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DOCKET NO(S). 50-397
Mr. G. C. Sorensen, Manager
Regulatory Programs
Washington Public Power Supply System
P. O. Box 968
3000 George Washington Way
Richland, Washington 99352
SUBJECT: WASHINGTON PUBLIC POWER SUPPLY SYSTEM - WNP-2

The following documents concerning our review of the subject facility are transmitted for your information.

- [ ] Notice of Receipt of Application, dated _____.
- [ ] Draft/Final Environmental Statement, dated _____.
- [ ] Notice of Availability of Draft/Final Environmental Statement, dated _____.
- [ ] Safety Evaluation Report, or Supplement No._____ dated _____.
- [ ] Environmental Assessment and Finding of No Significant Impact, dated _____.
- [ ] Notice of Consideration of Issuance of Facility Operating License or Amendment to Facility Operating License, dated _____.
- [x] Bi-Weekly Notice; Applications and Amendments to Operating Licenses Involving No Significant Hazards Considerations, dated 7/15/87 [*see page(s)*] 26608.
- [ ] Exemption, dated _____.
- [ ] Construction Permit No. CPPR-_____, Amendment No. _____ dated _____.
- [ ] Facility Operating License No. _____, Amendment No. _____ dated _____.
- [ ] Order Extending Construction Completion Date, dated _____.
- [ ] Monthly Operating Report for _____ transmitted by letter dated _____.
- [ ] Annual/Semi-Annual Report- _____ _____ transmitted by letter dated _____.

Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See next page

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| OFFICE | DRSP/PD5 | | | | | | |
| SURNAME | JLee | | | | | | |
| DATE | 7/23/87 | | | | | | |

NRC FORM 318 (10/80) NRCM 0240 OFFICIAL RECORD COPY

Mr. G. C. Sorensen, Manager
Washington Public Power Supply System

WPPSS Nuclear Project No. 2
(WNP-2)

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## NUCLEAR REGULATORY COMMISSION

### Bi-weekly Notice Applications and Amendments to Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to Public Law (P.L.) 97-415, the Nuclear Regulatory Commission (the Commission) is publishing this regular bi-weekly notice. P.L. 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 bi-weekly notice includes all notices of amendments issued, or proposed to be issued from June 22, 1987 through July 2, 1987. The last bi-weekly notice was published on July 1, 1987 (52 FR 24542).

## NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION AND OPPORTUNITY FOR 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 amendments 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. The Commission will not normally make a final determination unless it receives a request for a hearing.

Written comments may be submitted by mail to the Rules and Procedures Branch, Division of Rules and Records, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this Federal Register notice. Written comments may also be delivered to Room 4000, Maryland National Bank Building, 7735 Old Georgetown Road, Bethesda, Maryland from 8:15 a.m. to 5:00 p.m. Copies of written comments received may be examined at the NRC Public Document Room, 1717 H Street, NW., Washington, DC. The filing of requests for hearing and petitions for leave to intervene is discussed below.

By August 14, 1987, 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 petition for leave to intervene. Requests for a hearing and petitions 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. 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 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 fifteen (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 fifteen (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, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. 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 involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

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 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.

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, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, NW., Washington, DC, by the above date. Where petitions are filed during the last ten (10) days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at (800). 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to (*Project Director*): petitioner's name and telephone number; date petition was mailed; plant

name; and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel-Bethesda, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding 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, 1717 H Street, N.W., Washington, D.C., and at the local public document room for the particular facility involved.

**Alabama Power Company, Docket Nos. 50-348 and 50-364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama**

*Date of amendments request:* August 25, 1986, superseded June 2, 1987

*Description of amendments request:* The proposed amendment was first noticed and described in 51 FR 36082 dated October 7, 1986. The description is briefly restated as follows: changes would revise Figure 2.1-1 to increase the steam generator tube plugging from 5% to 10% and would revise the $F_Q$ coefficient of Technical Specification 3.2.2 and Bases 3/4.2.1 to be 2.32 for greater than 50% rated thermal power and 4.64 for less than or equal to 50% rated thermal power. The Technical Specification changes are the same as the changes noticed previously.

