line for maintenance. This additional manual valve is needed due to the change in location of the tie-in to the main header. (The Unit 1 bypass line ties into the main header downstream of a control valve; therefore, it did not need a separate isolation valve for maintenance.)

The saltwater and SRW piping configuration will be modified as necessary to allow proper fit-up to the new components. A flow control scheme to throttle saltwater flow to the heat exchangers and the associated bypass lines will be added. Saltwater strainers with an automatic flushing arrangement will be added upstream of each heat exchanger. The majority of the physical work associated with this modification is restricted to the SRW pump room.

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.

None of the systems associated with the proposed modification are accident initiators. The SW and SRW Systems are used to mitigate the effects of accidents analyzed in the UFSAR [Updated Final Safety Analysis Report]. The SW and SRW Systems provide cooling to safety-related equipment following an accident. They support accident mitigation functions; therefore, the proposed modification does not increase the probability of an accident previously evaluated.

The proposed modification will increase the heat removal capacity of the SRW System. The design provided under this activity ensures that the safety features provided by the SW and SRW are maintained, and in some instances enhanced; i.e., the availability of important-to-safety equipment required to mitigate the radiological consequences of an accident described in the UFSAR is enhanced by the flexibility and increased thermal margin provided with this design.

The redundant cooling capacity of the SW and SRW Systems have not been altered. Furthermore, the proposed activity will not change, degrade, or prevent actions described or assumed in any accident described in the UFSAR. The proposed activity will not alter any assumptions previously made in evaluating the radiological consequences of any accident described in the UFSAR. Therefore, the consequences of an accident previously evaluated in the UFSAR have not increased.

Therefore, the proposed modification 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 activity involves modifying the SW and SRW System components necessary to support the installation of new SRW heat exchangers. None of the systems associated with this modification are identified as accident initiators in the UFSAR. The SW and SRW Systems are used to mitigate the effects of accidents analyzed in the UFSAR. None of the functions required of the SRW or SW System have been changed by this modification. This activity does not modify any system, structure, or component such that it could become accident initiator, as opposed to its current role as an accident mitigator.

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 design basis for the SW and SRW System is the availability of sufficient cooling capacity to ensure continued operation of equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with assumptions used in the accident analysis.

The design, procurement, installation, and testing of the equipment associated with the proposed modification are consistent with the applicable codes and standards governing the original systems, structures, and components. The design of instruments and associated cabling ensures that physical and electrical separation of the two subsystems is maintained. Common-mode failure is not introduced by the activity. The equipment is qualified for the service conditions stipulated for that environment. New cable and raceways for this design will be installed in accordance with seismic design requirements. The additional electrical load has been reviewed to ensure the load limits for the vital 1E buses are not exceeded. The circuits and components related to the control valves control loops are safety-related, are similar to those used for the other safety-related flow control functions. The proposed modification will not have any adverse effects on the safety-related functions of the SW and SRW Systems.

For the above reasons, the existing licensing bases have not been altered by the proposed modification. This activity will not reduce the margin of safety as it exists now. In fact, the margin of safety has been increased by this activity due to the increase in the thermal capacity of the dual train design (i.e., two heat exchangers per train versus one heat exchanger per train of the original design) and the increased availability of safety-related components.

Therefore, this proposed modification does not significantly reduce the margin of safety.

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

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 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.

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

Date of amendment request: July 13, 1998.

Description of amendment request: The proposed amendments would revise the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS–1 and BVPS–2) Updated Final Safety Analysis Report (UFSAR) descriptions of the Intake Structure main entrance and interconnecting cubicle doors. The current UFSAR descriptions state that the cubicle access doors are open to permit excess water from a major pipe rupture to flow out of the cubicles thereby avoiding internal flooding. The proposed changes would address a new failure mode of safety-related equipment that had not been previously considered for BVPS–1. The proposed changes would state that the cubicle interconnecting flood protection doors are normally closed with their inflatable seals depressurized and that the associated security/fire doors are normally closed. The proposed door closure arrangement is intended to protect the safety-related equipment in the interconnecting cubicles from the consequences of potential internal flooding.

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

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

The proposed change revises the text of the UFSAR for Unit 1 and Unit 2 to describe how protection is provided against potential internal floods in the cubicles that house the Unit 1 River Water and Unit 2 Service Water Pumps. The previous description concluded that the Unit 1 River Water pumps were protected because open cubicle access doors will permit excess water to flow out of the cubicles. The practice that has changed, and is described in the proposed revisions to the Unit 1 and Unit 2 UFSARs, will provide protection of the Unit 1 River Water Pumps and the Unit 2 Service Water Pumps so that no flooding event can adversely affect more than one Unit 1 or Unit 2 pump. Therefore, it can be concluded that the proposed changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

The effect of flooding the pump cubicles was considered in BVPS–1 to have no adverse effect because open cubicle access doors would permit excess water to flow out of the cubicles, and pipe cracks in moderate energy piping was not part of the design basis. Revising the door arrangement described in the BVPS–1 UFSAR such that the security/fire doors are normally closed, requires that the effects of flooding be considered. Engineering analysis shows that a moderate energy pipe crack, (i.e., the BVPS–2 design basis internal flood), produces a leak rate of 1162 gpm, which results in a maximum water level of 0.82 feet, with the security/fire doors closed. The water level in the adjacent cubicle would reach a level at 0.37 feet. This is below the level which would cause failures of the MCCs [Motor Control Centers] in the pump cubicles.

The maximum leak rate from a failure of a Unit 1 rubber expansion joint in a pump cubicle would result in water rising to a level which would cause the MCCs to be flooded and fail; therefore, maintaining the flood door between the adjacent cubicles closed limits the impact to a single train.

Failure of a single train of River Water is analyzed in the USAR; therefore, this change would not introduce a new or different type of accident.

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

The proposed change in the Unit 1 and Unit 2 UFSARs describes how protection is provided for the Unit 1 River Water, and the Unit 2 Service Water pumps. Protection of the Unit 1 River Water Pumps and the Unit 2 Service Water pumps is provided so that no flooding event can adversely affect more than one Unit 1 or Unit 2 pump. Therefore, it can be concluded that the proposed changes do not involve any reduction in a margin of safety.

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

Local Public Document Room location: B.F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: July 9, 1998.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.7.1.1 and associated Bases for both units. TS 3.7.1.1 currently provides requirements for reducing the power range high neutron flux trip setpoint when one or more main steam safety valves are inoperable. The current basis for determining the amount of trip setpoint reduction has been determined to be non-conservative. The proposed amendment would specify maximum allowable reactor power level based on the number of operable main steam safety valves rather than requiring a reduction in reactor trip setpoint. This change would be consistent with the NRC staff's guidance provided in the NRC's improved Standard Technical Specifications for Westinghouse plants (NUREG–1431, Revision 1). The maximum allowable reactor power level with inoperable safety valves would be calculated based on the recommendations of Westinghouse Nuclear Safety Advisory Letter (NSAL) 94–01. The proposed change to the Unit 1 TS 3.7.1.1 would also delete reference to 2 loop operation since 2 loop operation is not a licensed condition for either unit.

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

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

The proposed change will generally incorporate the Improved Standard Technical Specification (ISTS) main steam safety valve (MSSV) requirements of NUREG–1431 into Specification 3.7.1.1 and associated Bases. The Unit 1 specification currently includes reference to 2 loop operating requirements in Action ''b'' and Table 3.7–2. Reference to 2 loop operation is being deleted since it is not addressed in the ISTS and is not a licensed condition for these plants. The limiting condition for operation has been modified to incorporate the ISTS wording and requires MSSV operability in accordance with Tables 3.7–1 and 3.7–2. Table 3.7–1 lists the maximum allowable power level as a function of the number of operable MSSVs per steam generator and continues to require a minimum of 2 operable MSSVs per steam generator for continued plant operation. Table 3.7–2 specifies the MSSV lift setting and tolerance for each MSSV. The valve lift setting remains unchanged along with the current tolerance of +1 percent −3 percent. The Applicability statement has not been changed since it is consistent with the ISTS requirements.

Proposed Action ''a'' applies with one or more inoperable MSSVs and requires that within 4 hours power must be reduced in accordance with the value specified in Table 3.7–1; otherwise, shut down. This action satisfies the same goal as the current action by restricting thermal power so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity for that steam generator. Proposed Action ''b'' incorporates additional conservatism by specifically requiring at least 2 operable MSSVs per steam generator. This ensures that a minimum overpressure protection is available during all applicable modes of operation. Proposed Action ''c'' provides an exception to Specification 3.0.4 which does not allow entry into a mode where the Limiting Condition for Operation (LCO) is not met and actions require a shutdown. This exception is not addressed in the ISTS requirements; however, an exception to Specification 3.0.4 allows entry into a mode where the LCO applies in conformance with the action statements.

Proposed Surveillance Requirement 4.7.1.1 requires verification of the lift setpoint for each MSSV listed in Table 3.7–2 in accordance with the Inservice Test Program. Note (1) is applied to Surveillance Requirement 4.7.1.1 to provide clarification of the testing requirements, such that this testing is required only in Modes 1 and 2 so that the plant can enter Modes 2 and 3 where this specification applies without first performing the test. A note (2) has been applied to the lift setting in Table 3.7–2 that requires a setting corresponding to the ambient conditions of the valve at the nominal operating temperature and pressure. The ISTS does not include this note but it has been included for consistency with the current note and provides a clear reminder to test personnel of the required test conditions.

