INDIVIDUALS RECEIVING ADVANCE NOTIFICATION OF NUCLEAR WASTE SHIPMENTS—Continued

| State | Part 71 | Part 73 |
|---|---|---|
| RHODE ISLAND ...... | William A. Maloney, Associate Administrator, Motor Carriers Section, Division of Public Utilities and Carriers, 100 Orange Street, Providence, RI 02903, (401) 222–3500; ext. 150. | Same. |
| SOUTH CAROLINA | Virgil R. Autry, Director, Division of Radioactive Waste Management, Bureau of Land and Waste Management, Department of Health and Environmental Control, 2600 Bull Street, Columbia, SC 29201, (803) 896–4244, Emergency: (803) 253–6488. | Same. |
| SOUTH DAKOTA ..... | Gary N. Whitney, Director, Division of Emergency Management, 500 E. Capitol Avenue, Pierre, SD 57501–5060, (605) 773–3231. | Same. |
| TENNESSEE ............ | John D. White, Jr., Director, Emergency Management Agency, 3041 Sidco Drive, Nashville, TN 37204–1504, (615) 741–0001, After hours: (Inside TN) 1–800–262–3300, (Outside TN) 1–800–258–3300. | Same. |
| TEXAS ...................... | Richard A. Ratliff, Chief, Bureau of Radiation Control, Texas Department of Public Health, 1100 West 49th Street, Austin, TX 78756, (512) 834–6688. | Col. Dudley Thomas, Director, Texas Department of Health Safety, Attn: EMS Tech. Hazards, P.O. Box 4087, Austin, TX 78773–0001, (512) 424–2429, (512) 424–2277 (24 hrs) |
| UTAH ........................ | William J. Sinclair, Director, Division of Radiation Control, 168 North 1950 West, P.O. Box 144850, Salt Lake City, UT 84114–4850, (801) 536–4250, After hours: (801) 536–4123. | Same. |
| VERMONT ............... | Lieutenant Col. John H. Sinclair, Director, Division of State Police, Department of Public Safety, 103 South Main Street, Waterbury, VT 05671–2101, (802) 244–7345. | Same. |
| VIRGINIA .................. | L. Ralph Jones, Jr., Director, Technological Hazards Division, Department of Emergency Services, Commonwealth of Virginia, 10501 Trade Court, Richmond, VA 23236, (804) 897–6570. | Same. |
| WASHINGTON ......... | Lieutenant Gail R. Otto, Washington State Patrol, P.O. Box 42600, Olympia, WA 98504–2600, (360) 753–0565, After hours (253) 536–6210 (ext. 0). | Same. |
| WEST VIRGINIA ...... | Colonel Gary L. Edgell, Superintendent, West Virginia State Police, 725 Jefferson Road, South Charleston, WV 25309, (304) 746–2111. | Same. |
| WISCONSIN ............. | Steven D. Sell, Administrator, Wisconsin Division of Emergency Management, P.O. Box 7865, Madison, WI 53707–7865, (608) 242–3232. | Same. |
| WYOMING ............... | Captain L. S. Gerard, Motor Carrier Officer, Wyoming Highway Patrol, 5300 Bishop Boulevard, P.O. Box 1708, Cheyenne, WY 82003–1708, (307) 777–4317, 24 hours: (307) 777–4321. | Same. |
| DISTRICT OF COLUMBIA. | Norma J. Stewart, Chief, Bureau of Food, Drug & Radiation Protection, Department of Health, 825 North Capitol St., NE, Room 5125, Washington, DC 20002, (202) 442–5919. | Same. |
| PUERTO RICO ........ | Hector Russe Martinez, Chairman, Environmental Quality Board, P.O. Box 11488, San Juan, PR 00910, (787) 767–8056 or (787) 767–8181. | Same. |
| GUAM ....................... | Jesus T. Salas, Administrator, Guam Environmental Protection Agency, P.O. Box 22439 GMF, Barrigada, Guam 96921, (671) 475–1658/9. | Same. |
| VIRGIN ISLANDS .... | Charles Turnbull, Governor, Governor's Office 21–22 Kongens Gade, St. Thomas, Virgin Islands 00802, (809) 774–0001. | Same. |
| AMERICAN SAMOA | Pati Faiai, Government Ecologist, Environmental Protection Agency, Office of the Governor, Pago Pago, American Samoa 96799, (684) 633–2304. | Same. |
| COMMONWEALTH OF THE NORTHERN MARIANA ISLANDS. | Joaquin A. Tenorio, Ph.D., Secretary, Department of Lands and Natural Resources, Commonwealth of Northern Mariana Islands Government, Saipan, MP 96950, (670) 322–9830 or (670) 322–9834. | Same. |

Questions regarding this matter should be directed to Spiros Droggitis, Office of State Programs, U.S. Nuclear Regulatory Commission, Washington, DC 20555, (Internet Address: SCD@NRC.GOV) or at (301) 415–2367.

Dated at Rockville, Maryland this 21st day of June , 1999.

For the Nuclear Regulatory Commission.

**Paul H. Lohaus,**

*Director, Office of State Programs.*

[FR Doc. 99–16393 Filed 6–29–99; 8:45 am]

BILLING CODE 7590–01–P

## NUCLEAR REGULATORY COMMISSION

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

### I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from June 5, 1999, through June 18, 1999. The last biweekly notice was published on June 16, 1999 (64 FR 32284).

**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 July 30, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention:

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

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

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

*Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units Nos. 1, 2, and 3, Maricopa County, Arizona*

*Date of amendments request:* May 26, 1999.

*Description of amendments request:* The proposed amendment would revise Technical Specification 3.3.1, ''Reactor Protective System (RPS) Instrumentation—Operating,'' to change the RPS reactor coolant flow trip setpoints. The change is intended to reduce spurious reactor trip hazards.

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

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

No. The proposed change will change the Reactor Protection System (RPS) reactor coolant flow trip setpoints. The RPS functions to mitigate the consequences of an accident. The changes to the low reactor coolant flow trip setpoints will reduce or eliminate unnecessary challenges to the RPS. Therefore, the proposed change will not involve a significant increase in the probability of an accident previously evaluated.

These changes will result in an increased time delay for the RPS low reactor coolant flow trip. The reanalysis of the affected UFSAR [updated final safety analysis report] Chapter 15 events (UFSAR 15.3.4, Reactor Coolant Pump Shaft Break with Loss of Offsite Power and UFSAR 15.1.5, Steam System Piping Failures Inside and Outside Containment—Modes 1 and 2 Operations), with the increased time delay, shows that the dose consequences for these events remain bounded by the UFSAR analysis. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

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

No. The proposed change will change the RPS reactor coolant flow trip setpoints. The RPS functions to mitigate the consequences of an accident. The changes to the low reactor coolant flow trip setpoints will reduce or eliminate unnecessary challenges to the RPS. The proposed change only changes the mitigating actions of the RPS, without changing the required function of the RPS. Therefore, the change to the low reactor coolant flow trip setpoints does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed change will change the RPS reactor coolant flow trip setpoints. The reanalysis of the affected UFSAR Chapter 15 events (UFSAR 15.3.4, Reactor Coolant Pump Shaft Break with Loss of Offsite Power and UFSAR 15.1.5, Steam System Piping Failures Inside and Outside Containment—Modes 1 and 2 Operations), with the revised reactor coolant flow trip setpoints, shows that the minimum DNBR [departure from nucleate boiling ratio] and SAFDLs [specified acceptable fuel design limits] for these events remain bounded by the UFSAR analysis. 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 that review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involve no significant hazards consideration.

*Local Public Document Room location:* Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004.

*Attorney for licensee:* Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999

*NRC Section Chief:* Stephen Dembek.

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

*Date of amendment request:* June 2, 1999.

*Description of amendment request:* The proposed amendment would relocate Shearon Harris Nuclear Power Plant (HNP) Technical Specification (TS) Section 6.5, ''Review and Audit,'' TS 6.8.2, TS 6.8.3, and TS Section 6.10, ''Record Retention,'' intact from the HNP TS to the Quality Assurance Program Description currently located in the HNP Final Safety Analysis Report Section 17.3. Future changes to the associated relocated TS would be processed in accordance with 10 CFR 50.54(a). The proposed change is consistent with NUREG–1431, Revision 1, ''Standard Technical Specifications, Westinghouse Plants,'' dated April 1995, and with the guidance provided in NRC Administrative Letter 95–06, ''Relocation of Technical Specification Administrative Controls related To Quality Assurance,'' dated December 12, 1995.

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

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

This TS change relocates administrative requirements from HNP TS to the Quality Assurance Program Description (QAPD). The proposed amendment will not introduce any new equipment or require existing equipment to function different from that previously evaluated in the Final Safety Analysis Report (FSAR) or TS.

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

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

The proposed amendment will not introduce any new equipment or require existing equipment to function different from that previously evaluated in the Final Safety Analysis Report (FSAR) or TS. The changes are consistent with NUREG–1431, Revision 1 and the Commission's Final Policy Statement on Technical Specification improvements. The proposed amendment will not create any new accident scenarios, because the change does not introduce any new single failures, adverse equipment or material interactions, or release paths.

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 amendment does not involve a significant reduction in the margin of safety.

This TS change relocates administrative requirements from HNP TS to the Quality

Assurance Program Description (QAPD). The QAPD will be revised to include the requirements associated with this proposed change. NRC Administrative Letter 95–06 states that administrative requirements for review and audit and the independent safety engineering group may be relocated from TS to the quality assurance program. HNP proposes relocating the associated requirements from TS to the QAPD intact. Future changes to these requirements will be processed in accordance with 10 CFR 50.54(a). This proposed TS change is administrative in nature and does not alter NRC acceptance limits with respect to accident mitigation or accident analysis.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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:* Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

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

*NRC Section Chief:* Sheri R. Peterson.

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

*Date of amendment request:* July 22 and October 22, 1998; May 6, 1999.

*Description of amendment request:* The amendments would revise the Technical Specifications (TS) to reflect the licensee's planned use of fuel supplied by Westinghouse. The staff has published a Notice of Consideration of Issuance of Amendments and Proposed No Significant Hazards Consideration Determination on November 3, 1998 (63 FR 69338) covering the July 22 and October 22, 1998, submittals. In the May 6, 1999, submittal the licensee proposed to expand the original amendment request, revising Section 5.6.5 of the Technical Specifications. Section 5.6.5 specifies a list of NRC-approved topical reports that the licensee is required to use to determine reactor core operating limits. The licensee proposed to update this list to show the current approval status of these topical reports.

*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 for the proposed changes conveyed by the May 6, 1999, submittal. The NRC staff has reviewed the licensee's analyses against the standards of 10 CFR 50.92(c). The NRC staff's analysis is presented below.

*First Standard*

No. The proposed changes to Section 5.6.5 will not affect the safety function, and will not involve any change to the design or operation of any plant system or component. The topical reports were previously approved by the NRC staff under separate licensing actions. The use of methodologies in these approved topical reports will ensure that previously evaluated accidents remain bounding. Therefore, no accident probabilities or consequences will be impacted.

*Second Standard*

No. The proposed changes would not lead to any hardware or operating procedure change. Hence no new equipment failure modes or accidents from those previously evaluated will be created.