*Basis for proposed no significant hazards consideration determination:* During the NRC staff review of the August 25, 1986, submittal the licensee was notified that Generic Letter 88-16 dated October 22, 1986, relating to BART code errors must be considered prior to further action on the licensee request. As a result Westinghouse performed the large break LOCA analysis for the licensee. The new analysis used the Westinghouse 1981 ECCS Large Break Evaluation Model (WCAP-9220-P-A and WCAP-9221) with BASH (WCAP-10266, Revision 2). Further enhancements to the BASH code and methodology are shown in Addendum 2 to WCAP-10266, Revision 2 and are used in the licensee's submittal as the basis for the changes.

The licensee provided the following analysis of the significant hazards considerations per the requirements of 10 CFR 50.92:

(1)The proposed changes will not increase the probability or consequences of any accident previously evaluated because the revised ECCS analysis provided... was performed to support these changes, has demonstrated that the acceptance criteria for 10 CFR 50.46 have been met. The proposed changes have also been demonstrated to have no impact on the conclusions of the small break LOCA analysis and all the non-LOCA transients or Reactor Coolant System (RCS) structural integrity. Therefore, the probability or consequences of any accident previously evaluated will not be increased.

(2)The proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated because both changes consist of changes to assumptions in previously evaluated accidents. Additionally, the increase in steam generator tube plugging has been evaluated for impact on RCS average temperature, thermal design flow and secondary side pressure and determined to have no impact on current plant operating limits for these parameters.

Furthermore, the increase in the steam generator tube plugging limit will have no effect on RCS structural integrity. Thus, these proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3)The proposed changes will not involve a reduction in a margin of safety because RCS structural integrity is maintained and the revised ECCS analysis has demonstrated the requirements of 10 CFR 50.46 are met. Additionally, the calculated peak clad temperature from this revised analysis is even less than the present Farley analysis and provides additional margin to the limit of 2200&F. Therefore, these proposed changes will not involve a reduction in a margin of safety. analysis has demonstrated the requirements of 10 CFR 50.46 are met. Additionally, the calculated peak clad temperature from this revised analysis is even less than the present Farley analysis and provides additional margin to the limit of 2200&F. Therefore, these proposed changes will not involve a reduction in a margin of safety.

Based upon the analysis provided above, the licensee, Alabama Power Company, determined that the proposed changes to the Technical Specifications will not increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a reduction in a margin of safety. Thus, the licensee determined that these proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards considerations.

The NRC staff agrees with the licensee's analysis that the action would involve no significant hazards consideration. Also, as stated in our previous notice (51 FR 36082), Commission example "(vi) A change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method," fits the proposed changes. Therefore, we propose to determine that the amendment does not involve a significant hazards consideration.

*Local Public Document Room location:* George S. Houston Memorial Library, 212 W. Burdeshaw Street, Dothan, Alabama 36303

*Attorney for licensee:* Ernest L. Blake, Esquire, 2300 N Street, NW., Washington, DC 20037

*NRC Project Director:* Elinor G. Adensam

**Arizona Public Service Company, et al., Docket No. STN 50-528, Palo Verde Nuclear Generating Station (PVNGS), Unit 1, Maricopa County, Arizona**

*Date of amendment request:* May 11, 1987

*Description of amendment request:* The proposed amendment consists of a proposed change to the Technical Specifications (Appendix A to Facility Operating License No. NPF-41 for PVNGS, Unit 1).

Technical Specification 2.2, "Reactor Trip Setpoints," provides setpoints for various parameters in Table 2.2-1. Technical Specification 3/4.3.2, "Engineered Safety Features Actuation System Instrumentation," provides the required number of operable channels for various parameters in Table 3.3-3. Both Table 2.2-1 and Table 3.3-3 include notations which provide the allowances for manually reducing the trip setpoints for low pressurizer pressure and low steam generator pressure. Currently those notations state that the applicable modes for those allowances are Modes 3-6. The proposed amendment would change the applicable modes to Modes 3-4 on the basis that the current Technical Specifications do not require these protection functions to be operable in Modes 5 and 6. The proposed change would also make those portions of the Technical Specifications consistent with the Technical Specifications for Palo Verde, Units 2 and 3 (Appendix A to Facility Operating.

License Nos. NPF-51 and NPF-65, respectively) previously reviewed and approved by the staff.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards considerations if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

A discussion of the proposed change, as it relates to these standards is presented below.