The safety valve Bases have been revised to generally incorporate the ISTS Bases which significantly improve the content and understanding of the MSSV requirements. These changes are consistent with the UFSAR [Updated Final Safety Analysis Report] design description and analysis assumptions where the MSSVs provide the required overpressure protection. The proposed changes are consistent with the regulations and provide additional assurance that the secondary side pressure remains within the bounds of the safety analyses; therefore, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes generally incorporate the ISTS MSSV requirements to ensure adequate secondary side overpressure protection is available and properly maintained. The revised Limiting Condition for Operation (LCO) limits plant power level based on the number of operable MSSVs as stated in Table 3.7–1 and provides the valve lift settings and tolerances as shown in Table 3.7–2. The actions require a reduction in power when the number of valves is less than the full complement for each steam generator and also require at least 2 operable MSSVs per steam generator. When these requirements cannot be met a plant shutdown is required. An action also provides an exception to Specification 3.0.4 and is consistent with the exception currently provided. These actions are more conservative than the current requirements and provide additional assurance that Specification 3.7.1.1 will continue to govern the MSSV limitations in a manner consistent with the accident analyses assumptions. The revised surveillance requirement provides clearly understandable testing requirements to ensure the MSSVs are adequately monitored and will perform in accordance with the accident analysis assumptions. The proposed change does not introduce any new mode of operation or require any physical modification to the plant; therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The MSSVs ensure the ASME [American Society of Mechanical Engineers] Code, Section III requirements are maintained to limit the secondary system pressure to within 110 percent of the design pressure when passing the design steam flow. This ensures that the overpressure protection system can cope with all operational and transient events. Operation with less than the full number of MSSVs is permitted as long as thermal power is restricted to meet the ASME Code requirements. This limitation is provided in the proposed technical specifications along with operability and surveillance requirements to ensure the level of overpressure protection is maintained. MSSV operability is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. MSSV operability is determined by surveillance testing in accordance with the Inservice Test program which provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the reactor coolant pressure boundary. The proposed change continues to ensure that the required components are properly maintained and that the assumed parameters are verified during the applicable conditions and on a consistent basis; therefore, this change will not reduce the margin of safety.

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

Local Public Document Room location: B.F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

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

NRC Project Director: Robert A. Capra.

GPU Nuclear, Inc. et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: July 21, 1998.

Description of amendment request: The proposed change request would permit an alternative to the requirement to perform Control Rod Drive (CRD) scram time testing with the reactor pressurized prior to resuming power operation. The change would permit: (1) scram time testing with the reactor depressurized prior to resuming operation, and (2) a second scram time test with the reactor pressure above 800 psig, prior to exceeding 40% reactor power.

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

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

There will not be an increase in the probability of occurrence of an accident previously evaluated in the Safety Analysis Report (SAR) because the requested change provides additional assurance that the CRD System is able to perform its safety function, and therefore does not change the probability of occurrence of an accident.

There will not be an increase in the consequences of an accident previously evaluated in the Safety Analysis Report (SAR) because the requested change will ensure that the CRD System is able to perform its safety function, and therefore does not change the consequences of an accident.

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

The requested change will not create the possibility of a new or different kind of accident from any accident previously evaluated. The first issue associated with the requested change is increased wear on the CRDs, resulting in increased buffer seal wear or failure. This wear or failure of the buffer seal would result in difficulty or inability to withdraw the rod subsequent to the depressurized scram. The safety function of the rod to insert on a scram signal, however, would be unaffected by this seal degradation. Therefore, there is no safety concern with the increased wear due to performance of the cold scram test.

The other consideration associated with the new requested change is the possible increased risk of stub tube leakage during the cold (depressurized) test. Without the download due to reactor pressure, the momentary upward loading on the CRD stub tube puts the stub tube into tension. Any flaws in the stub tube could grow and eventually result in a stub tube leak. The likelihood of flaws in the stub tubes, however, is very small, based on the extensive repair work on the stub tube surfaces performed prior to plant operation. The integrity of the stub tube repairs is verified by the 1000 pound leak test performed during every startup of the reactor. This test, therefore, poses very minimal risk of stub tube leakage.

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

The change will not decrease the margin of safety as defined in the basis of any Technical Specification. This is because the requested change, like the existing Technical Specification test, provides assurance that the CRD System is able to perform its safety function, and therefore does not change 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: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753

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

NRC Project Director: Cecil O.Thomas.

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

Date of amendment request: June 11, 1998

Description of amendment request: The proposed amendment would incorporate an alternative high radiation area control for Three Mile Island Nuclear Station, Unit No. 1 (TMI–1) in accordance with 10 CFR 20.1601(c). The alternative would modify Technical Specification 6.12 to allow for a conspicuously posted barricade and flashing light in individual high radiation areas that are located within large areas where no enclosure exists for locking, and no enclosure can be reasonably erected. A minor clarification to indicate that the requirement of paragraph 6.12.1.a also applies to 6.12.1.b and an editorial change were added.

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 the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed amendment involves changes to the TMI–1 Technical Specifications, which are consistent with Regulatory Guide 8.38. This change does not involve any change to system or equipment configuration. The proposed amendment incorporates an alternative high radiation area control, which has been previously found to be acceptable by the NRC. The reliability of systems and components relied upon to prevent or mitigate the consequences of accidents previous evaluated is not degraded by the proposed changes. Therefore, this change does not increase the probability or consequences of an accident previously evaluated.

2. 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 previously evaluated. This change only involves controls for access to high radiation areas. Access to plant equipment during normal or accident conditions will not be affected by utilizing this alternate method. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The proposed amendment is consistent with Regulatory Guide 8.38. The proposed amendment involves high radiation area access control and is not related to the margin of safety associated with any plant operation or transients. Therefore, it is concluded that operation of the facility in accordance with the proposed amendment does not involve a significant reduction in a margin of safety.

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

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.

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

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: May 20, 1998.

Description of amendment request: The proposed change would revise the Refueling Water Storage Tank (RWST) setpoint associated with Automatic Switchover to the Containment Sump. This change would require a revision to the Engineered Safety Features Actuation System Instrumentation Trip Setpoints, Table 3.3–4, Functional Unit 8.b, RWST Level—Low-Low, along with associated Bases Section 3/4.3.2.

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 does not adversely affect accident initiators or precursors and does not alter the design assumptions affecting the ability of the RWST and the ECCS [Emergency Core Cooling System] pumps to mitigate the consequences of an accident.

Revising the RWST Level Low-Low setpoint has a negligible effect on the operating margin for the RWST. The revised setpoint assures that the minimum RWST volume assumed in the accident analyses is injected prior to switchover to the recirculation mode. The effect on containment flood level, equipment qualification, and pH of the containment sump and the containment spray fluid, remain within the limits assumed in the accident analyses.

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

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

The setpoint change does not affect the function of the level monitoring channels or any function of the accident mitigation equipment associated with the RWST. No new components or physical changes are involved with this change. There are no changes to the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences in the Seabrook Station [updated final safety analysis report] UFSAR. The new setpoint will continue to initiate the automatic ECCS transfer from the injection mode to the recirculation mode and provide the alarm to alert the operator(s) to begin the manual actions necessary to complete the transfer to the recirculation mode. Manual operator action is required to complete the switchover to the recirculation mode. With the new setpoint, sufficient time remains available for the operator(s) to complete the transfer prior to receipt of the RWST EMPTY alarm and reaching the vortexing level in the RWST. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

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

The design bases for the RWST Level Low-Low setpoint is to ensure that the minimum volume of water to support the assumptions made in the safety analysis is injected prior to switchover and that there is adequate time available for the operators to complete the manual actions necessary to complete the switchover to the recirculation mode prior to actuation of the RWST EMPTY alarm. The minimum injection volume assumed in the accident analyses, and time required for the operator(s) to initiate and complete manual actions to complete switchover to the recirculation mode prior to receipt of the RWST EMPTY alarm, remains unaffected by this change. 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: Exeter Public Library, Founders Park, Exeter, NH 03833.

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

NRC Project Director: Cecil O.Thomas.

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

Date of amendment request: May 21, 1998.

Description of amendment request: The proposed change would revise selected Technical Specification (TS) surveillance requirements to accommodate fuel cycles of up to 24 months for surveillances that are currently performed at each 18-month or other specified outage interval. Specifically, the following TS surveillance requirements would be revised by the proposed change: 4.1.3.3, Digital Rod Position Indication; 4.8.1.1.1.b, A.C. Sources—Operating—Transfer of 1E Bus Power from Normal to Alternate Source; 4.8.1.1.2.f.1 through 15, A.C. Sources—Operating—Emergency Diesel Generator Surveillances; 4.8.3.3, Onsite Power Distribution—Trip Circuit For Inverter I–2A; 4.8.2.1.c, d & f, D.C. Sources—Operating—125V D.C. Batteries and Chargers; 4.8.4.2.a.1) & a.2), Containment Penetration Conductor Overcurrent Protective Devices and Protective Devices for Class 1E Power Sources Connected to Non-Class 1E Circuits; 4.8.4.3, Motor Operated Valves Thermal Overload Protection. In addition, the components listed in Technical Specification 4.8.2.2, D.C. Sources—Shutdown—125V DC Batteries and Chargers, have been evaluated to support an extension in frequency to 24 months (+25%).

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

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

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. The proposed changes do not alter or prevent the ability of structures, systems, or components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). The proposed changes are administrative in nature and do not change the level of programmatic controls or the procedural details associated with aforementioned surveillance requirements.

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

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

Deletion of the restriction ''during effect on safe operation of the plant is given prior to conduct of a particular surveillance in a condition or mode other than shutdown.

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

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

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

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

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

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

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

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

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

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

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

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: July 2, 1998.

Description of amendment request: The proposed amendment would revise the updated Final Safety Analysis Report (FSAR) by changing FSAR Sections 9.7.2, ''Service Water,'' and 9.4, ''Reactor Building Closed Cooling Water,'' to discuss the use of various types of internal protective coatings and liners used in the piping and components of the systems. The proposed change also indicates that periodic maintenance, surveillances, and inspections would be conducted to ensure that coating or liner degradation would be promptly detected and corrected to provide reasonable assurance that the systems can perform their safety-related functions.

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

The proposed change does not involve significant hazards consideration because the changes would not:

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

The SWS [Service Water System] provides cooling water directly or indirectly to a multitude of mitigating and support systems such as safety injection, containment spray, and RBCCW [Reactor Building Closed-Cooling Water]. Therefore either directly or indirectly, the SWS is credited in the mitigation of virtually all analyzed operating events and accidents. However, there are no failures of the SWS which would directly initiate any of the licensing basis accidents. Therefore, the probability of occurrence of accidents previously evaluated is not increased by this activity.

The SWS is comprised of two separate and independent trains, each capable of providing the cooling capacity required for normal and accident operation. Therefore, the failure of a single heat exchanger or train will not influence the consequences of an accident. Only a common mode loss of SWS function could affect accident consequences. It can be postulated that lining material could be released as a result of the SWS response to an accident or as a result of a seismic event, resulting in heat exchanger blockage in both trains (common mode). However, the discussion below provides the basis for concluding that lining degradation will not increase the consequences of an accident.