**Third Standard**

No. Margin of safety is associated with confidence in the design and operation of the plant; specifically, the ability of the fission product barriers to perform their design functions during and following an accident. The proposed changes to Section 5.6.5 do not involve any change to plant design, operation, or analysis. Thus the margin of safety previously analyzed and evaluated is maintained.

Based on this analysis, it appears that the three standards of 10 CFR 50.92(c) are satisfied for the proposed changes to Section 5.6.5. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

*Attorney for licensee:* Mr. Albert Carr, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina

*NRC Section Chief:* Richard L. Emch, Jr.

*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:* April 5, 1999, supplemented May 27, 1999.

*Description of amendment request:* The proposed amendments would revise the Improved Technical Specifications (TS), Updated Final Safety Analysis Report, and Core Operating Limits Report to incorporate Topical Report (TR) DPC–NE–3005–P, ''Thermal-Hydraulic Transient Analysis Methodology.'' This analysis has been completed for Unit 2 and is ongoing for Units 1 and 3. Therefore, the proposed changes that reflect the TR provisions affect Unit 2 only. Other proposed changes affect all three units. Specifically, (1) a note to TS Surveillance Requirement (SR) 3.4.1.2, ''RCS [Reactor Coolant System] Pressure, Temperature, and Flow DNB [Departure from Nucleate Boiling] Limits,'' would be modified to address application of the delta-$T_{cold}$ limits; (2) TS 3.4.10, ''Pressurizer Safety Valves,'' would be modified to increase the setpoint range of the lift settings for the pressurizer safety valves for the Oconee unit that has been analyzed in accordance with the TR and state that the range is not changed for the other units; (3) a statement to SR 3.4.10.1 would be added that will specify the pressurizer safety valve lift setpoint in order to clarify the difference between the operability setpoint range for a test lift and the range required when the setpoint is reset following the surveillance test; (4) TS 3.7.4, ''Atmospheric Dump Valve (ADV) Flow Paths,'' would be added to address the applicability and required actions related to the ADS valves; (5) TS 3.9.7, ''Unborated Water Source Isolation Valves,'' would be added to require valves that are used to isolate unborated water sources to be secured in the closed position while in Mode 6, incorporate SRs, and provide required actions if one or more of the valves is not secured in the closed position; (6) TS 5.6.5b would be changed to update the Core Operating Limits Report references; and (7) the appropriate Bases would be changed to reflect the above changes, other changes consistent with the revisions to the TR analysis, and the Updated Final Safety Analysis Report revisions that were provided in the submittal.

*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?

No. The proposed changes to the Technical Specifications, Bases, Updated Final Safety Analysis Report (UFSAR), and Core Operating Limits Report (COLR) incorporate the accident analyses established in Topical

Report DPC–NE–3005–P, ''UFSAR Chapter 15 Transient Analysis Methodology.'' On July 30, 1997, Duke submitted Topical Report DPC–NE–3005–P to the NRC for approval. The NRC found DPC–NE–3005–P acceptable, with noted exceptions, in a Safety Evaluation issued on October 1, 1998. To resolve the noted NRC exceptions, Duke submitted Revision 1 of DPC–NE–3005–P to the NRC for review on February 1, 1999. Additional information regarding Revision 1 of DPC–NE–3005–P was submitted on April 19 and May 5, 1999. This LAR is dependent upon the NRC approval of Revision 1 of DPC–NE–3005–P. [This Topical Report was approved by the NRC on May 25, 1999.]

The analyzed events are initiated by the failure of specific plant structures, systems or components. These proposed changes do not impact the condition or performance of those structures, systems or components.

The revised accident analyses in DPC–NE–3005–P demonstrate that the applicable acceptance criteria are met. In addition, the preliminary calculations show that the applicable radiological and environmental acceptance criteria continue to be met.

Based on the above, the proposed changes 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?

No. The proposed changes do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. Where setpoints and operating limits have been revised, the revised accident analyses demonstrate that the applicable acceptance criteria are met. As a result, no new failure modes are being introduced.

Based on the above, the proposed changes do not create the possibility of any new or different kind of accident from any accident previously evaluated.

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

No. 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 a event. The proposed changes do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. Where setpoints and operating limits have been revised, the revised accident analyses in DPC–NE–3005–P demonstrate that the applicable acceptance criteria are met.

Based on the above, the proposed changes do not involve a significant reduction in a margin of safety.

Based upon the preceding evaluation, performed pursuant to 10 CFR 50.92, Duke has concluded that the proposed changes to the Oconee Nuclear Station Technical Specifications, Bases, UFSAR, and O2C18 COLR will not involve a significant hazards consideration.

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:* Conee County Library, 501 West South Broad Street, Walhalla, South Carolina

*Attorney for licensee:* Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC.

*NRC Section Chief:* Richard L. Emch, Jr.

*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:* May 24, 1999

*Description of amendment request:* The proposed amendments would revise the maximum local fuel pin centerline temperature safety limit in Technical Specification 2.1.1.1 from the limit determined using the TACO2 fuel performance computer code to the value determined using a newer TACO3 computer code.

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 following discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92 (c) requirements to demonstrate that all three standards for no significant hazards consideration are satisfied. A no significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendment would not:

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

*First Standard*

Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. The use of the revised maximum local fuel pin centerline temperature limit is appropriate since the new limit uses a fuel melt temperature which has been conservatively reduced to account for code uncertainties in calculating fuel centerline temperature. NRC has previously found the use of the TACO3 code by DPC [Duke Power Company] in performing reload licensing to be acceptable. The use of the revised limit for fuel analyzed using an approved code ensures centerline fuel melting is avoided by ensuring the maximum fuel temperature is less than the melting temperature of the fuel. Therefore this change would not involve a significant increase in the probability or consequences of an accident previously evaluated.

*Second Standard*

Implementation of this amendment will not create the possibility of a new or different kind of accident from any previously evaluated. The use of the revised maximum local fuel pin centerline temperature limit has no affect on accident precursors. Implementation of this amendment will not impact any plant systems that are accident initiators. No other modifications are being proposed in the plant that would result in the creation of a new accident mechanism. Also, no changes are being made to the way the plant is operated; therefore, no new failure mechanisms will be initiated.

*Third Standard*

The revised maximum local fuel pin centerline temperature limit has been appropriately reduced to account for uncertainties in predicting centerline fuel temperatures. NRC has previously found the use of the TACO3 code by DPC in performing reload licensing to be acceptable. Therefore, implementation of this amendment would not involve a significant reduction in a margin of safety.

Therefore, Duke has concluded that the proposed amendment does not involve a significant hazards consideration.

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

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

*Attorney for licensee:* Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC.

*NRC Section Chief:* Richard L. Emch, Jr.

*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:* May 27, 1999.

*Description of amendment request:* The proposed changes would relocate the seismic monitoring instrumentation requirements contained in Technical Specification (TS) 3/4.3.3.3 to the Licensing Requirements Manual based on the guidance provided in Generic Letter 95–10, ''Relocation of Selected Technical Specifications Requirements

Related to Instrumentation.'' The Bases section for Specification 3/4.3.3.3 will also be relocated to the LRM. The appropriate Index pages, Table Index page (Unit No. 1 only), TS pages and Bases pages will be revised to reflect the removal of the seismic monitoring instrumentation specification from the TSs. An additional specification page will be added to reflect that Specification Number 3/4.3.3.4 is not used. This additional page will also denote the number of the following page. The Bases section will also be modified to denote that Specification Number 3/4.3.3.4 is not used.

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

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

The proposed amendment would relocate Technical Specification (TS) 3/4.3.3.3 titled ''Seismic Instrumentation'' and the associated Bases section to the Licensing Requirements Manual (LRM) (based on the guidance provided in Generic Letter (GL) 95–10, ''Relocation of Selected Technical Specification Requirements Related to Instrumentation''). The proposed amendment would also revise the TS Index and Beaver Valley Power Station (BVPS) Unit No. 1 List of Tables to reflect the relocation of this TS and associated Bases. The relocated Specification will be controlled in accordance with the requirement of 10 CFR 50.59, ''Controls, Tests, and Experiments.'' Additional administrative changes are also included to reflect that Specification Number 3/4.3.3.4 is not used.

The proposed amendment does not involve a significant increase in the probability of an accident previously evaluated because no changes are being made to any accident initiator. No analyzed accident scenario is being changed. The initiating condition and assumptions remain as previously analyzed. The failure of the seismic monitoring instrumentation to detect a seismic event is not an accident initiating event.

The seismic monitoring instrumentation performs no role in mitigating a seismic event or in achieving a safe shutdown condition after a seismic event has occurred. Seismic instrumentation is not assumed to function in the safety analysis. The seismic instrumentation is not associated with a process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Seismic instrumentation does not actuate any protective equipment or play any direct role in the mitigation of an accident. The capability of the plant to withstand a seismic event or other design basis accident is determined by the initial design and construction of systems, structures, and components. This instrumentation is used to alert operators to the seismic event and evaluate the plant response.

The proposed revisions to the Index pages, Table Index page (BVPS Unit No. 1 only), Specification pages and Bases pages are administrative in nature and do not affect plant safety.

Therefore, the proposed amendment does 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 amendment does not involve any physical changes to the plant or the modes of plant operation defined in Appendix A of the operating license. The proposed amendment does not involve the addition or modification of plant equipment nor does it alter the design or operation of plant systems. Seismic instrumentation does not actuate any protective equipment or play any direct role in the mitigation of an accident. The capability of the plant to withstand a seismic event or other design basis accident is determined by the design and construction of systems, structures, and components. This instrumentation is used to alert operators to the seismic event and evaluate the plant response.

Therefore, operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendment does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed amendment does not affect the ability of systems, structures or components important to ensure the safe shutdown of the facility, or the mitigation and control of accident conditions within the facility. In addition, the proposed amendment does not affect the ability of safety systems to ensure that the facility can be maintained in a shutdown or refueling condition for extended periods of time, or the availability of sufficient instrumentation and control capability for monitoring and maintaining the unit status.

The proposed revisions to the Index pages, Table Index page (BVPS Unit No. 1 only), Specification pages and Bases pages are administrative in nature and do not affect plant safety.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the 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 Section Chief:* S. Singh Bajwa.

*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:* May 27, 1999.

*Description of amendment request:* The proposed amendments would (1) revise the frequency for performing the CHANNEL FUNCTIONAL TEST (CFT) of the manual initiation functional units specified in the Beaver Valley Power Station, Unit Nos. 1 and 2, Engineered Safety Features Actuation System (ESFAS) Instrumentation Technical Specifications (TSs) from monthly, with an accompanying footnote which allows the manual initiation to be tested on a refueling interval, to each refueling interval; (2) Revise footnotes associated with TS ESFAS tables; (3) revise associated TS Bases.