*Standard 1 - Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated*

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated since the change does not alter the design of the plant or method of operation for the facility. The only change being requested is to achieve consistency between the notations in Tables 2.2-1 and 3.3-3, and the limiting conditions for operation in Specifications 2.2 and 3/4.3.2.

*Standard 2 - Create the Possibility of a New or Different Kind of Accident from any Accident Previously Evaluated*

The proposed change is only an administrative change and does not involve any changes to plant equipment or plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously analyzed will not be created.

*Standard 3 - Involve a Significant Reduction in a Margin of Safety*

The proposed change does not alter any limiting condition for operation, any action statement, or any surveillance requirement currently in the Technical Specifications. Therefore, a significant reduction in a margin of safety is not involved.

Accordingly, the Commission has proposed to determine that the above change does not involve a significant hazards consideration.

*Local Public Document Room location:* Phoenix Public Library, Business, Science and Technology Department, 12 East McDowell Road, Phoenix, Arizona 85004.

*Attorney for licensees:* Mr. Arthur C. Gehr, Snell & Wilmer, 3100 Valley Center, Phoenix, Arizona 85007.

*NRC Project Director:* George W. Knighton

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

*Date of application for amendment:* May 22, 1987

*Description of amendment request:* Changes in the Technical Specifications are requested for Cycle 8 operation of the nuclear reactor following core reload 7. For the first time, only retrofit fuel will be loaded into the core. All non-retrofit fuel will be discharged to the spent fuel pool. The proposed Technical Specification changes are: (1) removing reference to the non-retrofit 8x8 fuel; (2) revising the description of low and low-low reactor water level setpoints to reflect the change in height of the top of the active fuel length; (3) slightly reducing the operating limit minimum critical power ratio to permit operational flexibility; and (4) several editorial changes to correct the spelling of "MFLPD', identify the unit of measurement as megawatt days per standard ton, and include a reference which had inadvertently been deleted.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided guidance for determining whether a proposed amendment involves a significant hazards consideration (48 FR 14870). Examples of amendments that are not likely to involve a significant hazards consideration are "(i) A purely administrative change... to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature." and "(iii)..., a change resulting form a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the Technical Specifications, that the analytical methods used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable."

The staff considers the proposed amendment to be similar to example (i) since it corrects several editorial errors and identifies consistent units of measurement. In addition, the proposed amendment is similar to example (iii) since the staff has previously reviewed the retrofit fuel design and found that the operating characteristics and safety margins are acceptable. No changes in the previously accepted analytical methods used to demonstrate conformance with the Technical Specifications and regulations are involved. Therefore, no significant difference in safety to the public is expected.

Since the amendment involves proposed changes for which no significant hazards consideration exists, the staff has made a proposed determination that this application for amendment involves no significant hazards consideration.

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

*Attorney for licensee:* W. S. Stowe, Esq., Boston Edison Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199.

*NRC Project Director:* Victor Nerses, Acting Director

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

*Date of amendment request:* November 17, 1986, as supplemented July 1, 1987

*Description of amendment request:* The proposed amendments would revise Technical Specification (TS) Tables 3.8-1A and 3.8-1B to add two Containment Penetration Conductor Overcurrent Protective Devices to Table 3.8-1A for Unit 1 and Table 3.8-1B for Unit 2. These four devices were added at Catawba Units 1 and 2 to accommodate decontamination pressure washers and welding machines. Thus, the corresponding tables must be revised to reflect these additions.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided certain examples (51 FR 7744) of actions likely to involve no significant hazards considerations. The request involved in this case does not match any of those examples. However, the staff has reviewed the licensee's request for the above amendments and determined that should this request be implemented, it would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated because the addition of the four overcurrent protective devices would not significantly affect the operation of the station. These devices would be required to be operable through TS 3.8.4 Limiting Condition for Operation. Also, the licensee's proposed

revisions would not (2) create the possibility of a new or different kind of accident from any accident previously evaluated because the design of the facility would not be significantly affected. Only two overcurrent protective devices per Unit are added to the over 200 such devices already existing at each Unit. Futhermore, adding these devices would provide additional protection which otherwise may not be available. Finally, the proposed revisions would not (3) involve a significant reduction in a margin of safety because of the reasons stated above in items (1) and (2).