In response to a Safety Injection Actuation Signal or a Loss of Normal Power event, the quantity of flow in safety related SWS heat exchangers may increase significantly, imparting higher loads on the pipe linings than are typically present during normal operation. In spite of this flow increase, it is considered to be much more likely that any lining degradation will occur and be detected under normal operating conditions, and will be corrected prior to the occurrence of an event of the type discussed above. SWS pump flow surveillances, performed periodically during normal operation, subject significant portions of the SWS to flow levels which equal or exceed those expected to occur during accidents. Any degraded lining material prone to be released during an accident is expected to be released during these pump surveillances. The inspections, operating procedures, and surveillances ensure that significant lining releases will be promptly detected and investigated. In addition, SWS design features provide the system with a significant level of protection against degraded lining debris (e.g., standby spare RBCCW heat exchanger and EDG [Emergency Diesel Generator] engine cooler strainers) both during normal operation and while responding to an accident.

An evaluation was performed to assess the significance of loading on the linings due to a postulated seismic event. The importance of seismic loads depends upon their magnitude relative to normal operating loads, and on their relative frequency of occurrence. Normal operating loads include steady state flow loads as well as transients due to pump swaps and realignments for surveillances. The evaluation determined that normal operating loads are significantly greater than anticipated seismic loads concurrent with steady state flow loads. Therefore, if normal operating loads do not cause lining to become detached, it is very unlikely that a random seismic event would cause detachment. In addition, while flow loads are continuously present in most of the system and normal transients occur many times during an operating cycle, seismic events at the Millstone site are very infrequent (the repetition rate of an OBE [Operating Basis Earthquake] is hundred of years). Should normal operating loads cause lining detachment, it is much more probable that this released material will be detected, and the degraded condition corrected, prior to the occurrence of a seismic event.

Based upon these discussions, and given the random nature of lining degradation and the scrutiny with which the SWS is operated and maintained, it is not considered to be credible that the operability of both SWS trains will be simultaneously impaired by lining degradation and release.

Therefore, there is no significant increase in the probability or consequences of an accident previously evaluated.

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

As discussed above, the failure of a single heat exchanger or a single SWS train will not cause an accident. Only a common mode loss of SWS function could create the possibility of a previously unanalyzed accident, and this loss would not directly initiate an accident. However, for the reasons discussed above, lining degradatiion will not cause common mode failures to occur.

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

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

The margins of safety of the protective boundaries (fuel matrix/cladding, reactor coolant system pressure boundary, and containment) would not be impacted by the postulated release of lining material into the SWS. The accident analyses in the FSAR [Final Safety Analysis Report] demonstrate the performance of the protective boundaries. As discussed previously, it is not considered to be credible that lining degradation will cause a common mode loss of SWS function. Therefore, since the accident analyses credit only one SWS train, released lining would not affect accident analyses assumptions. On this basis, it is concluded that margins of safety as demonstrated by the accident analyses would not be affected by postulated lining material release.

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

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

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

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

NRC Deputy Director: Phillip F. McKee.

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

Date of amendment request: July 17, 1998.

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) surveillance requirements for the onsite emergency diesel generators (EDGs) to achieve an overall improvement in the EDGs reliability and availability. The proposed changes would modify the requirement for operability tests of an EDG when the other EDG is inoperable, delete the requirement for operability tests when one or both offsite A.C. sources are inoperable, eliminate fast loading of the EDGs except for the 18-month testing, and eliminate fast starts (15 seconds) except for once per 6 months and during the 18-month testing. These proposed changes are generally consistent with the guidance provided in Generic Letter (GL) 84–15, ''Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability,'' dated July 2, 1984, and GL 93–05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation,'' dated September 27, 1993. Justification for deviations from the guidance provided in the GLs is provided in the licensee's submittal.

In addition, the licensee proposes to revise the wording in the TS requirements for offsite circuits to be consistent with NUREG–0212, ''Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors,'' Revision 2, fall 1980, and the guidance provided in GL 91–04, ''Changes in Technical Specification Surveillance Intervals to Accommodate 24-Month Fuel Cycle,'' dated April 2, 1991. The associated TS Bases will be updated to reflect the proposed changes.

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

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

The LCOs [Limiting Conditions for Operation] for Technical Specifications [TSs] 3.8.1.1 and 3.8.1.2 will be changed to require a transmission network between offsite power and the onsite Class 1E distribution system, instead of just between offsite and the switchyard. This change, which will expand the requirement, is consistent with the current Millstone Unit No. 2 interpretation of the required distribution system. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously analyzed.

The diesel generators (DGs) supply power to the emergency busses at Millstone Unit No. 2 in the event of a loss of normal power (LNP). The emergency busses supply the vital equipment used to mitigate the consequences of design basis accidents. Therefore, the diesel generators are vital equipment used to mitigate the consequences of design basis accidents. Failure of the DGs will not cause a design basis accident to occur. However, failure of the DGs will affect the consequences of design basis accidents if a concurrent LNP occurs.

The proposed changes will revise the action requirements regarding operability testing of the DGs. The requirement to test the DGs if offsite circuits are inoperable will be deleted. An inoperable offsite circuit, by itself, will not affect the operability of the DGs. The requirement to test the remaining operable DG if one DG is inoperable will be modified. Testing will not be required provided a common cause failure is not the reason for declaring the DG inoperable. The requirement contained in the first footnote (*) to Technical Specification 3.8.1.1 to complete the test of the remaining DG will be deleted. The need to test the remaining DG will be based on the determination of a common cause failure. These changes will improve DG reliability by reducing the number of unnecessary starts and by requiring more appropriate testing of the DGs when there is a potential for common mode failure. The proposed changes to the action requirements will not change the response of the DGs to an LNP. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously analyzed.

The requirement contained in the second footnote (**) to Technical Specification 3.8.1.1 to allow a one time extension of the allowed outage time to 7 days will be deleted. This provision is no longer necessary since the Millstone Unit No. 1 work has been completed. The statements that a successful test of the DG performed for the current Action Statements c, d, or e will satisfy the required testing of Action States a or b are no longer necessary with the proposed changes. These statements will be deleted. The removal of these items will not change the response of the DGs to an LNP. Therefore, these proposed changes will not result in a significant increase in the probability or consequences of an accident previously analyzed.

The proposed changes to the DG surveillance requirements will allow an engine prelube period before all DG tests starts, allow slow starting of the DGs, and allow the DGs to be loaded in accordance with manufacturer recommendations. This will decrease the wear on the DGs. The proposed changes will also allow adequate time for the completion of all manufacturer recommended DG engine prelube procedures. Modifying starting and loading requirements, consistent with the manufacturer recommendations, is intended to enhance diesel reliability by minimizing severe test conditions which can lead to premature failures. In addition, specifying that the 184 day DG SRs [surveillance requirements] will satisfy the 31 day DG starting and loading SRs will eliminate redundant testing. These proposed changes will minimize unnecessary DG testing while maintaining DG reliability. The proposed changes will not change the response of the DGs to an LNP. Therefore, these changes will not result in a significant increase in the probability or consequences of an accident previously analyzed.

The ASTM [American Society for Testing and Materials] standards referenced for diesel fuel oil sampling will be modified in SR 4.8.1.1.2.b. The proposed changes will replace an outdated standard, and will remove the year of issuance or revision from the ASTM standards referenced. This will allow use of the current approved ASTM standard. These proposed changes do not affect the sampling frequency or acceptance criteria of this SR. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously analyzed.

The proposed wording changes to eliminate any possible confusion when SRs 4.8.1.1.1 and 4.8.1.1.2 are referenced by SR 4.8.1.2, to state that the DGs start from standby conditions instead of ambient conditions, and to remove the requirement to perform a DG surveillance only during shutdown will not affect any technical aspect of the SRs. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously analyzed.

SRs will be added to test the DGs every 184 days at conditions similar to the current 31 day SRs. These conditions are more restrictive than the new proposed 31 day SRs. The 184 day SRs will require the diesel generators to start and obtain speed and voltage within 15 seconds and will also require the diesel generators to be synchronized, loaded, and to maintain the load for at least 60 minutes. However, it will allow gradual loading, based on manufacturer recommendations, to be used. A 184 day surveillance interval is sufficient to verify DG fast-start capability, and is consistent with GL [Generic Letter] 84–15, GL 93–05, and NUREG–1432. Therefore, the posed changes will not result in a significant increase in the probability or consequences of an accident previously analyzed.

The list of SRs, contained in SR 4.8.1.2, that do not have to be performed for the operable diesel generator in Modes 5 and 6 will be expanded to take into account the 184 day DG SR that will be added. This proposed change will exclude the one operable DG from being loaded when the 184 day SR is performed. This is consistent with the current SR which excludes performance of SR 4.8.1.1.2.a.3. Loading the one required operable diesel generator could subject this diesel generator to grid faults which could adversely affect its ability to perform its safety function. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously analyzed.

The Bases of these Technical Specifications will be modified and expanded to discuss the proposed changes, and to provide guidance to ensure the requirements are correctly applied. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously analyzed.

These proposed changes do not alter the way any structure, system, or component functions. The intent of the proposed changes is to improve the reliability of the DGs by eliminating unnecessary surveillance testing and allowing most of the surveillance testing to be performed in accordance with the recommendations of the manufacturer. There will be no adverse effect on equipment important to safety. The response of the DGs to an LNP, as described in the Millstone Unit No. 2 FSAR [Final Safety Analysis Report], will remain the same. There will be no effect on any of the design basis accidents previously evaluated. Therefore, this License Amendment Request will not result in a significance increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This License Amendment Request proposes to modify the LCOs for electrical power sources, DG surveillance requirements and the required actions for inoperable electrical power sources contained in the Millstone Unit No. 2 Technical Specifications. The proposed changes will revise LCO wording to be consistent with the required offsite power distribution requirements and improve DG reliability by minimizing excessive wear of the DGs, and changing the starting and loading requirements of the DGs, in accordance with manufacturer recommendations, during most DG surveillance and operability tests. Improving the reliability of the DGs will help ensure the DGs will respond to an LNP as described in the Millstone Unit No. 2 FSAR. Therefore, this License Amendment Request will not result in a significant reduction in the margin of safety as defined in the Bases for the Technical Specifications addressed by the proposed changes.