*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 frequency notation specified for the channel functional test of the manual initiation functions listed on Table 4.3–2 of TS 3/4.3.2, ''Engineered Safety Feature Actuation System (ESFAS) Instrumentation.'' The proposed change revises the current TS requirement for surveillance testing these functions to clarify that testing be performed on a refueling basis. The revision to the surveillance frequency specified in Table 4.3–2 does not physically impact the Instrumentation, its setpoints, or the actual frequency at which the manual initiation functions are tested. The revision eliminates the potential for confusion regarding the testing required for the manual initiation function by deleting Footnote (1) to Table 4.3–2. The proposed change to the Surveillance Requirements of Table 4.3–2 for the manual initiation functions eliminates the need for Footnote (1). Footnote (1) requires testing the manual actuation switches every 18 months and performing a Channel Functional Test on all other circuitry associated with manual safeguards actuation every 31 days. As there is no other circuitry for which a 31 day CFT is applicable, the proposed change simplifies the TS requirement consistent with the current Standard TS for Westinghouse plants. Footnote (1) is consistent with early versions of the Standard Technical Specifications of

NUREG–0452; however, later versions of the Standard Technical Specifications and the Improved Standard Technical Specifications of NUREG–1431 simply require testing manual initiation functions on a refueling or 18 month basis. The proposed refueling frequency for testing this instrumentation recognizes that the manual initiation functions can not be tested at power since this would introduce the potential for a significant plant transient.

The deletion of Table 4.3–2 Footnote (1) resulted in renumbering Footnote (2) to (1). In addition, expired Unit 2 Table 4.3–2 Footnote (3) (only applicable to the first refueling outage) was also deleted. In addition, changes to the TS bases are made to further clarify the channel functional test requirements. The reorganization of the Table 4.3–2 footnotes and bases modifications are considered to be editorial changes.

The manual initiation instrumentation will continue to be tested in the same manner as before (every refueling). This test frequency is consistent with the licensing basis for testing this instrumentation described in the Updated Final Safety Analysis Report (UFSAR) and with the testing frequency specified in the standard Westinghouse Plant TS. Therefore, this test frequency is considered adequate to verify instrumentation operability. In addition, failure of a manual initiation function is not an accident initiator. As such, the ESFAS instrumentation will continue to be capable of providing the required safety functions described in the UFSAR. Therefore, operation of the facility in accordance with the proposed amendment does not involve a significant increase in the probability or consequence 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?

There are no hardware changes associated with this license amendment nor are there any changes in the method by which any safety-related plant system performs its safety function. No new accident scenarios, transient precursors, failure mechanisms or limiting single failures are introduced as a result of these changes. These changes do not introduce any adverse effects or challenges to any safety-related systems. No change is required to any system configurations, plant equipment or analyses. Therefore, these changes will not create the possibility of any 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 margin of safety depends on the maintenance of specific operating parameters and systems within design requirements. Updating the manual initiation function surveillance interval requirements specified on ESFAS TS Table 4.3–2 and deleting Table 4.3–2 Footnote (1) reflects the standard Westinghouse Plant TS requirements for this instrumentation and is consistent with the design and operation of the plant as described in the UFSAR. In addition, the proposed change does not reduce the current refueling interval testing performed on this instrumentation. The refueling test frequency specified for this instrumentation is consistent with industry standards and considered adequate to ensure the affected manual initiation functions are maintained operable. The proposed change will improve the clarity of the TS requirement by eliminating the potential for confusion as to when the surveillances are required to be performed. As such, the proposed change continues to ensure that the operation of the affected instrumentation is maintained within its design requirements and that it continues to be capable of providing the required safety functions described in the UFSAR. Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

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

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

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

*NRC Section Chief:* S. Singh Bajwa.

*Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas*

*Date of amendment request:* June 1, 1999.

*Description of amendment request:* The proposed amendment would revise the surveillance requirements and applicable Bases relevant to inservice inspection requirements for the portions of the once-through steam generator (OTSG) tubes adjacent to the primary cladding region of the upper and lower OTSG tubesheets.

*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:

Criterion 1—Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The OTSGs are used to remove heat from the reactor coolant system during normal operation and during accident conditions. The OTSG tubing forms a substantial portion of the reactor coolant pressure boundary. An OTSG tube failure is a breach of the reactor coolant pressure boundary and is a specific accident analyzed in the Arkansas Nuclear One, Unit 1 (ANO–1), Safety Analysis Report (SAR).

The purpose of the periodic surveillance performed on the OTSGs in accordance with ANO–1 Technical Specification (TS) 4.18 is to ensure that the structural integrity of this portion of the reactor coolant system will be maintained. The TS plugging limit of 40% of the nominal tube wall thickness requires tubes to be repaired or removed from service because the tube may become unserviceable prior to the next inspection. Unserviceable is defined in the TS as the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an operating basis earthquake, a loss-of-coolant accident, or a steam line or feedwater line break. The proposed TS change allows OTSG tubes with axial TEC [tube end cracking] indications that do not extend from the cladding region into the carbon steel interface within the tube-to-tubesheet rolled joint of the tubesheets to remain in service with existing degradation exceeding the existing 40% through-wall (TW) plugging limit.

Extensive testing and plant experience has illustrated that TEC flaws confined to this area within the OTSG will not result in tube burst or significant tube leakage under MSLB [main steamline break] conditions. Potential leakage from tubes with TEC will be bounded by the MSLB evaluation presented in the SAR. Therefore, allowing TEC flaws in this specific region to remain in service will not alter the conditions assumed in the current ANO–1 accident analysis for OTSG tube failures under postulated accident conditions. In addition, the condition of the OTSG tubes in this region are monitored during regular inspection intervals to assess for evidence of growth. Any growth noted will be addressed through the operational assessment. Therefore, Entergy Operations has determined that the identification, monitoring, assessment, and corrective action programs * * * [associated with the proposed changes] sufficiently support this change request.

Application of the TEC alternate repair criteria will allow leaving tubes with TEC indications found in the defined area of the tubesheets in service while ensuring safe operation by monitoring and assessing the present and future conditions of the tubes. Through the inspection, monitoring, and assessment programs previously mentioned, and the on-line leak detection capabilities available during plant operation, continued safe operation of ANO–1 is reasonably assured.

Therefore, the application of the TEC alternate repair criteria * * * does *not* involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2—Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The implementation of the TEC alternate repair criteria will not result in any failure mode not previously analyzed. The OTSGs are passive components. The intent of the TS surveillance requirements are being met by these proposed changes in that adequate structural integrity will be maintained. Potential leakage under MSLB conditions will remain bounded by the current SAR analysis. Additionally, the proposed change does not introduce any new modes of plant operation.

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

Criterion 3—Does Not Involve a Significant Reduction in the Margin of Safety.

The application of an alternate repair criteria for TEC provides adequate assurance with margin that ANO–1 steam generator tubes will retain their structural integrity under normal and accident conditions. The structural requirements of TEC affected tubes have been evaluated satisfactorily and meet or exceed regulatory requirements. The tubing region where TEC occurs is constrained within the tubesheet bore; therefore, there is no additional risk associated with tube rupture. Main steam line break leakage rates for these tubes are reasonably assured to remain within the assumptions of the accident analysis by proper application of the TEC alternate repair criteria program. Because no appreciable impact is evidenced on the tubes structural integrity or its potential leakage rate, the margin to safety remains unaltered.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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:* Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

*Attorney for licensee:* Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

*NRC Section Chief:* Robert A. Gramm.

*Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida*

*Date of amendment request:* June 1, 1999.

*Description of amendment request:* The amendments would revise the St. Lucie, Units 1 and 2, Technical Specifications (TS), Sections 3.5.2, to allow up to 7 days to restore an inoperable Low Pressure Safety Injection System train to operable status. The amendments would also revise the associated surveillance requirements and TS Bases sections to be consistent with the revisions to TS Section 3.5.2. Minor editorial changes for the specified Recirculation Actuation Signal (RAS) verification test are also included to ensure the terminology used in the specification is consistent with plant design.

*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 amendments for St. Lucie Plant, Units 1 and 2 will extend the action completion/allowed outage time (AOT) for a single inoperable Low Pressure Safety Injection (LPSI) train from 72 hours to 7 days. A LPSI train is designed as a part of each Emergency Core Cooling System (ECCS) subsystem to supplement Safety Injection Tank (SIT) inventory during the early stages of mitigating a Design Basis Accident. As such, components of the LPSI system are not accident initiators, and an extended AOT to restore operability of an inoperable LPSI train would not increase the probability of occurrence of accidents previously analyzed.

The safety analyses for both St. Lucie Units demonstrate that ECCS performance acceptance criteria are satisfied with only one of the two redundant ECCS subsystems operating during the postulated Design Basis Accident. The proposed technical specification revisions involve the AOT for a single inoperable LPSI train, and do not change the conditions assumed for the minimum amount of operating equipment needed for accident mitigation. Therefore, the consequences of an accident previously evaluated will not be significantly increased.

In addition to the preceding evaluation, a Probabilistic Safety Analysis (PSA) was performed to quantitatively assess the risk impact of the proposed amendments. It was concluded from the results of that assessment that the risk contribution of the AOT extension is very small, and that the net impact of the proposed amendment can be risk beneficial.

The editorial corrections proposed for the specified RAS verification test do not alter existing test requirements and have no impact on the accident analyses. Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant increase in 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 accident previously evaluated.

The proposed amendments will not change the physical plant or the modes of plant operation defined in either Facility License. The changes do not involve the addition or modification of equipment nor do they alter the design of plant systems. Therefore, operation of either facility in accordance with its proposed amendment would 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 margin of safety associated with the ECCS system is established by acceptance criteria for system performance defined in 10 CFR 50.46. The proposed amendments will not change these acceptance criteria or the operability requirements for equipment that is used to achieve such performance as demonstrated in the plant safety analyses. Moreover, an integrated assessment of the risk impact of extending the AOT for a single inoperable LPSI train has concluded that the risk contribution is very small, LPSI system reliability can potentially be improved, and the net impact of the proposed change can be risk beneficial. The editorial corrections proposed for the specified RAS verification test do not alter existing test requirements and have no impact on the accident analyses. Therefore, operation of either facility in accordance with its proposed amendment would 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:* Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

*Attorney for licensee:* M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

*NRC Section Chief:* Sheri R. Peterson.

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

*Date of amendment request:* May 13, 1999.

*Description of amendment request:* The proposed amendment would make changes to the TMI–1 Facility Operating License No. DPR–50 Sections 2.a, 2.c.(3), and 2.c.(7) to delete obsolete or outdated portions of the license conditions, and would change the Bases for Technical Specification 3.1.1.

*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 of occurrence or consequences of an accident previously evaluated. Most of the proposed amendment is only administrative; it adds to the Technical Specifications generic references to various documents. These changes have no affect upon the plant design or operation.

The proposed change to the Technical Specification Bases 3.1.1 is the removal of the specified pressurizer code safety valve flow-rate for which no basis could be found and the acceptance of a 3% setpoint drift (as-found) as per the ASME code. The 3% code limit is in accordance with the plant's Inservice Test Program submittal, which was evaluated by the NRC staff for the current 10 year interval and documented under NRC TAC No. M93777. The [c]orrect pressurizer code safety valve flow is provided in the FSAR Table 4.2–8. The proposed change is supported by a revise[d] Startup Accident analysis with the revised safety valve flow-rate at the 3% setpoint drift, which demonstrated that the acceptance criteria for the event were met with considerable margin. The proposed change does not affect the Technical Specification 3.1.1.a, pressurizer code safety valve operable (as-left) requirement of [plus or minus] 1%.

Therefore, operation in accordance with the proposed amendment does not involve a significant increase in 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 accident previously evaluated, because no new failure modes are created by the proposed changes. The administrative changes are cosmetic and have no impact on plant design or operation.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. 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, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

*NRC Section Chief:* S. Singh Bajwa.

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

*Date of amendment request:* May 26, 1999.