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

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

NRC Deputy Director: Phillip F. McKee.

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

Date of amendment request: July 21, 1998.

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) by changing various Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) setpoints and allowable values; correct the specified maximum reactor power level limited by the high power level RPS trip; add new TS and requirements associated with the automatic isolation of steam generator blowdown; and make several editorial and changes to correct various errors and to provide needed clarification. The applicable TS Bases sections would also be changed to reflect the proposed changes, correct previous errors identified during the licensee's review of the TS, eliminate redundant information, and expand the TS Bases to discuss the new requirements for the automatic isolation of the steam generator blowdown.

Specifically, the proposed changes would modify TS 2.1.1, ''Safety Limits—Reactor Core,'' TS 2.2.1, ''Limiting Safety System Settings—Reactor Trip Setpoints,'' TS 3.3.1.1, ''Instrumentation—Reactor Protective Instrumentation'' TS 3.3.2.1, ''Instrumentation—Engineered Safety Features Actuation System Instrumentation,'' and would add a new TS 3.7.1.8, ''Plant Systems—Steam Generator Blowdown Isolation Valves.'' As previously noted, the applicable TS Bases sections will be updated to reflect the proposed changes.

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

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

The proposed change to correct the maximum reactor power level from 112% to 111.6% is consistent with the maximum high power trip setpoint of 106.6%, plus 5% uncertainty, currently used in the safety analyses. This does not change the Technical Specification required high power reactor trip setpoint. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to the trip setpoints and allowable values for the Reactor Protection System (RPS) trips on high pressurizer pressure, high containment pressure, low steam generator pressure, and low steam generator level are the result of revisions to the instrument loop uncertainty and setpoint calculations. These calculations were revised to incorporate calculation methodology changes, analytical limit changes, correct errors identified, and to include the effects of a harsh environment (pressure, temperature, and radiation), where appropriate. The proposed setpoints and allowable values will ensure a reactor trip signal is generated at, or before the analytical limits used in the respective accident analyses are reached. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to the trip setpoints and allowable values for the Engineered Safety Features Actuation System (ESFAS) actuations on low pressurizer pressure, high containment pressure, low steam generator pressure, low refueling water storage tank level, and low steam generator level are the result of revisions to the instrument loop uncertainty and setpoint calculations. These changes were revised to incorporate calculation methodology changes, analytical limit changes, correct errors identified, and to include the effects of a harsh environment (pressure, temperature, and radiation), where appropriate. The proposed setpoints and allowable values will ensure an ESF [engineered safety feature] actuation signal is generated at, or before the analytical limits used in the respective accident analyses are reached. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to add Technical Specification requirements for the steam generator blowdown isolation valves will provide additional assurance that the automatic isolation of steam generator blowdown will occur as assumed in the loss of main feedwater accident analysis. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to the value of steam generator pressure when the steam generator low pressure reactor trip can be bypassed (from 780 psia to 800 psia) will reduce the range of plant operation when this trip is required to be available. However, this will not affect the range of plant operation when this RPS trip is required to be operable. This RPS trip is required in Modes 1 and 2. The expected steam generator pressure during a reactor startup (entry into Mode 2) is approximately 900 psia, which corresponds to a Reactor Coolant System (RCS) temperature of approximately 532°F. The proposed change will require the bypass to be automatically removed prior to exceeding a steam generator pressure of 800 psia. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to the value of pressurizer pressure (from 1750 psia to 1850 psia) when the pressurizer low pressure ESF actuations (SIAS, CIAS, and EBFAS) [safety injection actuation system, containment isolation actuation system, and enclosure building filtration actuation system] can be blocked will reduce the range of plant operation when these functions are required to be available. However, since the plant would normally be in Mode 3 when pressurizer pressure is in this range, automatic actuation of these ESF functions on high containment pressure, as well as manual actuation, is required to be operable. In addition, the plant would not normally maintain pressurizer pressure between 1750 psia and 1850 psia. Therefore, since automatic actuation of these ESF functions on high containment pressure, as well as manual actuation, should be operable, and the time the plant will operate between 1750 psia and 1850 psia is small, the ESFAS will continue to function as before. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to the value of steam generator pressure (from 600 psia to 700 psia) when the steam generator low pressure ESF actuation (main steam line isolation) can be blocked will reduce the range of plant operation when this function is required to be available. However, since the plant would be in Mode 3 when steam generator pressure is in this range (RCS temperature of approximately 486°F to 503°F), automatic actuation of this ESF function on high containment pressure, as well as manual actuation, is required to be operable. In addition, the plant would not normally maintain steam generator pressure between 600 psia and 700 psia. Therefore, since automatic actuation of this ESF function on high containment pressure, as well as manual actuation, should be operable, and the time the plant will operate between 600 psia and 700 psia is small, the ESFAS will continue to function as before. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The minor editorial and non-technical changes to correct spelling errors, correct a capitalization error, add page amendment numbers, add the specific plant parameter (steam generator pressure) to use if an RPS or ESF function can be bypassed, change the value of the parameter (pressurizer pressure) used in action statements, and a ''[less than or equal to]'' symbol, change ''value'' to ''setpoint,'' and update the index will have no effect on plant operation. These changes will not result in any technical changes to the Millstone Unit No. 2 Technical Specifications. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to the Technical Specification Bases will incorporate the RPS and ESFAS setpoint changes, correct errors, eliminate redundant information, and expand the Bases to discuss the new requirements for steam generator blowdown isolation. These changes will have no effect on equipment operation. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes have no adverse effect on any of the design basis accidents previously evaluated and have no adverse effect on how the RPS and ESFAS function to mitigate the consequences of design basis accidents. Therefore, the license amendment request does not impact the probability of an accident previously evaluated nor does it involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes will correct the maximum reactor power level specified; change RPS trip setpoints, allowable values, and bypass setpoints; change ESFAS trip setpoints, allowable values, and block setpoint changes; add a new Technical Specification and additional requirements associated with the automatic isolation of steam generator blowdown; and make various minor editorial and non-technical changes. There will be no adverse effect on equipment important to safety. The RPS and ESFAS will continue to function as designed to mitigate the consequences of design basis accidents. Therefore, there will be no significant reduction of the margin of safety as defined in the Bases for the Technical Specifications affected by the proposed changes.

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

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

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

NRC Deputy Director: Phillip F. McKee.

Pennsylvania Power and Light Company, Docket No. 50–387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of amendment request: June 19, 1998.

Description of amendment request: The amendment to Unit 1 Technical Specifications (TS) involves the addition of a new section entitled ''Oscillation Power Range Monitoring (OPRM) Instrumentation'' and revisions to Section 3.4.1 ''Recirculation Loops Operating'' to remove the specifications related to thermal power stability which will not be required after the installation of the OPRM instrumentation. Unit 1 is currently operating under Interim Corrective Actions (ICAs) defined in TS 3.4.1 that specify restrictions on plant operation and actions by operators in response to instability events. The OPRM system provides an automatic long-term solution to the instability issue and eases the burden on the operator.

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 proposal does not involve an increase in the probability or consequences of an accident previously evaluated.

The OPRM most directly affects the APRM and LPRM portions of the Power Range Neutron Monitoring system. Its installation does not affect the operation of these sub-systems. None of the accidents or equipment malfunctions affected by these sub-systems are affected by the presence or operation of the OPRM.

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux changes. The APRM Fixed Neutron Flux-High function is capable of generating a trip signal to prevent fuel damage or excessive reactor pressure. For the ASME overpressurization protection analysis in FSAR Chapter 5, the APRM Fixed Neutron Flux-High function is assumed to terminate the main steam isolation valve closure event. The high flux trip, along with the safety/relief valves, limit the peak reactor pressure vessel pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis in Chapter 15 takes credit for the APRM Fixed Neutron Flux-High function to terminate the CRDA. The Recirculation Flow Controller Failure event (pump runup) is also terminated by the high neutron flux trip. The APRM Fixed Neutron Flux-High function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the Safety Limits (e.g., MCPR and Reactor pressure) being exceeded.

The installation of the OPRM equipment does not increase the consequences of a malfunction of equipment important to safety. The APRM and RPS systems are designed to fail in a tripped (fail safe) condition; the OPRM will have no affect on the consequence of the failure of either system. An inoperative trip signal is received by the RPS any time an APRM mode switch is moved to any position other than Operate, an APRM module is unplugged, the electronic operating voltage is low, or the APRM has too few LPRM inputs. These functions are not specifically credited in the accident analysis, but are retained for the RPS as required by the NRC approved licensing basis.

The OPRM allows operation under current operating conditions presently restricted by the current Technical Specifications by providing automatic suppression functions in the area of concern in the event an instability occurs. The consequences of any accident or equipment malfunction are not increased by operating under those conditions. Although protected by the OPRM from thermal-hydraulic core instabilities above 30% core power, operation under natural core recirculation conditions is not allowed. No accidents or transients of a type not analyzed in the FSAR are created by operating under these conditions with the protection of the OPRM system.

This change does not increase the probability of an accident as previously evaluated. The OPRM is designed and installed to not degrade the existing APRM, LPRM, and RPS systems. These systems will still perform all of their intended functions. The new equipment is tested and installed to the same or more restrictive environmental and seismic envelopes as the existing systems. The new equipment has been designed and tested to the electromagnetic interference (EMI) requirements of Reference 2, which assures correct operation of the existing equipment. The new system has been designed to single failure criteria and is electrically isolated from equipment of different electrical divisions and from non-1E equipment. The electrical loading is within the capability of the existing power sources and the heat loads are within the capability of existing cooling systems. The OPRM allows operation under operating conditions presently forbidden or restricted by the current Technical Specifications. No other transient or accident analysis assumes these operating restrictions.

Based upon the analysis presented above, PP&L concludes that the proposed action does not involve an increase in the probability or consequences of an accident previously evaluated.

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

This proposal does not create the probability of a new or different type of accident from any accident previously evaluated. The OPRM system is a monitoring and accident mitigation system that cannot create the possibility for an accident.