*Description of amendment request:* The proposed amendment would approve changes to the TMI–1 Updated Final Safety Analysis Report (UFSAR) which would allow use of the EPRI (Electric Power Research Institute) Conservative Deterministic Failure Margin (CDFM) methodology for seismic analysis of the portions of the auxiliary steam line located in the Auxiliary, Control and Fuel Handling buildings at TMI–1. The licensee determined that these changes to the UFSAR required prior NRC approval in accordance with 10 CFR 50.59.

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

1. The proposed amendment, use of CDFM methodology for the analysis of the auxiliary steam system piping, would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The analysis of the auxiliary steam pipe using the CDFM methodology demonstrates that the pipe wall will maintain integrity sufficient to prevent adverse impact on safety related equipment during a safe shutdown earthquake (SSE). The methodology is based on actual earthquake experience data and has been shown to be adequate to demonstrate that piping systems will maintain integrity. The CDFM methodology was developed by experts in the field of seismic analysis and is based on actual earthquake experience and the results of dynamic tests with large seismic accelerations. The methodology provides a conservative mechanism for analytically predicting performance during actual earthquakes, and thus its application would not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment, use of CDFM methodology for the analysis of the auxiliary steam system piping, would not create the possibility of a new or different kind of accident from any accident previously evaluateed.

No changes to plant systems, structures or components are proposed and no changes to methods of operation [of the plant] are involved.

3. The proposed amendment, use of CDFM methodology for the analysis of the auxiliary steam system piping, would not involve a significant reduction in a margin of safety.

No changes are proposed to operating limits or safety system settings, or to accident analysis acceptance criteria. The CDFM methodology provides a conservative mechanism for analytically predicting system performance during actual earthquakes. Its application to the auxiliary steam system piping would 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, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

*NRC Section Chief:* S. Singh Bajwa.

*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 4, 1999.

*Description of amendment request:* The amendment revises decay heat removal capability requirements to ensure that at least two active methods of decay heat removal capability will be available during shutdown conditions except when the reactor vessel head is removed and the fuel transfer canal water level is greater than or equal to 23 feet above the reactor vessel flange.

*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:

GPU Nuclear has determined that this Technical Specification Change Request poses no significant hazards as defined by NRC in 10 CFR 50.92. 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 because the proposed changes would remove exceptions for decay heat removal system operability requirements during the time the plant is in a Refueling Shutdown with the RCS loop not filled. The proposed changes effectively add requirements to maintain redundancy in decay heat removal systems.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes would not introduce any new failure modes or modify existing systems.

3. Involve a significant reduction in a margin of safety because the proposed amendment would not involve changes to the safety limits, limiting safety system settings, or operating limits.

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, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

*NRC Section Chief:* S. Singh Bajwa.

*Northeast Nuclear Energy Company, et al., Docket No. 50–245, Millstone Nuclear Power Station, Unit No. 1, New London County, Connecticut*

*Date of amendment request:* April 19, 1999.

*Description of amendment request:* The proposed amendment would replace the current set of technical specifications for the Millstone Unit 1 plant with a new set of technical specifications for the permanently shutdown status of the plant.

*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, a summary of which is presented below:

The proposed change does not:

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

This proposed change is consistent with the STS [standard technical specifications]. The relocation of requirements from the MP1 TS [Millstone Unit 1 Technical Specifications] to the licensee controlled documents is consistent with the criteria set forth in 10 CFR 50.36 for the content of Technical Specifications. The removal of definitions, generic LCO [limiting condition for operation] actions and generic surveillance requirements has no impact on facility SSCs [structure, system, and components] or the methods of operation of such SSCs. The deletion of design features and safety limits not applicable to the permanently shutdown and defueled status of MP1 has no impact on the remaining DBA [design-basis accident], the fuel handling accidents in the fuel storage pool. The removal of LCOs and surveillance requirements which are related only to the operation of the nuclear reactor or only to the prevention, diagnosis or mitigation of reactor-related transients or accidents do not affect the applicable DBA previously evaluated. The critical safety functions involving core reactivity control, reactor heat removal, reactor coolant system inventory control and containment integrity are no longer necessary at MP1. The proposed accidents involving damage to the reactor coolant system, main steam lines, reactor core, and the subsequent release of radioactive material are no longer possible at MP1. Fuel pool cooling and makeup related equipment and support equipment (e.g., electrical power systems) are not required to be continuously available since recent analysis demonstrated that there is up to ten days before fuel storage pool boiling to effect repairs, establish alternate sources of make up flow, or establish steady state natural air circulation cooling of the Reactor Building atmosphere and fuel storage pool water in the event of a loss of cooling and makeup flow to the fuel pool. The radioactive decay of the irradiated fuel since shutdown of the reactor in November, 1995 has reduced the consequences of the fuel handling accident to levels well below those previously analyzed. The relevant parameter (water level) associated with the fuel pool provides an initial condition for the fuel handling accident analyses and is included in the PDTS [Permanently Defueled Technical Specifications]. The Reactor Building crane LCOs are retained to preserve the engineered controls which preclude a spent fuel cask drop from occurring over the fuel storage pool. The deletion and modification of provisions of the administrative controls do not directly affect the design of SSCs necessary for safe storage of irradiated fuel or the methods used for handling and storage of such fuel in the fuel pool. The relocation of administrative controls related to quality assurance to the Northeast Utilities Quality Assurance Program is also consistent with the guidance provided in NRC Administrative Letter AL 95–06, ‘‘Relocation of Technical Specification Administrative Controls Related to Quality Assurance,’’ dated December 12, 1995. The changes to the administrative controls are administrative in nature and do not affect any accidents applicable to the safe storage of irradiated fuel or the permanently shutdown and defueled condition of the reactor. Therefore, the proposed changes to the MP1 TS do not involve any increase in the probability or consequences of any accident previously evaluated.

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

The proposed changes have no impact on facility SSCs affecting the safe storage of irradiated fuel or on the methods of operation of such SSCs, or handling and storage of such fuel. These changes are consistent with the STS and add to the clarity and ease of use of the proposed PDTS. The removal of Technical Specifications which are related only to the operation of the nuclear reactor or only to the prevention, diagnosis, or mitigation of reactor-related transients or accidents cannot result in different or more adverse failure modes or accidents than previously evaluated because the reactor is permanently shutdown and defueled and MP1 is no longer authorized to operate the plant. The proposed deletion of provisions of the MP1 TS do not affect systems credited in the accident analyses for the fuel handling accident in the fuel storage pool at MP1. The proposed PDTS continue to require proper control and monitoring of safety significant parameters and activities. The proposed restriction on the fuel pool level is fulfilled by normal operating conditions and preserves initial conditions assumed in the analyses of the postulated DBA. Reactor Building crane LCOs are retained from current Technical Specifications to preclude the possibility of a spent fuel cask drop over the fuel storage pool. Therefore, the proposed changes to this section of the MP1 TS would 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 deletion of provisions of the MP1 TS, which are not related to the storage of irradiated fuel or which are inconsistent with the scope of the STS, will not affect the analyses of the remaining DBA applicable to MP1. The postulated DBAs involving the reactor are no longer possible due to the permanently shutdown and defueled condition of the reactor. The requirements for SSCs which have been deleted from the MP1 TS are not credited in the existing accident analyses for the remaining applicable postulated accidents and therefore, do not contribute to the margin of safety associated with the accident analysis. Therefore, the proposed changes to this section of the MP1 TS do not involve any reduction in a margin of safety.

*Conclusion*

NNECO has concluded that the proposed change to the MP1 Technical Specifications does not involve a significant hazards consideration as defined by 10 CFR 50.92.

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 06360, 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 Section Chief:* Michael T. Masnik.

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

*Date of amendment request:* January 25, 1996, as supplemented April 26, 1996, September 12, 1996, March 17, 1997, September 9, 1997, December 30, 1998, and May 19, 1999.

*Description of amendment request:* The proposed changes extend the allowed outage time for an emergency diesel generator (EDG) system from 7 to

14 days. At FitzPatrick, an EDG system consists of 2 EDGs powering one of two emergency AC power buses. The proposal includes provisions for a Configuration Risk Management Program (CRMP) consistent with the guidance of Regulatory Guide (RG) 1.177, ''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications.'' The NRC staff had previously published a notice on these topics on March 27, 1996 (61 FR 13532). This revised notice on these topics is required to address revisions made in the licensee's supplemental submittals.

The licensee's January 25, 1996, submittal also proposed two line-item changes to reduce EDG testing at power and to revise AC power requirements for cold shutdown and refueling modes. The two line-item changes have not been affected by the supplemental information provided by the licensee, so the March 27, 1996, proposed finding of no significant hazards considerations remains valid for these items.

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

Operation of the FitzPatrick plant in accordance with the additional changes to the proposed Amendment discussed above, would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

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

The proposed changes to the Technical Specifications will allow longer Allowed Out of Service Times to perform necessary repair and maintenance on Emergency Diesel Generators while at power. This extended AOT [allowed outage time] will enhance scheduling of preventive maintenance of individual EDGs without significantly increasing the probability or consequences of an accident previously evaluated. The risk evaluations for the EDGs determined that the probability of an accident by increasing the AOT for an EDG System from 7 days to 14 days is non-risk-significant.

Increasing the EDG AOT does not involve physical alteration of any plant equipment and does not affect analysis assumptions regarding functioning of required equipment designed to mitigate the consequences of accidents. Further, the severity of postulated accidents and resulting radiological effluent releases will not be affected by the increased AOT for an EDG System.

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

[The CRMP provides administrative controls to ensure equipment configurations do not result in any significant increase in plant risk. In RG 1.177, the NRC staff established a standard for the content of the CRMP. The licensee's proposal is consistent with that standard, and so does 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.

Extending the AOT for an EDG system does not necessitate physical alteration of the plant or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for [the] JAF [FitzPatrick] plant.

[The CRMP provides administrative controls to ensure equipment configurations do not result in any significant increase in plant risk. These administrative controls do not create any new equipment configurations, or provide for operation of equipment in a new or different manner. Therefore, the CRMP does 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.

As discussed above, a Fitzpatrick evaluation determined that the change in risk associated with extending the AOT for a[n] EDG System is non-risk-significant. In addition, the design provides adequate redundancy for safe shut down during the AOT with an EDG System out of service. This is supported by the LOCA [loss-of-coolant accident] analyses including analyses for long term suppression pool cooling and reactor shutdown cooling.

[The CRMP provides administrative controls to ensure equipment configurations do not result in any significant increase in plant risk. These administrative controls do not create any new equipment configurations, or provide for operation of equipment in a new or different manner. Therefore, the proposed CRMP does not involve a significant reduction in the margin of safety.]

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

*Local Public Document Room location:* 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 Section Chief:* S. Singh Bajwa.

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

*Date of amendment request:* May 24, 1999.

*Description of amendment request:* The proposed amendment would revise the Technical Specifications (TS) to correct typographical and editorial errors, and is considered administrative in nature.

*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 editorial and administrative changes involve typographical errors and/or reflect changes that were previously reviewed and approved by the NRC. These changes, therefore, do not modify or add any initiating parameters that would significantly increase the probability or consequences of any previously analyzed accident.

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

These proposed changes do not involve any potential initiating events that would create the possibility of a new or different kind of accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

These changes are editorial in nature and/or reflect information previously reviewed and approved by the NRC. The proposed changes will make the information in the TS consistent with that already approved by the NRC. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

*Local Public Document Room location:* 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 Section Chief:* James W. Clifford.