The OPRM will allow operation in conditions currently restricted by the current Technical Specifications. Although protected by the OPRM from thermal-hydraulic core instabilities above 30% core power, operation under natural circulation conditions is not allowed. No accidents or transients of a type not analyzed in the FSAR are created by operating under these conditions with the protection of the OPRM system. No new failure modes of either the new OPRM equipment or of the existing APRM equipment have been introduced. Quality software design, testing, implementation and module self-health testing provides assurance that no new equipment malfunctions due to software errors are created. The possibility of an accident of a new or different type than any evaluated previously is not created.

The new OPRM equipment is designed and installed to the same system requirements as the existing APRM equipment and is designed and tested to have no impact on the existing functions of the APRM system. Appropriate isolation is provided where new interconnections between redundant separation groups are formed. The OPRM modules have been designed and tested to assure that no new failure modes have been introduced.

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

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

There has been no reduction in the margin of safety as defined in the basis for the Technical Specifications. The OPRM system does not negatively impact the existing APRM system. As a result, the margins in the Technical Specifications for the APRM system are not impacted by this addition.

Current operation under the ICAs provides an acceptable margin of safety in the event of an instability event as the result of preventive actions and Technical Specification controlled response by the control room operators. The OPRM system provides an increase in the reliability of the protection of the margin of safety by providing automatic protection of the MCPR safety limit, while the protection burden is significantly reduced for the control room operators. This protection is demonstrated as described above, and in the NRC reviewed and approved Topical Reports NEDO–32465–A and CENPD–400–P–A.

Replacement of the ICA operating restrictions from Technical Specifications with the OPRM system does not affect the margin of safety associated with any other system or fuel design parameter.

Therefore, the change does not involve a 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: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701.

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: July 6, 1998

Description of amendment request: The proposed Technical Specification (TS) changes represent revisions to the Radiological Effluent Technical Specification (RETS) Section 3.5.b.1, ''Main Condenser Steam Jet Air Ejector (SJAE)'' and Table 3.10–1 ''Radiation Monitoring Systems that Initiate and/or Isolate Systems'' including associated TS Bases. The existing RETS for radiation monitoring instrumentation systems that initiate and/or isolate systems will be changed by adding Allowable Outage Times (AOTs) and incorporating editorial and administrative changes to clarify 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. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The inherent redundancy and reliability of the protective instrumentation trip systems ensure that the consequences of an accident are not significantly increased. In addition, the restrictive Allowable Outage Time (AOT) interval limits the probability of the protective instrument channel being unavailable and an accident requiring its function from occurring simultaneously. The requirement that the associated trip function maintains trip capability for selected instrumentation ensures that the protective instrumentation response will occur such that the consequences of an accident are not different from those previously evaluated. The proposed changes provide AOTs for test and repair of plant instrumentation. The changes do not introduce any new modes of plant operation, make any physical changes, or alter any operational setpoints. Therefore, the changes do not degrade the performance of any safety system assumed to function in the accident analysis. Consequently, there is no effect on the probability of occurrence of an accident.

Regarding the consequences of an accident, the GE Licensing Topical Reports (References 1 and 2) [GE Topical Report NEDC–31677P–A, ''Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation,'' July 1990 and GE Topical Report GENE–770–06–1–A, ''Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications,'' December 1992] conclude that the proposed AOT for the safety system instrumentation results in an insignificant change in the core damage frequency. The AOTs result in a slight increase in the unavailability of the safety functions. The overall effect on the probability of an accident is negligible. The NRC concurred in their SERs [safety evaluation reports] (References 3 and 4) [NRC Safety Evaluation Report, letter from Charles E. Rossi, NRC to S.D. Floyd, BWR Owners Group, ''General Electric Company Topical Report NEDC–31677P, Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation'', June 18, 1990 and NRC Safety Evaluation Report, letter from Charles E. Rossi, NRC to R.D. Binz, BWR Owners Group, ''General Electric Company Topical Report GENE–770–06–1, Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications,'' July 21, 1992] with this conclusion. Consequently, there is not a significant increase in the consequences of an accident.

Since the editorial and administrative items do not alter the meaning or intent of any requirements, they do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes to the protective instrumentation trip system specifications do not create the possibility of a new or different kind of accident because they do not introduce any new operational modes or physical modifications to the plant.

For systems with only one channel (Main Control Room Ventilation) or two-out-of-two logic system (SJAE Radiation Monitors) a six-hour surveillance AOT is being proposed and a repair time AOT is not allowed. This is consistent with GE Topical Reports referenced in current TS Bases 4.2 and STS [Standard Technical Specifications] and therefore, will not introduce a new or different kind of accident than previously evaluated.

Since the editorial and administrative items do not alter plant configurations or operating modes, they do not create the possibility of a new or different kind of accident.

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

The protective instrumentation surveillance requirements provide verification of the operability of the trip system instrumentation channels. In addition, the redundant channel that monitors the identical Trip Function maintains trip capability for the relatively short duration of the test or repair time period. This ensures that protective instrumentation reliability is maintained. The proposed change provides for a specific time period to perform required surveillances on instrument channels without trips present in associated trip systems. This time allotment tends to enhance the margin of safety by decreasing the probability of unnecessary challenges to safety systems and inadvertent plant transients. The evaluations presented in the referenced GE Licensing Topical Reports concluded that the overall effect of the proposed changes provides a net increase in plant safety.

The only action resulting from the proposed changes to RETS is to add AOTs for selected instrumentation. Spurious signals during testing could initiate plant transients. These transients are bounded by the current transient analysis. These tests do not subject the instruments to any conditions beyond their design specifications and are performed in accordance with approved testing standards. This testing ensures equipment operability by identifying degraded conditions, initiating corrective action and properly retesting them. Therefore, the proposed RETS do not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Project Director: S. Singh Bajwa, Director

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

Date of amendment request: June 25, 1998.

Description of amendment request: The proposed changes affect Technical Specification (TS) Surveillance Requirement 4.5.1.d.2.b by deleting the requirement to perform in-situ functional testing of the Automatic Depressurization System (ADS) safety relief valves (SRVs) during startup testing activities. The proposed changes also affect TS Surveillance Requirement 4.4.2.1.b such that the 18-month channel calibration for the SRV acoustic monitors will no longer require an exception to the provisions of TS 4.0.4, nor adjustments to SRV full open noise levels.

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

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

The proposed TS change does not involve any physical changes to plant structures, systems or components (SSC). The ADS will continue to function as designed. The ADS is an Emergency Core Cooling System (ECCS) designed to mitigate the consequences of an accident, and therefore, can not contribute to the initiation of any accident. The ADS utilizes five of the 14 main steam line SRVs as the primary method for depressurizing the reactor pressure vessel to permit low pressure core cooling capability in the event of a small break Loss-of-Coolant-Accident (LOCA) if the high pressure cooling systems (i.e., High Pressure Cooling Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems) fail to maintain adequate reactor vessel water level.

Deleting the TS surveillance requirements to perform the in-situ testing of the ADS/SRVs during startup, as proposed, should reduce the probability of an inadvertent opening of an SRV as discussed in Section 15.1.4 of the Hope Creek [Updated Final Safety Analysis Report] UFSAR since deleting this testing requirement will eliminate a known initiator of SRV pilot leakage and subsequent erosion. This proposed TS change will have a tendency to increase, rather than decrease, the reliability of the ADS/SRVs by eliminating the in-situ ADS functional startup testing. The probability of the ADS/SRVs to open on demand has been demonstrated to be extremely high and is not measurably improved through the in-situ ADS functional startup testing.

Using the provisions of 10CFR50.59, PSE&G will establish a method for performing SRV acoustic monitor channel calibration that does not require reactor steam pressure or SRV opening. This testing method will comply with the current TS definition of CHANNEL CALIBRATION. Since the notes associated with TS Surveillance Requirement 4.4.2.1 (providing a compliance exception to the provisions of TS 4.0.4 to allow for proper reactor steam pressure to perform the test and an allowance for noise level adjustments) are no longer needed, their removal will not affect plant operation or testing and will not involve an increase in the probability or consequences of an accident previously evaluated.

This proposed TS change will not increase the probability of occurrence of a malfunction of any plant equipment important to safety. Alternate testing methods at Hope Creek and at the offsite test facility adequately demonstrate proper ADS valve operation and assure that the valves will continue to function as designed. Existing surveillance testing and inspections of the ADS/SRVs at Hope Creek verify that the ADS initiation logic, solenoid valve operation, pneumatic gas supply integrity and air operator assembly (including pilot rod) will operate as designed. Offsite testing verifies pilot disc operation, setpoint calibration, stroke time and main valve disc operation.

Deleting the in-situ testing requirement, as proposed, will reduce the probability of increasing SRV leakage, which should reduce the probability of an inadvertent opening of an SRV. Therefore, any SRV pilot leakage that can be eliminated would reduce the probability of occurrence of a malfunction of that SRV. Deleting the ADS/SRV in-situ functional test will in no way increase any consequences of a malfunction of plant equipment important to safety. The consequences of a malfunction of an ADS/SRV as discussed in the Hope Creek UFSAR remain unchanged.

In addition, eliminating a known initiator of SRV leakage, as proposed in this TS change, would help reduce operator workarounds in the form of suppression pool cooling and letdown operation activities. As a result, this will reduce the unnecessary operation of the Residual Heat Removal (RHR) and its supporting systems.

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

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

The proposed TS changes do not involve any physical changes to plant SSC. The design and operation of the ADS/SRVs are not changed from that currently described in the UFSAR. The ADS will continue to function as designed to mitigate the consequences of an accident. No changes of any kind are being made to the valves, auxiliary components or ADS logic. Deleting the requirement to perform the ADS in-situ functional test during plant startup as proposed in this TS change request reduces the likelihood of an SRV developing a leak and degrading throughout the subsequent operating cycle. Therefore, there is no possibility that implementing this proposed TS change would create a different type of malfunction to the ADS/SRVs than any previously evaluated.

Eliminating the requirement to perform the in-situ testing of the ADS/SRVs during startup activities does not create a new or different type of accident than any previously evaluated. There is no accident scenario associated with testing the ADS/SRVs other than the inadvertent opening of a relief valve, which is currently discussed in Section 15.1.4 of the UFSAR. The proposed TS changes do not alter the conclusions described in the UFSAR regarding an inadvertent opening of an SRV. No new or different type of accident will be created as a result of these proposed changes.

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

Using the provisions of 10CFR50.59, PSE&G will establish a method for performing SRV acoustic monitor channel calibration that does not require reactor steam pressure or SRV opening. This testing method will comply with the current TS definition of CHANNEL CALIBRATION. Since the notes associated with TS Surveillance Requirement 4.4.2.1 (providing a compliance exception to the provisions of TS 4.0.4 to allow for proper reactor steam pressure to perform the test and an allowance to perform noise level adjustments) are no longer needed, their removal will not affect plant operation or testing and will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed TS change involves deleting the requirement to perform in-situ functional testing of the ADS/SRVs during startup activities. This testing imposes an unnecessary challenge on the ADS/SRVs and has been linked to SRV degradation (e.g., pilot valve and/or main valve leakage). This proposed TS change should reduce SRV leakage and improve ADS/SRV reliability by reducing the potential for spurious SRV actuation. Since ADS operability can be readily demonstrated with extremely high confidence by the existing surveillance tests and inspections performed for the ADS, there will be no reduction in any margin of safety resulting from this proposed TS change. Therefore, the proposed TS change does not involve a significant reduction in a margin of safety.

Using the provisions of 10CFR50.59, PSE&G will establish a method for performing SRV acoustic monitor channel calibration that does not require reactor steam pressure or SRV opening. This testing method will comply with the current TS definition of CHANNEL CALIBRATION. Since the notes associated with TS Surveillance Requirement 4.4.2.1 (providing a compliance exception to the provisions of TS 4.0.4 to allow for proper reactor steam pressure to perform the test and an allowance to perform noise level adjustments) are no longer needed, their removal will not affect plant operation or testing and will not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: February 18, 1998.

Description of amendment request: The proposed amendment would revise the Watts Bar Nuclear Plant (WBN) Technical Specifications (TS) and associated Bases to address a new condition (Condition B) and associated actions in which one train (consisting of two valves) of Steam Generator Atmospheric Dump Valves (ADVs), although functional, would be considered technically INOPERABLE in the event of one train of the auxiliary control air system (ACAS) was out of service. The action required for the new condition is to restore the ADV lines to OPERABLE status within 72 hours. In addition, the proposed amendment would make a correction to the required action for Condition B (new Condition C) to clarify that the required action for two or more inoperable ADV lines (with the exception of new Condition B) is to restore all but one ADV line to operable status. The current Required Action for Condition B incorrectly states that only one ADV line must be restored to operable status.

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

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

The addition of the 72 hour completion time and clarification to existing TS do not increase the probability of an accident previously evaluated since these changes do not result in hardware or procedural changes which will affect probability of occurrence of an accident. The probability of an accident occurring during the 72 hour period as compared to the 24 hour completion time currently in the TS remains small. Further, addition of the 72 hour completion time and clarification to existing TS does not increase the consequences of an accident previously evaluated since sufficient equipment and procedures remain available to mitigate accidents previously evaluated. With two ADVs inoperable under this LCO, two ADVs remain in service. As indicated in the Applicable Safety Analysis of the TS Basis, two valves are adequate to cool the unit to the RHR [residual heat removal] entry conditions subsequent to accidents accompanied by a loss of offsite power. In addition, as indicated in the background discussion of the Bases of 3.7.4, the ADVs can be operated by use of a bottled nitrogen system designed to open the valves in the event of loss of normal and emergency air supplies. The valves may also be operated manually by using the valve hand wheels. Consequently, the two inoperable ADVs under this LCO are still expected to remain functional and could be placed in service and used to cool the steam generators, if necessary, in the event of an accident. Based on the above, the addition of the 72 hour completion time and clarifications to existing TS in accordance with this proposed amendment do not significantly increase the probability or consequences of an accident previously evaluated.

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

The addition of the 72 hour completion time and clarifications to existing TS does not cause the initiation of any accident nor create any new credible limiting failure for safety-related systems and components. The change does not result in an event previously deemed incredible being made credible. As such, it does not create the possibility of an accident different than any evaluated in the FSAR [Final Safety Analysis Report]. The change has an insignificant effect on the ability of the safety-related systems to perform their intended safety functions. Although the period during which a safety-related function (ACAS air supply) is assumed inoperable is extended from 24 to 72 hours, sufficient remaining equipment (two ADVs supplied by the opposite train ACAS) is available to mitigate the limiting [steam generator tube rupture] SGTR accident, assuming no single failure occurs. Also, additional redundant and diverse equipment (normal control air, emergency bottled nitrogen, and the valve hand wheels) is available and expected to remain functional to ensure the ADVs accomplish their function following an accident. The change does not create failure modes that could adversely impact safety-related equipment. Therefore, the change will not create the possibility of a malfunction of equipment important to safety different than previously evaluated in the FSAR. Thus, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The TS currently allow two or more ADVs to be out of service for 24 hours, based on low probability of an event occurring during the period which would require use of the ADVs, and based on availability of the steam dump valves and the MSSVs [main steam safety valves]. Providing a 72 hour completion time specifically for loss of two ADV valves due to loss on one train of ACAS to the ADVs does not significantly reduce the margin of safety since the probability of an event occurring during the 72 hour period is still small, and the capability exists to use the inoperable ADVs by manually operating the valves using the valve hand wheels, or by connecting the valve nitrogen bottle system, which was designed to operate the valves upon loss of air. In addition, the MSSVs, and the condenser steam dump valves would normally also be available. Thus, the proposed change does not significantly reduce the margin of safety.

Further, the NRC staff notes that the proposed change to the TS action statement for two or more ADV lines inoperable to require restoration of all but one of the four ADV lines, instead of the previous requirement to restore only one ADV line to operable status, is more restrictive and more conservative than the action statement as currently written. The change also makes the action statement consistent with the existing TS Bases in Section B 3.7.4, Action B.1. Accordingly, the staff proposes to find that this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, and 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 and the staff's additional assessment as provided above, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: May 6, 1998.

Description of amendment request: The proposed amendment would modify the Watts Bar Nuclear Plant (WBN) Technical Specifications (TSs) by revising the allowed enrichment of fuel stored in the new fuel storage racks from 4.3 to 5.0 weight percent uranium-235 (U–235). The revision also places limitations on fuel storage locations that may be utilized in the storage racks and provides additional limits on k(effective) when flooded with unborated water.

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

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

The proposed change to the allowed enrichment of new fuel stored in the new fuel storage racks does not change the criticality potential with the proposed fuel arrangement requirements for the storage racks. The potential k_{eff} values are maintained the same as the current TS requirements. In addition, the storage racks are not modified and the processes for loading and unloading fuel in these racks and the controls for these racks remain the same except for the storage limitations dictated by the criticality analysis. Additional controls are required with appropriate verification to assure the fuel is stored within the analysis assumptions. Handling procedures contain additional steps to specifically verify prohibited cells remain empty after fuel movement. This verification assures that the probability of a criticality event is not increased by the enrichment change. Since the k_{eff} limits and operating processes are unchanged by the proposed revision, there is no increase in the probability of an accident previously evaluated. Likewise, there is no impact to the consequences of an accident or increase in offsite dose limits as a result of the proposed TS change because the criticality requirements are unchanged and plant equipment will be utilized and operated without change considering the fuel storage location limits imposed by this request.

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

As stated above, the plant equipment and operating processes will not be altered by the proposed TS change with the exception of allowed fuel storage locations in the new fuel storage racks. The limitations on acceptable fuel storage locations in the racks ensure that the k(effective) limits are maintained at the same limits as currently required. TVA has not postulated a criticality event at WBN for the spent or new fuel storage locations because the design of the associated storage racks, potential moderation, and TS allowable fuel enrichments do not support the potential for this condition. Therefore, this change does not create the potential for a new accident from any previously analyzed.

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

The proposed TS change maintains the existing requirements for criticality by utilizing limited storage locations in the new fuel pit storage racks. There is no change to operating practices associated with the use and control of these racks except for the storage limitations. For these reasons, there will be no reduction in the margin [of] the safety as a result of implementing the proposed TS change.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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, Ohio

Date of amendment request: July 13, 1998.

Description of amendment request: The proposed license amendment would revise Perry Nuclear Power Plant Technical Specification 3.4.4, ''Safety/ Relief Valves (S/RVs),'' by increasing the present [plus or minus] 1% tolerance on the safety mode lift setpoint for the safety/relief valves to [plus or minus] 3%. This change would be performed in accordance with General Electric Topical Report NEDC–31753P, ''BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report.''

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 identified.

The proposed change allows an increase in the as-found safety relief valve (SRV) safety mode setpoint tolerance, determined by test after the valves have been removed from service, from [plus or minus] 1% to [plus or minus] 3%. The proposed change does not alter the Technical Specification requirements on the nominal SRV safety mode lift setpoints, the SRV relief mode setpoints, the required frequency for the SRV lift setpoint tests, or the number of SRVs required to be operable. This change does not involve physical changes to the SRVs, nor does it change the operating characteristics or safety function of the SRVs.

Consistent with current requirements, this change continues to require that the SRVs be adjusted to within [plus or minus] 1% of their nominal lift setpoints following testing. This change does not change the behavior and operation of any SRV and therefore has no significant impact to reactor operation. It also has no significant impact on response to any perturbation of reactor operation including transients and accidents previously analyzed in the Updated Safety Analysis Report. In addition, this change does not change SRV actuation. Therefore, this change will not increase the probability of an accident previously evaluated.

Generic considerations related to the change in setpoint tolerance were addressed in NEDC–31753P, ''BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report,'' and were reviewed and approved by the NRC. The plant specific evaluations, required by the NRC's Safety Evaluation for NEDC–31753P and performed to support this proposed change, are contained in NEDC–32307P, ''Safety Review for PNPP Safety/Relief Valve Setpoint Tolerance Relaxation/Out-of-Service Analyses,'' dated May 1994. These analyses and evaluations show that there is adequate margin to the design core thermal limits and to the reactor vessel pressure limits using a [plus or minus] 3% SRV setpoint tolerance. They also show that operation of the high pressure injection systems will not be adversely affected; and the containment response from a loss of coolant accident will be acceptable.

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

The proposed change to allow an increase in the SRV safety mode setpoint tolerance from [plus or minus] 1% to [plus or minus] 3% does not alter the nominal SRV lift setpoints or the number of SRVs required to be operable. This change does not involve physical changes to the SRVs, nor does it change the operating characteristics or the safety function of the SRVs. The proposed change does not involve a physical alteration of the plant. No new or different equipment is being installed. The proposed change does not impact core reactivity nor the manipulation of fuel bundles. There is no alteration to the parameters within which the plant is normally operated. As a result no new failure modes are being introduced. There are no changes in the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered.