*Sacramento Municipal Utility District (the District), Docket No. 50–312, Rancho Seco Nuclear Station, Sacramento County, California*

*Date of amendment request:* April 23, 1999.

*Description of amendment request:* The proposed amendment would change Permanently Defueled Technical Specification (PDTS) D3/4.1, ''Spent Fuel Pool Level,'' to replace a specific reference to spent fuel pool (SFP) level alarm switches with a generic reference to SFP level instrumentation. This would allow the licensee to replace the old level alarm switches with a new ultrasonic level transmitter.

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:

PA–193 will not create a significant increase in the probability or consequences of an accident previously evaluated in the SAR [Safety Analysis Report], because the proposed PDTS change is editorial in nature and only changes the type of equipment that is referenced in surveillance specification D4.1.2. The SFP level instrument reference in D4.1.2 is changed from a specific reference (i.e., SFP level alarm switches) to a more generic reference (i.e., SFP level instrumentation). In addition:

1. SFP level monitoring instrumentation is not relied on to mitigate the consequences of the accidents analyzed in the SAR (i.e., Fuel Handling Accident, Loss-Of-Offsite-Power event, Liquid Tank Ruptures, and Decommissioning Accidents),

2. PA–193 does not alter the SFP level monitoring, SFP cooling, or fuel handling functions during the PDM [Permanently Defueled Mode],

3. PA–193 continues to require an 18-month calibration of SFP level instrumentation, and

4. SFP level and alarm indication in the Control Room is maintained with the new SFP level instrumentation. Also, the SFP level alarm setpoints remain unchanged with the new SFP level detection system.

PA–193 will not create the possibility of a new or different type of accident than previously evaluated in the SAR, because SFP level instrumentation does not provide any control function and does not affect any equipment associated with SFP cooling, fuel handling, or inventory control. The proposed wording change to PDTS D4.1.2 accommodates upgrading the SFP level instrumentation without changing the intent of surveillance specification D4.1.2. Also, the new SFP level detection system will (1) maintain the existing SFP level alarm setpoints and Control Room indication features and (2) have no adverse impact on the SFP level monitoring function.

PA–193 will not involve a significant reduction in the margin of safety, because the proposed PDTS change is editorial in nature and necessary and only accommodates replacing an unreliable, antiquated SFP level monitoring system with a new, state-of-the-art, ultrasonic level detection system. The new SFP level detection system will improve the accuracy, reliability, and serviceability of the SFP level monitoring function. The District is maintaining the requirement to perform a[n] SFP level calibration and is only changing the type of equipment that is referenced in D4.1.2 from a specific reference (i.e., SFP level alarm switches) to a more generic reference (i.e., SFP level instrumentation).

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

*Local Public Document Room location:* Central Library, Government Documents, 828 I Street, Sacramento, California 95814.

*Attorney for licensee:* Dana Appling, Esq., Sacramento Municipal Utility District, P.O. Box 15830, Sacramento, California 95852–1830.

*NRC Section Chief:* Michael T. Masnik.

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

*Date of amendment requests:* June 8, 1999 (PCN–495).

*Description of amendment requests:* The licensee has re-evaluated its small break loss-of-coolant accident (SBLOCA) using ABB Combustion Engineering (ABB–CE) S2M evaluation model. Based on this re-evaluation, the licensee proposes to revise the Technical Specifications (TSs) for the San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 to reflect that charging flow is not required to mitigate the effects of the SBLOCA, add a surveillance requirement to verify that each charging pump is operable for boration based on the Inservice Testing Program, increase the maximum as-found lift pressure positive tolerance of main steam safety valves (MSSVs) from +1% to +2% of the lift setting, and list the ABB–CE S2M model as an acceptable method for determining linear heat rate. The licensee will also revise the TS Bases and the Updated Final Safety Analysis Report (UFSAR) 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 any accident previously evaluated?

Response: No.

The new Small Break Loss Of Coolant Accident (SBLOCA) evaluation model (ABB Combustion Engineering (ABB–CE) S2M SBLOCA evaluation model, CENPD 137 Supplement 2–P–A, ''Calculative Methods of the ABB–CE Small Break LOCA Evaluation Model,'' dated April 1998) more accurately models the heat transfer mechanisms that occur during a SBLOCA. As a result of this modeling improvement, there is no longer a need to credit charging flow during a SBLOCA. The reanalysis, with an as-found tolerance of +2%/−3% of the lift setting on Main Steam Safety Valves (MSSVs) 2(3)–PSV–8401 and 2(3)–PSV–8410 in Table 3.7.1–2, determined that the peak cladding temperature (PCT) that occurs in a SBLOCA is within the acceptance criteria limit of 2200 [degrees] F specified in 10CFR50.46.

This proposed change removes the charging pump Emergency Core Cooling System (ECCS) surveillance requirement from the Technical Specifications (TS) which effectively removes the charging system from the ECCS. This is based on the SBLOCA reanalysis using the new ABB–CE S2M SBLOCA evaluation model. The reanalysis using the new model did not credit charging system flow to the reactor coolant system.

Because this proposed change to remove the charging pump ECCS flow surveillance requirement is based on a reanalysis of the SBLOCA rather than physical changes to the plant or the way it is operated, the probability of the SBLOCA is not affected. The results of the reanalysis demonstrate the consequences of the SBLOCA without charging flow do not exceed the consequences of the limiting LOCA. This is based on the fact that the SBLOCA PCT [peak clad temperature] does not exceed the limiting large break LOCA PCT.

The addition of Surveillance Requirement (SR) 3.1.9.5 to require the charging pump to be tested in accordance with the Inservice Testing (IST) program will ensure that the charging pumps remain capable of performing their emergency boration requirements.

Use of the NRC approved ABB–CE S2M SBLOCA analysis methodology identified in TS 5.7.1.5 for calculating the core operating limits further assures that there is no significant increase in the probability or consequences of any accident.

Therefore, the probability or consequences of any accident previously evaluated are not increased.

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

Response: No.

This change does not involve a physical change to the plant, or a change to the way the plant is operated. The as-left tolerance of [plus or minus] 1% on MSSVs 2(3)–PSV–8401 and 2(3)–PSV–8410 in Table 3.7.1–2 is not being changed. The charging system will still be verified capable of meeting its emergency boration requirements.

Use of the NRC approved ABB–CE S2M SBLOCA analysis methodology identified in TS 5.7.1.5 for calculating the core operating limits further assures that there is no increase in the possibility of a new or different kind of accident from any previously evaluated. Therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

(3) Involve a significant reduction in a margin of safety?

Response: No.

This proposed change to remove the ECCS surveillance requirement for the charging pumps, and increase the as-found tolerance on MSSVs 2(3)–PSV–8401 and 2(3)–PSV–8410, is based on a SBLOCA reanalysis using the new ABB–CE S2M SBLOCA evaluation model. The NRC Safety Evaluation for the ABB–CE S2M evaluation model determined that the new evaluation model contains sufficient conservatism such that an adequate margin of safety exists when the S21VI evaluation model is used. The results of the SBLOCA reanalysis are within the acceptance criteria specified in 10 CFR 50.46.

Testing of the charging pumps per the Inservice Testing Program, combined with the existing Technical Specification 3.1.9—''Boration System—Operating'' surveillance requirements ensure that the emergency boration requirements remain met without any reduction in a margin of safety.

Use of the NRC approved S2M ABB–CE SBLOCA analysis methodology identified in TS 5.7.1.5 for calculating the core operating limits further assures that there is no significant reduction in any margin of safety.

Therefore, a significant reduction in margin of safety is not involved.

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

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

*Attorney for licensee:* Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

*NRC Section Chief:* Stephen Dembek.

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

*Date of amendment request:* June 7, 1999.

*Description of amendment request:* The proposed amendments would revise Technical Specification (TS) 2.2.1, Reactor Trip System (RTS) Instrumentation Setpoints, and TS 3.3.2, Engineered Safety Features Actuation System (ESFAS) Instrumentation, and the associated Bases, by removing the Total Allowance (TA), Sensor Error (S), and Z terms from the RTS and ESFAS Instrumentation Trip Setpoints Tables. This would replace the five-column methodology with a two-column methodology that consists of the trip setpoint and allowable value columns.

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

The proposed change eliminates the option to evaluate the equation (Z+R+S [is less than or equal to] TA), within 12 hours, from Technical Specification 2.2.1, when the trip setpoint is outside the allowable value limit. The equation established a threshold for submitting a Licensee Event Report. The change does not affect the probability of an accident. The evaluation of the equation is an administrative provision and has no relevance to the initiation of any analyzed event. The consequences of an accident are not affected. The change will not alter assumptions relative to the mitigation of an accident or transient event.

The proposed amendment is a programmatic and administrative change that does not physically alter safety-related systems, nor does it affect the way in which safety-related systems perform their functions. Because the design of the facility and system operating parameters are not being changed, the proposed amendment does not involve an increase in the probability or consequences of any accident previously evaluated.

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

The proposed amendment is a programmatic and administrative change that does not physically alter safety-related systems, nor does it affect the way in which safety-related systems perform their functions. The changes in methods governing normal plant operation are consistent with current safety analysis assumptions. The proposed change eliminates the option to evaluate the equation (described above) within 12 hours, when the trip setpoint is outside the allowable limit. Because the design of the facility and system operating parameters are not being changed, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed amendment is a programmatic and administrative change that provides assurance that plant operations continue to be conducted in a safe manner. As stated above, the proposed amendment does not physically alter safety-related systems, nor does it affect the way in which safety-related systems perform their functions. The proposed change eliminates the option to evaluate the equation (described above) within 12 hours, when the trip setpoint is outside the allowable limit.

The margin of safety is not affected by eliminating an administrative provision in Technical Specifications. The determination for submitting a Licensee Event Report when a trip setpoint is outside the allowable value will be performed with the guidelines of 10CFR50.73. The safety analysis assumptions will still be maintained, thus, no question of safety exists. Because the design of the facility and system operating parameters are not being changed, the proposed amendment does not involve a significant reduction in the margin of safety.

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

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

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

*NRC Section Chief:* Robert A. Gramm.

*Tennessee Valley Authority, Docket Nos. 50–260, 50–296, Browns Ferry Nuclear Power Plant, Units 2 and 3. Limestone County, Alabama*

*Date of amendment request:* March 12, 1997 as supplemented by letters dated March 30, 1999, April 23, 1999 and June 18, 1999.

*Description of amendment request:* The proposed amendment would revise the Technical Specifications to extend, from 7 days to 14 days, the Allowable Outage Time (AOT) applicable to an inoperable emergency diesel generator.

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

**No Significant Hazards Consideration Determination**

TVA has concluded that operation of BFN in accordance with the proposed change to the TS does not involve a significant hazards consideration. TVA's conclusion is based on it's evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

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

The EDGs are designed as backup AC power sources in the event of loss of off-site power. The proposed AOT does not change the conditions, operating configurations, or minimum amount of operating equipment assumed in the safety analysis for accident mitigation. No changes are proposed in the manner in which the EDGs provide plant protection or which create new modes of plant operation. In addition, a PSA evaluation concluded that the risk contribution of the AOT extension is non-risk significant. Therefore, the proposed amendment does not involve a significant increase in 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 proposed change does not introduce any new modes of plant operation or make physical changes to plant systems. Therefore, extension of the allowable AOT for EDGs does not create the possibility of a new or different accident.