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 will not involve a significant reduction in the margin of safety.

The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed change does not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation. The proposed change does not significantly impact any safety analysis assumptions or results.

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 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: Ronald R. Bellamy (Acting).

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

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

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

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

Date of amendment request: July 8, 1998.

Description of amendment request: The proposed amendments would allow temporary noncompliance with the Penetration Room Ventilation System air flow surveillance requirements of Technical Specification 4.5.4.1.b.1 until modifications can be completed to support testing in accordance with ANSI Standard N510–1975, as required by the Technical Specifications.

Date of publication of individual notice in **Federal Register**: July 16, 1998 (63 FR 38433).

Expiration date of individual notice: August 17, 1998.

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

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

Date of application for amendment: June 18, 1998.

Brief description of amendment: Amend the Crystal River Unit 3 (CR3) Improved Technical Specifications to allow operation with a number of indications previously identified as tube end anomalies and multiple tube end anomalies in the CR3 Once Through Steam Generator tubes.

Date of publication of individual notice in the **Federal Register**: June 30, 1998 (63 FR 35615).

Expiration date of individual notice: July 15, 1998.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629.

Northern States Power Company, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of amendment request: June 19, 1998 (supersedes April 11, 1997, application), as supplemented July 1, 1998, and information provided in a letter of May 5, 1997.

Brief description of amendment request: The proposed amendment would revise Section 3.6.C, Coolant Chemistry, and 3/4.17.B, Control Room Emergency Filtration System, of the Technical Specifications (TS), Appendix A of the Operating License for the Monticello Nuclear Generating Plant. The changes were proposed to establish TS requirements consistent with modified analysis inputs used for the evaluation of the radiological consequences of the main steam line break accident. This amendment request was originally noticed in the **Federal Register** on May 6, 1998 (63 FR 25115). On June 19, 1998, supplemented July 1, 1998, the licensee submitted an application that superseded in its entirety the licensee's previous submittal dated April 11, 1997.

Date of publication of individual notice in **Federal Register**: July 28, 1998 (63 FR 40321).

Expiration date of individual notice: August 27, 1998.

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

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

Date of application for amendment: February 24 1998, as supplemented by letter dated May 27, 1998.

Brief description of amendment: The amendment would support a modification to the Callaway Plant, Unit 1 to increase the storage capacity of the spent fuel pool.

Date of individual notice in **Federal Register**: July 13, 1998 (63 FR 37598).

Expiration date of individual notice: August 12, 1998.

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

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

Date of amendment request: March 20, 1998, as supplemented by letter dated May 28, 1998.

Brief description of amendment: The amendment would support a modification to the Wolf Creek Nuclear Generating Station, Unit 1 to increase the storage capacity of the spent fuel pool.

Date of individual notice in **Federal Register**: July 13, 1998 (63 FR 37601).

Expiration date of individual notice: August 12, 1998.

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

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.

Boston Edison Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: February 20, 1998.

Brief description of amendment: This amendment changed the Pilgrim Nuclear Power Station Technical Specification (TS) 3/4.5.B and its Bases to incorporate the ultimate heat sink (UHS) temperature of 75 °F, as required by Amendment No. 173. The introduction of a UHS temperature restriction requires new specifications, actions, and surveillances for the salt service water system. The amendment also replaced existing specification 3.5.B ''Containment Cooling System'' with new Specification 3/4.5.B.1 ''Residual Heat Removal (RHR) Suppression Pool Cooling'', 3/4.5.B.2 ''Residual Heat Removal (RHR) Containment Spray'', 3/4.5.B.3 ''Reactor Building Closed Cooling Water (RBCCW) System'', and 3/4.5.B.4 ''Salt Service Water (SSW) System and Ultimate Heat Sink (UHS)''.

Date of issuance: July 28, 1998.

Effective date: July 28, 1998.

Amendment No.: 176.

Facility Operating License No. DPR–35: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 8, 1998 (63 FR 17221).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 28, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

Boston Edison Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: September 19, 1997, as supplemented June 15, 1998.

Brief description of amendment: The amendment relocates the Radioactive Effluent Technical Specifications and the Radiological Environmental Monitoring Program to the Offsite Dose Calculation Manual, in accordance with the recommendations of Generic Letter 89–01. Changes are also being made to other sections of the Technical Specifications to align them with NUREG–1433, to minimize changes when converting to the Improved Standard Technical Specifications.

Date of issuance: July 31, 1998.

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

Amendment No.: 177.

Facility Operating License No. DPR–35: Amendment revised the Technical Specifications and the license.

Date of initial notice in **Federal Register**: February 25, 1998 (63 FR 9591).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 31, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

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

Date of application for amendment: June 26, 1998, as supplemented July 22, 1998.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.7.8, ''Ultimate Heat Sink (UHS),'' to permit an 8-hour delay in the UHS temperature restoration period prior to entering the plant shutdown required actions. This TS amendment is given as a one-time amendment change effective until September 30, 1998, after which the TS will revert back to the original TS provisions.

Date of issuance: July 29, 1998.

Effective date: July 29, 1998.

Amendment No.: 179.

Facility Operating License No. DPR–23. Amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (63 FR 36967 dated July 8, 1998). 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 August 7, 1998, 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 July 29, 1998.

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

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

Commonwealth Edison Company, Docket Nos. 50–295 and 50–304, Zion Nuclear Power Station Units 1 and 2, Lake County, Illinois

Date of application for amendments: March 30, 1998.

Brief description of amendments: The amendments will (1) restore Custom Technical Specifications (CTS) and the associated license conditions that had been replaced by Improved Technical Specifications (ITS), (2) change certain management titles and responsibilities to reflect the permanently shutdown condition of the plant, (3) allow use of Certified Fuel Handlers in lieu of licensed operators, (4) modify shift crew composition, and (5) eliminate verbiage that imples the units are operational.

Date of Issuance: July 24, 1998.

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

Amendment Nos.: 179 & 166.

Facility Operating License Nos. DPR–39 and DPR–48: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: May 6, 1998 (63 FR 25105). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 24, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085.

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

Date of application for amendment: June 6, 1997, as supplemented September 25, 1997.

Brief description of amendment: The amendment revises Technical Specifications (TS) Table 4.1–2, Frequency for Sampling Tests, to delete the requirement to sample the spray additive tank and delete the requirement for a sodium hydroxide (NaOH) spray additive in TS Section 5.2.C.1.

Date of issuance: July 29, 1998.

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

Amendment No.: 197.

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

Date of initial notice in **Federal Register**: January 28, 1998 (63 FR 4310).

The September 25, 1997, letter provided clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 29, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 3, 1998, as supplemented by letters dated April 24, May 7, and July 22, 1998.

Brief description of amendments: The amendments revise Figure 5.1–1 of the Technical Specifications (TS) to show the new location of the meteorological tower. The meteorological tower will be relocated to a new location to facilitate use of the current location as a construction site. The proposed TS change does not change the related TS Section 5.1.1.

Date of issuance: July 30, 1998.

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

Amendment Nos.: Unit 1—179; Unit 2—161.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 29, 1998 (63 FR 35293).

The July 22, 1998, submittal provided clarifying information that did not change the scope of the March 3, 1998, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 30, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina.

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

Date of application for amendments: June 19, 1998, as supplemented June 23, 1998.

Brief description of amendments: These amendments revise the BVPS–1 and BVPS–2 Technical Specifications (TSs) definitions of a channel calibration to add two sentences stating that (1) the calibration of instrument channels with resistance temperature detector or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel and (2) whenever a sensing element is replaced, the next required channel calibration shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. This change makes the BVPS–1 and BVPS–2 TS definition of channel calibration consistent with the definition of a channel calibration contained in the NRC's improved Standard Technical Specifications for Westinghouse Plants (NUREG–1431, Revision 1).

Date of Issuance: July 28, 1998.

Effective date: Both units, effective immediately, to be implemented within 30 days.

Amendment Nos.: 216 and 93.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 26, 1998 (63 FR 34939).

The June 23, 1998, letter provided minor editorial changes to the TS pages that did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the June 26, 1998 **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 28, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

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

Date of application for amendment: March 20, 1998, and supplemented May 22, 1998.

Brief description of amendment: The amendment proposed to revise Improved Technical Specification Safety Limits and Administrative Controls to replace the titles of the Senior Vice President, Nuclear Operations and the Vice President, Nuclear Production with the position of Chief Nuclear Officer.

Date of issuance: July 20, 1998.

Effective date: July 20, 1998.

Amendment No.: 168.

Facility Operating License No. DPR–72: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: May 6, 1998 (63 FR 25109).

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

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

Date of application for amendment: December 29, 1997, as supplemented by June 15, 1998.

Brief description of amendment: The amendment will modify the Technical Specifications for selected cycle-specific reactor physics parameters to refer to the St. Lucie Unit 2 Core Operating Limits Report for limiting values.

Date of Issuance: July 24, 1998.

Effective Date: July 24, 1998.

Amendment No.: 92.

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

Date of initial notice in **Federal Register**: February 11, 1998 (63 FR 6985).

The June 15, 1998, supplement provided clarifying information that did not change the scope of the December 29, 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 July 24, 1998.

No significant hazards consideration comments received: No.

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

Nebraska Public Power District, Docket No. 50–298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: February 10, 1997, as supplemented December 26, 1997, and July 16, and July 28, 1998.

Brief description of amendment: The amendment revised the Technical Specifications to reflect the adoption of the BWR Owner's Group Long-Term Solution Stability System Option 1–D in addressing reactor operation in or near a region of potential thermal hydraulic instability.

Date of issuance: July 29, 1998.

Effective date: July 29, 1998, to be implemented within 30 days.

Amendment No.: 177.

Facility Operating License No. DPR–46: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 26, 1997 (62 FR 14462).

The December 26, 1997, July 16, and July 28, 1998, submittals provided clarifying information and an administrative change that did not alter the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 29, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Auburn Memorial Library, 1810 Courthouse Avenue, Auburn, NE 68305.

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

Date of application for amendments: January 15, 1998, as supplemented May 29, 1998.