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

BFN's emergency AC system is designed with sufficient redundancy such that an EDG may be removed from service for maintenance or testing. The remaining EDGs are capable of carrying sufficient electrical loads to satisfy the UFSAR requirements for accident mitigation or unit safe shutdown.

Increasing the allowable EDG AOT will likely increase EDG unavailability on the average since it expected that the provision would occasionally be used to accommodate unplanned major EDG maintenance. However, a conservative PSA evaluation concluded that the risk contribution of the AOT extension is non-risk significant. For the 12-year EDG PM work activity, it is expected that the proposed TS would actually reduce unavailability since multiple outages would not be necessary to accomplish the maintenance activity.

The proposed change does not impact the redundancy or availability requirements of off-site power supplies or change the ability of the plant to cope with station blackout events. For these reasons, 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:* Athens Public Library, 405 E. South Street, Athens, Alabama.

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

*NRC Section Chief:* Sheri R. Peterson.

*TU Electric Company, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station (CPSES), Units 1 and 2, Somervell County, Texas*

*Date of amendment request:* May 4, 1999, as supplemented by letter dated June 4, 1999.

*Brief description of amendments:* The proposed license amendments would revise the Technical Specifications for CPSES, Units 1 and 2. Specifically, the changes would revise the surveillance requirements associated with the plant battery and emergency diesel generators, and correct miscellaneous editorial errors that resulted from the issuance of Amendment No. 64. The original application was noticed and published in the **Federal Register** on June 2, 1999 (64 FR 29715). The June 4, 1999, supplement provided proposed additional editorial corrections. The supplemental information is being noticed herein to address the issue of no significant hazards consideration.

*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. Do the proposed changes involve a significant increase in the probability or consequence of an accident previously evaluated?

(1) Batteries are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed change would not effect the design or performance of the batteries. The allowance to perform the modified performance discharge test in lieu of the service test at any time is permissible since the test's discharge rate envelopes the duty cycle of the service test. Therefore, the allowance for unrestricted substitution of the modified performance discharge test in lieu of the service discharge test does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) The diesel generators are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed change does not affect the accident analysis assumption that the DG reaches minimum conditions to accept load within 10 seconds. The ability of the DG to maintain steady state operation within 10 seconds is not an accident analysis assumption and is primarily used to identify degradation of governor and voltage regulator performance. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(3) The editorial changes are non-technical and therefore do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

(1) The allowance for unrestricted substitution of the modified performance discharge test in lieu of the service discharge test does not involve any physical alteration to the plant. No new failure mechanisms will be introduced and the change does not affect the ability of the batteries to fulfill their safety-related function. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(2) The separation of the DG start surveillance criteria into those criteria required to be met within 10 seconds, and those criteria required to be met following achievement of steady state conditions, does not involve any physical alteration to the plant. No new failure mechanisms will be introduced and the change does not affect the ability of the DGs to fulfill their safety-related function. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) The editorial changes are non-technical and therefore do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

(1) The allowance for unrestricted substitution of the modified performance discharge test in lieu of the service discharge test will not alter any accident analysis assumptions, initial conditions, or results. Consequently, it does not have any effect on the margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

(2) The proposed change to delete the requirement to demonstrate that the DG can achieve and maintain steady state operation within 10 seconds is not an accident analysis assumption. The accident analysis assumption that the DG reaches minimum conditions to accept load within 10 seconds is preserved. Consequently, it does not have any effect on the margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

(3) The editorial changes are non-technical and therefore do not involve a significant reduction in a margin of safety.

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

*Local Public Document Room location:* University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, Texas 76019.

*Attorney for licensee:* George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

*NRC Section Chief:* Robert A. Gramm.

*TU Electric Company, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station (CPSES), Units 1 and 2, Somervell County, Texas*

*Date of amendment request:* May 14, 1999.

*Brief description of amendments:* The proposed license amendments would change the name of the CPSES licensee from ''Texas Utilities Electric Company'' to ''TXU Electric Company'' in the Facility Operating Licenses of CPSES, Units 1 and 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. Does the proposed change involve a significant increase in the probability or consequence of an accident previously evaluated?

No. This request involves an administrative change only. The Operating Licenses (OLs) are being changed to reference the new corporate name of the licensee. No actual plant equipment or accident analyses will be affected by the proposed change. Therefore, TU [Texas Utilities] Electric concludes that this request will have no impact on the possibility of any type of accident, whether new, different or previously evaluated.

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

No. This request involves an administrative change only. The OLs are being changed to reference the new corporate name of the licensee. No actual plant equipment or accident analyses will be affected by the proposed change and no failure modes not bounded by previously evaluated accidents will be created. Therefore, TU Electric concludes that this request will have no impact on the possibility of any type of accident, whether new, different or previously evaluated.

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

No. Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. This request involves an administrative change only. The OLs are being changed to reference the new corporate name of the licensee. No actual plant equipment or accident analyses will be affected by the proposed change. Additionally, the proposed change will not relax any criteria used to establish safety limits, will not relax any safety systems settings, or will not relax the bases for any limiting conditions of operation. Therefore, this request will not impact 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:* University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, Texas 76019.

*Attorney for licensee:* George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

*NRC Section Chief: Robert A. Gramm.*

*TU Electric Company, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station (CPSES), Units 1 and 2, Somervell County, Texas*

*Date of amendment request:* May 24, 1999.

*Brief description of amendments:* The proposed license amendments would remove several cycle-specific parameter limits from the Technical Specifications (TSs) and add parameter limits to the Core Operating Limits Report. In addition, the core safety limit curves would be replaced with safety limits more directly applicable to the fuel and fuel cladding fission product barriers. The affected TSs are: (1) TS 2.0, ''Safety Limits (SLs)''; (2) TS 3.3.1, ''Reactor Trip System Instrumentation Setpoints''; (3) TS 3.4.1, ''RCS pressure temperature and flow from Nucleate Boiling (DNB) Limits''; and (4) TS 5.6.5, ''Core Operating Limits 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. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes remove cycle-specific parameter limits from the Technical Specifications, add them to the list of limits contained in the Core Operating Limits Report (COLR), and revise the Administrative Controls section of the Technical Specifications. The proposed changes also insert the original minimum RCS [reactor coolant system] flow limits into the Technical Specifications. The changes do not, by themselves, alter any of the parameter limits. The changes are administrative in nature and have no adverse effect on the probability of an accident or on the consequences of an accident previously evaluated. The removal of parameter limits from the Technical Specifications does not eliminate the requirement to comply with the parameter limits.

The parameter limits in the COLR may be revised without prior NRC approval. However, [Technical] Specification 5.6.5c continues to ensure that the parameter limits are developed using NRC-approved methodologies and that applicable limits of the safety analyses are met. While future changes to the COLR parameter limits could result in event consequences which are either slightly less or slightly more severe than the consequences for the same event using the present parameter limits, the differences would not be significant and would be bounded by the requirement of specification 5.6.5c to meet the applicable limits of the safety analysis.

Based on the above, addition of the minimum RCS flow limit into the Technical Specifications, removal of the parameter limits from the Technical Specifications and the addition of the described limits in the COLR, thus allowing revision of the parameter limits without prior NRC approval, has no significant effect on the probability or consequences of an accident previously evaluated.

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

The proposed changes add the minimum RCS flow limit into the Technical Specifications, remove certain parameter limits from the Technical Specifications and add these limits to the list of limits in the COLR, thus removing the requirement for prior NRC approval of revisions to those parameters. The changes do not add new hardware or change plant operations and therefore cannot initiate an event nor cause an analyzed event to progress differently. Thus, the possibility of a new or different kind of accident is not created.

3. Do the proposed changes involve a significant reduction in a margin of safety?

The margin of safety is the difference between the acceptance criteria and the associated failure values. The proposed changes do not affect the failure values for any parameter. Through the accident analyses, all applicable limits (i.e., relevant event acceptance criteria as described in the NRC-approved analysis methodologies) are shown to be satisfied; therefore, there is no impact on event acceptance criteria. Because neither the failure values nor the acceptance criteria are affected, the proposed change has no effect 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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, Texas 76019.

*Attorney for licensee:* George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

*NRC Section Chief:* Robert A. Gramm.

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

*Date of amendment request:* May 5, 1999.

*Description of amendment request:* The proposed change modifies the Technical Specifications (TS) to enhance limiting conditions for operation and surveillance requirements relating to the Standby Liquid Control (SLC) system and incorporates certain provisions of NRC's rule on anticipated transients without scram (ATWS) (10CFR50.62). The change involves the use of enriched boron in the SLC system and improves upon other aspects of the TS for this system.

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

1. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change deletes the requirement for standby liquid control (SLC) system operability during refueling and modifies the conditions for allowing the system to be inoperable when shutdown.

This change also permits changing the reactor mode switch to the ''Run'' or ''Startup/Hot Standby'' position to test mode switch interlock functions while the SLC system is inoperable. To allow testing of instrumentation associated with the reactor mode switch interlock functions, compensatory measures are provided for assuring that no core alterations are in progress and that all control rods remain fully inserted in core cells containing one or more fuel assemblies. These compensatory measures ensure that no credible mechanisms for an inadvertent criticality are introduced by administratively controlling the required functions of the reactor mode switch interlocks. Control rods are not required to be inserted in empty core cells (i.e., those containing no fuel) because, with one or more cells in this configuration, the overall shutdown margin is actually greater than when all control rods and all fuel assemblies are inserted.

The SLC system is not assumed in the initiation of any previously evaluated events and therefore the proposed change will not significantly increase the probability or consequences of a previously analyzed accident. The SLC system is not assumed to operate in the mitigation of any previously analyzed accidents which are assumed to occur during shutdown or refueling conditions. This change will not result in operation that will significantly increase the probability of initiating an analyzed event. This change will not alter assumptions relative to mitigation of an accident or alter the operation of process variables, structures, systems, or components as described in the final safety analysis report.

VY has determined that the proposed change to increase the standby liquid control system reactivity control capacity using a borated water solution enriched in the boron-10 isotope effectively increases the rate of injection of neutron absorber and does not alter the function of the system, method of operation or dual train configuration. The system response time to an anticipated transient without scram (ATWS) event has been reduced as the increased boron-10 enrichment of the solution provides faster negative reactivity insertion, thus reducing the consequences of the ATWS event. The SLC system is not credited in any of the design basis accident analyses and, as such, is considered to provide only an additional mitigative feature in the event of an accident. The SLC system sodium pentaborate solution concentration and flow rate required by the ATWS rule (10CFR50.62) for reactivity control independent of the control rods are not reduced from the values previously evaluated and presented in the Vermont Yankee Technical Specifications. The addition of enriched boron provides a shutdown margin greater than the previously calculated shutdown reactivity control capacity, and the change does not affect the probability of an ATWS event.

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

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change modifies the modes of applicability for the SLC system. Included in this change is allowance to permit changing the reactor mode switch to the ''Run'' or ''Startup/Hot Standby'' position to test mode switch interlock functions while the SLC system is inoperable. Precautions are taken when manipulating the mode switch to one of these positions to maintain all control rods fully inserted in core cells containing at least one fuel assembly and to not allow any core alterations. These two provisions eliminate the possibility of introducing any credible mechanisms for inadvertent criticality. The proposed change will not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not eliminate any valid requirements necessary for safe operation.

VY has determined that the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed change involves a system whose function is to provide an additional (backup) mitigative shutdown capability and no system modifications are made.

The addition of enriched boron does not affect any system or component that could initiate an accident. Thus, no new or different type of accident is created.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

VY has determined that the proposed change does not involve a significant reduction in a margin of safety. The proposed change would remove the backup to the available reactivity control systems when the reactor is in a shutdown or refueling condition. However, this backup is not considered in the margin of safety when determining the required reactivity for shutdown and refueling events. This change will have no impact on any safety analysis assumptions.

Included in this change is allowance to permit changing the reactor mode switch to the ''Run'' or ''Startup/Hot Standby'' position to test mode switch interlock functions while the SLC system is inoperable. The margin of safety will not be reduced during such testing of interlock functions with the SLC system inoperable because compensatory measures have been added to ensure that no credible mechanisms for inadvertent criticality exist with the reactor mode switch in other than the ''Shutdown'' or ''Refuel'' positions.

The use of enriched boron in the SLC system sodium pentaborate solution actually increases the capability of the SLC system to achieve cold shutdown; thus, no margin of safety is reduced.

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:* Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

*Attorney for licensee:* Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

*NRC Section Chief:* James W. Clifford.

*Washington Public Power Supply System, Docket No. 50–397, Nuclear Project No. 2, Benton County, Washington*

*Date of amendment request:* June 3, 1999.

*Description of amendment request:* The request is to amend the operating license such that the name of the licensee is changed from Washington Public Power Supply System to Energy Northwest. The name of the facility will be changed from WPPS Nuclear Project No. 2 to WNP–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.

This request involves an administrative change only. The Operating License (OL) is being changed to reference the new name of the licensee. No actual plant equipment or accident analyses will be affected by the proposed change. Therefore, this request will have no impact on the probability or consequence of any type of 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 request involves an administrative change only. The OL is being changed to reference the new name of the licensee. No actual plant equipment or accident analyses will be affected by the proposed change and no failure modes not bounded by previously evaluated accidents will be created. Therefore, this request will have no impact on the possibility of any new type of accident: new, different, or previously evaluated.

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

Margin of safety is associated with the confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. This request involves an administrative change only. The OL is being changed to reference the new name of the licensee.

No actual plant equipment or accident analyses will be affected by the proposed change. Additionally, the proposed change will not relax any criteria used to establish safety limits, will not relax any safety system settings, or will not relax the bases for any limiting conditions of operation. Therefore, this request will not impact 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:* Richland Public Library, 955 Northgate Street, Richland, Washington 99352.

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

*NRC Section Chief:* Stephen Dembek.

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

*Date of amendment request:* June 10, 1999.

*Description of amendment request:* The amendment would revise Technical Specification Table 3.3–4, Functional Unit 7.b., Automatic Switchover to Containment Sump (Refueling Water Storage Tank Level—Low-Low) to reflect the results of calculations that were performed for the associated instrumentation setpoints to consider the density variations due to temperature and boric acid concentration.

*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 protection system performance will remain within the bounds of the previously performed accident analysis. The protection systems will continue to function in a manner consistent with the plant design basis. The proposed changes will not affect any of the analysis assumptions for any of the accidents previously evaluated, since the changes are consistent with the setpoint methodology and ensure adequate margin to the Safety Analysis Limit. The proposed changes will not affect any event initiators nor will the proposed changes affect the ability of any safety related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation capabilities.

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

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

There are no changes in the method by which any safety related plant system performs its safety function. The normal manner of plant operation remains unchanged, and no new equipment is being introduced. The increase in the RWST [refueling water storage tank] Level Low-Low Allowable Value still provides acceptable margin between the nominal Trip Setpoint and Allowable Value while taking into account a temperature and boric acid density correction. The change in Allowable Value does not impact the systems capability to perform an ECCS [emergency core cooling system] switchover from injection to cold leg recirculation since the nominal Trip Setpoint remains the same. The change in Allowable Value also will not affect injection or recirculation of the Containment Spray System.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes do not affect the acceptance criteria for any analyzed event nor is there a change in any Safety Analysis Limit. There will be no effect on the manner in which safety limits or Engineered Safety Features Actuation System settings are determined nor will there be any affect on those plant systems necessary to assure the accomplishment of protection functions. Therefore, there will be no impact on any 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 locations:* Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

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

*NRC Section Chief:* Stephen Dembek.

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

*Date of amendment request:* June 11, 1999.

*Description of amendment request:* The amendment would revise Technical Specification 3.7.1.6, ''Steam Generator Atmospheric Relief Valves,'' and its associated Bases to (1) require four atmospheric relief valves (ARVs) to be operable; (2) eliminate the use of ''required'' in the action statements; (3) provide action statements to address inoperability of two ARVs and three or more ARVs due to causes other than excessive leakage; and (4) limit the Limiting Condition for Operation (LCO) 3.0.4 exception to one inoperable ARV.

*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.

Revising the LCO to require four ARVs to be OPERABLE rather than three; eliminating ''required'' from the Actions; adding a new ACTION for three or more ARVs inoperable; and limiting the LCO 3.0.4 exception to one ARV inoperable constitute more restrictive changes from the current Technical Specifications. The proposed changes do not affect initiating mechanisms or mitigation capabilities associated with SGTR [steam generator tube rupture] events analyzed in Chapter 15 of the Updated Safety Analysis Report. The proposed changes impose more stringent requirements to ensure that ARV OPERABILITY is maintained consistent with the safety analysis and licensing basis, and also to address all potential single failure scenarios. Therefore these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

With two ARVs inoperable, the allowed outage time for restoration of all but one ARV to OPERABLE status is changed from 24 hours to 72 hours. The existing specification allows one valve to be inoperable indefinitely and with one required ARV inoperable, the allowed outage time for restoration is seven days. By modifying the LCO to require four ARVs to be OPERABLE, an allowed outage time of 72 hours is more restrictive than the existing specification. Therefore, revising the allowed outage time from 24 hours to 72 hours is acceptable based on a more restrictive allowed outage time from the existing specification and the low probability of an event requiring decay heat removal occurring during the restoration period that would require the ARVs. With respect to Reactor Coolant System cooldown for SGTR accident mitigation, the increase in time is acceptable based on the low probability of a SGTR event occurring during the restoration period and the low probability of a SGTR event in conjunction with the failure of the turbine bypass system (i.e., loss of offsite power). Therefore, this change in allowed outage time does not result in a significant increase in the probability or consequences of previously analyzed accidents.

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

There are no hardware changes nor are there any changes in the method by which any safety related plant system performs its safety function. Revising the LCO to require four ARVs to be OPERABLE rather than three; eliminating ''required'' from the Actions; adding a new ACTION for three or more ARVs inoperable; and limiting the LCO 3.0.4 exception to one ARV inoperable will not impact the normal method of plant operation. The proposed changes ensure operation of the plant remains consistent with analysis assumptions. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. Based on the above discussion, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes do not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any affect on those plant systems necessary to assure the accomplishment of protection functions. The proposed changes ensure operation of the plant consistent with the analysis assumptions. Therefore, there will be no impact on any 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 locations:* Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

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

*NRC Section Chief:* Stephen Dembek.

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

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

*Date of amendment request:* May 24, 1999.

*Description of amendment request:* Clarify nonconservative wording of Technical Specification (TS) 3/4,5,1, ''Safety Injection Tanks,'' and revise TS 3/4.5.2, ''ECCS Subsystems—Tavg Greater Than or Equal to 325 degrees F,'' to align their associated surveillance requirements with the intent and design bases requirements intended to be verified.

*Date of publication of individual notice in the* **Federal Register**: June 10, 1999 (64 FR 31322).

*Expiration date of individual notice:* June 25, 1999.

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

## Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

*Date of application for amendment:* April 12, 1999.

*Brief description of amendment:* The amendment is a temporary amendment change effective until September 30, 1999, which revises Technical Specification 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.

*Date of issuance:* June 4, 1999.

*Effective date:* June 4, 1999.

*Amendment No.:* 183.

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

*Date of initial notice in* **Federal Register**: May 5, 1999 (64 FR 24193).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 4, 1999.

No significant hazards consideration comments received: No.

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

*Commonwealth Edison Company, Docket No. 50–249, Dresden Nuclear Power Station, Unit 3, Grundy County, Illinois*

*Date of application for amendment:* May 5, 1999.

*Brief description of amendment:* The amendment removes the safety valve function of the Target Rock safety/relief valve from Technical Specifications (TS) Section 3.6.E and moves the reactor coolant system safety valve lift pressure setpoints from TS Section 3.6.E to TS Section 4.6.E.

*Date of issuance:* June 4, 1999.

*Effective date:* As of the date of issuance and shall be effective within 30 days from the date of issuance.

*Amendment No.:* 168.

*Facility Operating License No. DPR–25:* The amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: May 21, 1999 (64 FR 27824).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 4, 1999.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* December 7, 1998, as supplemented May 12, 1999.

*Brief description of amendment:* The amendment revises Technical Specification 4.13A.2.a. to allow a one-time extension of the steam generator (SG) inspection interval. In addition, the amendment would remove the requirement of receiving NRC concurrence on the proposed SG examination program in TS 4.13C.1.

*Date of issuance:* June 9, 1999.

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

Amendment No.: 201.

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

*Date of initial notice in* **Federal Register**: February 10, 1999 (64 FR 6694).

The May 12, 1999, supplemental 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 June 9, 1999.

No significant hazards consideration comments received: No.

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

*Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana*

*Date of amendment request:* August 29, 1996, as supplemented January 8, 1998.

*Brief description of amendment:* The proposed changes revise requirements prescribed in Technical Specification Surveillance Requirement 3.3.1.1.8 and allow River Bend to increase the interval between whole core traversing in-core probe to local power range monitor calibrations from 1,000 megawatt days per ton (MWD/T) to 2,000 MWD/T.

*Date of issuance:* June 11, 1999.

*Effective date:* As of the date of issuance and shall be implemented 30 days from the date of issuance.

*Amendment No.:* 107.

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

*Date of initial notice in* **Federal Register**: October 23, 1996 (61 FR 55032).

The January 8, 1998, letter provided additional information that did not change the scope of the original 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 June 11, 1999.

No significant hazards consideration comments received: No.

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

*Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas*

*Date of amendment request:* April 30, 1998.

*Brief description of amendment:* The amendment revises the definition of quadrant power tilt to clearly allow the use of either the incore detectors or the excore detectors for determining quadrant power tilt.

*Date of issuance:* June 10, 1999.

*Effective date:* As of the date of issuance and shall be implemented within 30 days from the date of issuance.

*Amendment No.:* 197.

*Facility Operating License No. DPR–51:* Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: February 10, 1999 (64 FR 6694).

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

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* April 9, 1999.

*Brief description of amendment:* The proposed amendment modifies the Technical Specifications (TSs) to add Limiting Condition for Operation 3.0.6 and its associated Bases. This change allows equipment that has been removed from service or declared inoperable in compliance with the TS Action statement to be returned to service under administrative controls solely to perform testing required to demonstrate its operability or the operability of other equipment. The proposed change is consistent with TS 3.0.5 as discussed in NUREG–1432, Revision 1, ''Standard Technical Specifications for Combustion Engineering Plants.'' TS 3.0.2 is also modified to reflect that TS 3.0.6 is an exception to TS 3.0.2.