Brief description of amendments: The amendment allows a reduction in the required number of incore instrumentation detectors for the remainder of Unit 1, Cycle 19 operation.

Date of issuance: July 28, 1998.

Effective date: July 28, 1998, with full implementation within 30 days.

Amendment Nos.: 136.

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

Date of initial notice in **Federal Register**: January 30, 1998 (63 FR 4676) The May 29, 1998, supplement provided clarifying information within the scope of the **Federal Register** notice 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 July 28, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 12, 1997.

Brief description of amendment: The amendment revises the working hours for operating personnel to allow 8- to 12-hour work days, nominal 40-hour weeks. In addition, associated changes are being made to surveillance intervals to maintain the same frequency.

Date of issuance: July 24, 1998.

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

Amendment No.: 244.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 28, 1998 (63 FR 4321).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 24, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 31, 1997, as supplemented June 18, 1997, October 10, 1997, October 20, 1997, November 11, 1997, December 22, 1997, January 15, 1998, January 27, 1998, March 30, 1998, April 23, 1998, April 27, 1998, May 8, 1998, and May 22, 1998.

Brief description of amendment: This amendment changes the Technical Specifications to accommodate the modification of the spent fuel pool by replacing the three Region 1 rack modules with seven new borated stainless steel rack modules scheduled for implementation in 1998. Six new peripheral modules would be added at some future date. Two of the seven new modules planned to be installed in 1998 are to be designated as part of Region 2, effectively increasing the Region 2 area. The other five new modules compose Region 1, resulting in a total of 294 storage positions in Region 1. Region 2, with 1075 storage positions, consists of three rack types, Type 1, Type 2, and Type 4. Type 1 cells are the Boraflex cells that form Region 2 for the existing license. Two racks of Type 2 cells, containing borated stainless steel (BSS) absorber plates are be added to increase the storage capacity of Region 2. In addition, the capacity of Region 2 could be increased in the future by the addition of Type 4 racks, which also contain BSS absorber plates. The amendment increases the boron concentration from 300 ppm to 2300 ppm.

Date of issuance: July 30, 1998.

Effective date: July 30, 1998.

Amendment No.: 72.

Facility Operating License No. DPR–18: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 30, 1998 (63 FR 35617).

The May 8 and 22, 1998, letters provided clarifying information that did not change the proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 30, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

Southern Nuclear Power Company, Inc., et al. Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant (VEGP), Units 1 and 2, Burke County, Georgia

Date of application for amendments: May 8, 1998.

Brief description of amendments: The amendments revise VEGP Technical Specification 5.5.7, ''Reactor Coolant Pump Flywheel Inspection Program,'' to provide an exception to the examination requirements of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, dated August 1975.

Date of issuance: July 21, 1998.

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

Amendment Nos.: Unit 1—103; Unit 2—81.

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

Date of initial notice in **Federal Register**: June 17, 1998 (63 FR 33108).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 21, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 25, 1998 (TS 97–06).

Brief description of amendments: The amendments change the Technical Specifications (TS) by revising the surveillance requirements for the emergency diesel generators.

Date of issuance: July 22, 1998.

Effective date: To be implemented no later than 45 days after issuance.

Amendment Nos.: Unit 1—234; Unit 2—224.

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

Date of initial notice in **Federal Register**: April 8, 1998 (63 FR 17235).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 22, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: May 2, 1995, as supplemented October 12, 1995, March 26, 1996, December 15, 1997, and May 27, 1998 (TSCR 172).

Brief description of amendments: These amendments revise the Technical Specifications (TS) Table 15.4.1–1, ''Minimum Frequencies For Checks, Calibrations, and Tests Of Instrument Channels,'' to change the test frequency of the containment high range radiation monitor, revise note 7, and revise item 36 to clarify which monitors in the radiation monitoring system support current TS or meet the requirements of 10 CFR 50.36. In addition several administrative changes to referenced TS sections and plant system titles were made to correct omissions from previous amendments.

Date of issuance: July 17, 1998.

Effective date: July 17, 1998. The TS are to be implemented within 45 days from the date of issuance. Implementation shall also include relocation of certain TS requirements to licensee-controlled documents, as described in the licensee's application dated May 2, 1995, as supplemented October 12, 1995, March 26, 1996, December 15, 1997, and May 27, 1998, and evaluated in the staff's safety evaluation attached to these amendments.

Amendment Nos.: 185 and 189.

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: May 6, 1998 (63 FR 25122). The May 27, 1998, submittal provided additional clarifying information and updated TS pages. This information was within the scope of the original **Federal Register** notice and did not change the staff's initial no significant hazards considerations determination.

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

No significant hazards consideration comments received: No.

Local Public Document Room location: The Lester Public Library, 1001 Adams Street, Two Rivers, Wisconsin 54241.

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

Date of application for amendments: May 15, 1998 (TSCR 205, NPL–98–0303).

Brief description of amendment: This amendment revises the schedule for implementing the boron concentration changes from refueling outage 24 to refueling outage 23 for the planned conversion of Unit 2 to 18-month fuel cycles.

Date of issuance: July 21, 1998.

Effective date: July 21, 1998, with full implementation within 45 days.

Amendment No.: 190.

Facility Operating License No. DPR–27: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 17, 1998 (63 FR 33111).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 21, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: The Lester Public Library, 1001 Adams Street, Two Rivers, Wisconsin 54241.

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

Date of amendment request: July 17, 1998.

Brief description of amendment: The amendment revised Technical Specification 3/4.7.5, Ultimate Heat Sink, by adding a new Action Statement to be used in the event that plant inlet water temperature exceeds 90 degrees F.

Date of issuance: July 18, 1998.

Effective date: July 18, 1998.

Amendment No.: 118.

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

Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated July 18, 1998.

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

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

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

Date of amendment request: March 24, 1995, as supplemented by letters dated July 26, 1995, and September 5, 1996.

Brief description of amendment: The amendment adds a new action statement to Technical Specification (TS) 3.5.1 which provides a 72-hour allowed outage time (AOT) for one accumulator to be inoperable because its boron concentration did not meet the 2300–2500 parts per million band. In addition, TS surveillance requirements are changed to incorporate the guidance of Generic Letter 93–05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Operation'' that is applicable to the accumulators, and the TS Bases section for TS 3/4.5.1 is revised to reflect the changes described above. Instrumentation surveillance requirements associated with the accumulator are being relocated from the technical specifications to Chapter 16 of the Updated Safety Analysis Report.

Date of issuance: July 21, 1998.

Effective date: July 21, 1998, to be implemented within 30 days from the date of issuance.

Amendment No.: 119.

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

Date of initial notice in **Federal Register**: April 12, 1995 (60 FR 18632).

The July 26, 1995, and September 5, 1996, supplemental letters provided additional clarifying information and did not change the initial no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 21, 1998.

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 5th day of August 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–21724 Filed 8–11–98; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[Rel. No. IC–23380; 812–11216]

CIBC Oppenheimer Corp.; Notice of Application

August 5, 1998.

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

ACTION: Notice of application for an order under section 12(d)(1)(J) of the Investment Company Act of 1940 (the ''Act'') for an exemption from section 12(d)(1) of the Act, under section 6(c) of the Act for an exemption from section 14(a) of the Act, and under section 17(b) of the Act for an exemption from section 17(a) of the Act.

SUMMARY OF APPLICATION: CIBC Oppenheimer Corp. (''CIBC'') requests an order with respect to the REDSS trusts (''REDSS Trusts'') and future trusts that are substantially similar to the REDSS Trusts and for which CIBC will serve as a principal underwriter (collectively, the ''Trusts'') that would (i) permit other registered investment companies, and companies excepted from the definition of investment company under section 3(c)(1) or (c)(7) of the Act, to own a greater percentage of the total outstanding voting stock (the ''Securities'') of any Trust than that permitted by section 12(d)(1), (ii) exempt the Trusts from the initial net worth requirements of section 14(a), and (iii) permit the Trusts to purchase U.S. government securities from CIBC at the time of a Trust's initial issuance of Securities.

FILING DATES: The application was filed on July 8, 1998.

Hearing or Notification of Hearing: An order granting the application will be issued unless the SEC orders a hearing. Interested persons may request a hearing by writing to the SEC's Secretary and serving CIBC with a copy of the request, personally or by mail. Hearing requests should be received by the SEC by 5:30 p.m. on August 31, 1998, and should be accompanied by proof of service on CIBC, in the form of an affidavit, or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons may request notification of a hearing by writing to the SEC's Secretary.

ADDRESSES: Secretary, SEC, 450 Fifth Street, NW, Washington, DC 20549. CIBC Oppenheimer Corp., CIBC Oppenheimer Tower, World Financial Center, New York, New York 0281. Copy to Thomas A. McGavin, Jr., Esq., Rogers & Wells LLP, 200 Park Avenue, New York, New York 10166.

FOR FURTHER INFORMATION CONTACT: Brian T. Hourihan, Senior Counsel, at (202) 942–0526, or Mary Kay Frech, Branch Chief, at (202) 942–0564 (Division of Investment Management, Office of Investment Company Regulation).

SUPPLEMENTARY INFORMATION: The following is a summary of the application. The complete application may be obtained for a fee from the SEC's Public Reference Branch, 450 Fifth Street, NW, Washington, DC. 20549 (tel. (202) 942–8090).

Applicant's Representations

1. Each Trust will be a limited-life, grantor trust registered under the Act as a non-diversified, closed-end management investment company. CIBC will serve as a principal underwriter (as defined in section 2(a)(29) of the Act) of the Securities issued to the public by each Trust.

2. Each Trust will, at the time of its issuance of Securities, (i) enter into one or more forward purchase contracts (the ''Contracts'') with a counterparty to purchase a formulaically-determined number of a specified equity security or securities (the ''Shares'') of one specified issuer,[1] and (ii) in some cases, purchase certain U.S. Treasury securities (''Treasuries''), which may include interest-only or principal-only securities maturing at or prior to the Trust's termination. The Trusts will purchase the Contracts from counterparties that are not affiliated

[1] Initially, no Trust will hold Contracts relating to the Shares of more than one issuer. However, if certain events specified in the Contracts occur, such as the issuer of Shares spinning-off securities of another issuer to the holders of the Shares, the Trust may receive shares of more than one issuer at the termination of the Contracts.