*Date of issuance:* June 7, 1999.

*Effective date:* As of the date of issuance and shall be implemented within 30 days from the date of issuance: June 7, 1999.

*Amendment No.:* 207.

*Facility Operating License No. NPF–6:* Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: May 5, 1999 (64 FR 24196).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 7, 1999.

No significant hazards consideration comments received: No.

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

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

*Date of amendment request:* October 1, 1998, as supplemented by letters dated March 25 and May 6, 1999.

*Brief description of amendment:* The amendment modifies Technical Specification (TS) 3.3.3.7.3 and Surveillance Requirement 4.3.3.7.3 for the broad range gas detection system at Waterford 3. In addition, TS Bases 3/4.3.3.7 has been changed to reflect the new system.

*Date of issuance:* June 3, 1999.

*Effective date:* As of the date of issuance and shall be implemented within 90 days from the date of issuance.

*Amendment No.:* 151.

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

*Date of initial notice in* **Federal Register**: November 18, 1998 (63 FR 64114).

The March 25 and May 6, 1999, letters provided clarifying information that did not change the scope of the original 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 June 3, 1999.

No significant hazards consideration comments received: No.

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

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

*Date of amendment request:* January 25, 1999, as supplemented by letter dated April 16, 1999.

*Brief description of amendment:* The amendment removes certain administrative controls from the Waterford 3 Technical Specifications and instead relies on the requirements of the new Entergy common Quality Assurance Program Manual and the change controls of Title 10 of the *Code of Federal Regulations*, Section 50.54(a).

*Date of issuance:* June 16, 1999.

*Effective date:* As of the date of issuance and shall be implemented 60 days from the date of issuance.

*Amendment No.:* 152.

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

*Date of initial notice in* **Federal Register**: February 26, 1999 (64 FR 9192).

The April 16, 1999, letter provided clarifying information that did not change the scope of the original application and expand the initial proposed no significant hazards consideration determination as published in the **Federal Register** notice.

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

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* March 9, 1999.

*Brief description of amendment:* This amendment modifies the Technical Specifications to increase the inservice inspection interval, and reduces the scope of volumetric and surface examinations for the reactor coolant pump flywheels.

*Date of issuance:* June 8, 1999.

*Effective date:* June 8, 1999.

*Amendment No.:* 232.

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

*Date of initial notice in* **Federal Register**: May 5, 1999 (64 FR 24196).

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

No significant hazards consideration comments received: No.

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

*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:* September 30, 1998.

*Brief description of amendment:* The amendment corrected the description of the reactor coolant system leakage detection capability of the reactor building atmosphere gaseous radioactivity monitor in the Improved Technical Specification Bases and the Final Safety Analysis Report.

*Date of issuance:* June 14, 1999.

*Effective date:* June 14, 1999.

*Amendment No.:* 179.

*Facility Operating License No. DPR–31:* Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: November 18, 1998 (63 FR 64116).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 14, 1999.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* December 3, 1996.

*Brief description of amendment:* The amendment incorporates certain improvements from the Standard Technical Specifications for Babcock and Wilcox plants (NUREG–1430).

*Date of issuance:* June 15, 1999.

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

*Amendment No.:* 211.

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

*Date of initial notice in* **Federal Register**: December 18, 1996 (61 FR 66708).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 15, 1999.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* January 22, 1999.

*Brief description of amendment:* Revises Technical Specification (TS) Section 4.3, ''Fuel Storage,'' by updating the criticality requirements (k-infinity and U–235 enrichment limits) for storage of fuel assemblies in the spent fuel racks. This change would allow for storage of nuclear fuel assemblies with new designs, including GE–12 with a 10X10 pin array.

*Date of issuance:* June 8, 1999.

*Effective date:* June 8, 1999.

*Amendment No.:* 226.

*Facility Operating License No. DPR–49:* The amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: February 24, 1999 (64 FR 9192).

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

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* October 15, 1998, as supplemented on December 21, 1998.

*Brief description of amendment:* Revise the Technical Specifications (TS) by adding a new TS 3.7.9, ''Control Building/Standby Gas Treatment System Instrument Air System,'' and revises (TS) 3.6.1.3, ''Primary Containment Isolation Valves,'' Condition E.

*Date of issuance:* June 9, 1999.

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

*Amendment No.:* 227.

*Facility Operating License No. DPR–49:* The amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: February 24, 1999 (64FR9193).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 9, 1999.

No significant hazards consideration comments received: No.

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

*Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan*

*Date of application for amendments:* April 19, 1999.

*Brief description of amendments:* The amendments revise Technical Specification (TS) 3/4.8.1.2, ''Electrical Power Systems, Shutdown,'' and its associated bases to provide a one-time extension of the 18-month surveillance interval for specific surveillance requirements associated with the emergency diesel generators for Units 1 and 2. The surveillances will be performed prior to the first entry into Mode 4 following the current plant shutdown. In addition, for Unit 2 only, a minor administrative change is included to delete a reference to TS 4.0.8, which is no longer applicable. For Unit 1 only, an editorial change is made to add the word ''or'' to action statement 3.8.1.2.

*Date of issuance:* June 8, 1999.

*Effective date:* June 8, 1999, with full implementation within 45 days.

*Amendment Nos.:* 228 and 211.

*Facility Operating License Nos. DPR–58 and DPR–74:* Amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: April 29, 1999 (64 FR 23129).

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

No significant hazards consideration comments received: No.

*Local Public Document Room location:* Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085.

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

*Date of application for amendment:* May 15, 1998, as supplemented by letters dated September 25, October 13, December 9 (two letters), 1998; January 11, April 1, and April 22, 1999.

*Brief description of amendment:* This amendment changes Technical Specification (TS) 5.5, ''Storage of Unirradiated and Spent Fuel,'' to reflect a planned modification to increase the storage capacity of the spent fuel pool from 2776 to 4086 fuel assemblies. It also deletes an inappropriate statement and reference within TS 5.5.

*Date of issuance:* June 17, 1999.

*Effective date:* This license amendment is effective as of the date of its issuance to be implemented before spent fuel is stored within the new high-density spent fuel rack modules authorized for installation and use by this amendment.

*Amendment No.:* 167.

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

*Date of initial notice in* **Federal Register**: November 24, 1998 (63 FR 64973).

The September 25, October 13, December 9 (two letters) 1998, January 11, April 1, and April 22, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration.

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

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.

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

*Date of application for amendments:* January 21, 1999, which superseded application dated July 22, 1998.

*Brief description of amendments:* The amendments revise the Technical Specifications high radiation trip setpoints for the reactor building and the refueling floor ventilation exhaust monitors.

*Date of issuance:* June 9, 1999.

*Effective date:* As of the date of issuance and shall be implemented within 30 days from the date of issuance.

*Amendment Nos.:* Unit 1—216; Unit 2—157.

*Facility Operating License Nos. DPR–57 and NPF–5:* Amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: May 5, 1999 (64 FR 24200); this supersedes the original notice dated August 26, 1998 (63 FR 45529).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 9, 1999.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* Appling County Public Library, 301 City Hall Drive, Baxley, Georgia.

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

*Date of amendment request:* March 30, 1999.

*Brief description of amendments:* The amendments deleted Technical Specification 3/4.3.3.4, ''Meteorological Instrumentation,'' and its associated Bases. These requirements have already been relocated to the Technical Requirements Manual (TRM). Because the TRM is incorporated within the South Texas Project updated final safety analysis report for the units, changes to the relocated requirements will be controlled by 10 CFR 50.59.

*Date of issuance:* June 16, 1999.

*Effective date:* June 16, 1999, to be implemented within 30 days.

*Amendment Nos.:* Unit 1—111; Unit 2—98.

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

*Date of initial notice in* **Federal Register**: May 5, 1999 (64 FR 24201).

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

No significant hazards consideration comments received: No.

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

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

*Date of application for amendments:* February 16, 1999.

*Brief Description of amendments:* The amendments revise Technical Specifications (TS) Sections 3.6, 3.9, and 3.16 and the associated Bases for those sections for Units 1 and 2. The changes consolidate the auxiliary feedwater cross-connect requirements by relocating the electrical power requirements from Section 3.16 to Section 3.6. The TS are also clarified with regard to permitting simultaneous entry into certain conditions of operation on Units 1 and 2.

*Date of issuance:* June 7, 1999.

*Effective date:* June 7, 1999.

*Amendment Nos.:* 220 and 220.

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

*Date of initial notice in* **Federal Register**: May 5, 1999 (64 FR 24203).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 7, 1999.

No significant hazards consideration comments received: No.

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

**Notice of Issuance of Amendment to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)**

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 application for amendment, (2) the amendment to Facility Operating License, 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 room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By July 30, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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 the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

*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:* June 10, 1999.

*Brief description of amendments:* The amendments revised the Technical Specifications TS 3.7.9, ''Control Room Area Ventilation System (CRAVS),'' to establish actions to be taken for an inoperable control room ventilation system due to a degraded control room pressure boundary. This revision approves a one-time-only action for two CRAVS trains inoperable due to a degraded control room boundary in Modes 1, 2, 3, and 4, that is to be completed within 24 hours. The applicable TS Bases have been revised to document the TS changes and to provide supporting information.

*Date of issuance:* June 11, 1999.

*Effective date:* As of the date of issuance and shall be implemented upon receipt.

*Amendment Nos.:* Unit 1—185; Unit 2—167.

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

The Commission's related evaluation and the amendment, finding of emergency circumstances, consultation with the State of North Carolina, and final no significant hazards consideration determination are contained in a Safety Evaluation dated June 11, 1999.

*Attorney for licensee:* Mr. Albert Carr, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

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

*NRC Section Chief:* Richard L. Emch, Jr.

Dated at Rockville, Maryland, this 23rd day of June 1999.

For the Nuclear Regulatory Commission.

**John A. Zwolinski,**

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

[FR Doc. 99–16489 Filed 6–29–99; 8:45 am]

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## RAILROAD RETIREMENT BOARD

### Determination of Quarterly Rate of Excise Tax for Railroad Retirement Supplemental Annuity Program

In accordance with directions in Section 3221(c) of the Railroad Retirement Tax Act (26 U.S.C., Section 3221(c)), the Railroad Retirement Board has determined that the excise tax imposed by such Section 3221(c) on every employer, with respect to having individuals in his employ, for each work-hour for which compensation is paid by such employer for services rendered to him during the quarter beginning July 1, 1999, shall be at the rate of 27 cents.

In accordance with directions in Section 15(a) of the Railroad Retirement Act of 1974, the Railroad Retirement Board has determined that for the quarter beginning July 1, 1999, 35.8 percent of the taxes collected under Sections 3211(b) and 3221(c) of the Railroad Retirement Tax Act shall be credited to the Railroad Retirement Account and 64.2 percent of the taxes collected under such Sections 3211(b) and 3221(c) plus 100 percent of the taxes collected under Section 3221(d) of the Railroad Retirement Tax Act shall be credited to the Railroad Retirement Supplemental Account.

By Authority of the Board.

Dated: June 21, 1999.

**Beatrice Ezerski,**

*Secretary to the Board.*

[FR Doc. 99–16569 Filed 6–29–99; 8:45 am]

BILLING CODE 7905–01–M