Accordingly, the Commission has determined that the above changes involve no significant hazards consideration.

*Local Public Document Room location:* York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

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

*NRC Project Director:* B. J. Youngblood

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

*Date of amendment request:* June 3, 1987

*Description of amendment request:* The proposed amendments would revise Technical Specification (TS) 3/4.1.3 "Movable Control Assemblies" and its associated Bases 3/4.1.3 to allow the unit to remain at power (Modes 1 or 2) for up to 72 hours with more than one full-length rod inoperable but trippable.

The existing TS does not distinguish between immovable rods and immovable but trippable rods, and with more than one full-length control rod immovable, the unit is presently required by existing Action Statements to be in hot standby within 6 hours.

The specific change would be implemented by deleting "inoperable" from Action Statement b (which presently addresses both inoperable or misaligned rods), and adding a new Action Statement d which would require that, with more than one full-length rod trippable but inoperable due to causes other than being immovable as a result of excessive friction or mechanical interference or known to be untrippable, power operation may continue provided that (1) within one hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within +12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of existing TS Figure 3.1-1a or Figure 3.1-1b, as applicable, and providing that (2) thermal power level is restricted pursuant to existing TS 3.1.3.6 during subsequent operation. These two provisions regarding the remainder of the rods correspond to the existing requirements of Action Statement c which addresses no more than one full-length rod trippable but inoperable for the same causes. The new Action Statement d would also include a provision that (3) the inoperable rods are to be returned to operable status within 72 hours. Futhermore, Action Statement c.2 would be revised to change "Figure 3.1-1" to "Figure 3.1-1a or Figure 3.1-1b, as applicable."

Except that the "inoperable" portion of Action Statement b would be addressed separately by the proposed new Action Statement d and the administrative change to Action Statement c.2, the proposed amendments would not otherwise change TS Action Statements a, b or c.

*Basis for proposed no significant hazards consideration determination:* By letter dated December 31, 1984, Westinghouse recommended generic revisions to TS 3/4.1.3 and its Bases regarding multiple immovable, but trippable, control rods. Westinghouse noted they experienced several different plants with its NSSS in which a group or several groups of control rods became immovable (would not step in or out) because of a rod control system failure. The licensee cited similar experiences during control rod movement periodic tests in which control malfunctions prohibited a control rod bank or group from moving when selected, including a recent occurrence at Catawba Unit 2 when a fuse blew in the rod control circuitry. In all cases, the rods were still trippable (i.e., these rods would have been inserted into the core in the event of a reactor trip signal and, thus, were fully capable of performing their intended safety function of shutting down the reactor). The NRC has previously accepted the Westinghouse recommended generic revision and has incorporated the change into the TS of several operating plants (e.g., Diablo Canyon and Joseph Farley) and into a proposed revision to the Standard Technical Specifications (NUREG-0452). The additional time provided by the change to find and repair the cause of the rods' inoperability is justified because the rods are trippable.

The Commission has provided guidance concerning the application of its standards set forth in 10 CFR 50.92 for no significant hazards consideration by providing certain examples (51 FR 7744). The changes for the proposed amendments (except for the administrative change to Action Statement c.2) do not match those examples. However, on the basis of its previous reviews of similar Westinghouse plants and the licensee's submittal, and because trippability of the control rods is not affected by the proposed change, previous assumptions and results of accident analyses and the reliability of the reactor protection would not adversely change. Therefore, the change would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) involve a significant reduction in a margin of safety. The proposed change also would not involve any change in the design, nor any changes (other than the time allowed to find the cause and to repair the trippable rods' inability to be stepped into or out of the core) in the operation of the plant. Therefore, the proposed change would not (3) create the possibility of a new or different kind of accident from any accident previously evaluated.

One of the Commission's examples in 51 FR 7744 of actions likely to involve no significant hazards considerations is (i), "a purely administrative change to technical specifications: for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature." The proposed revision to Action Statement c.2 to change "Figure 3.1-1" to "Figure 3.1-1a or Figure 3.1-1b, as applicable," represents a revision that meets the guidance provided by this example.

Accordingly, the Commission proposes to determine that the requested license amendments involve no significant hazards consideration.

*Local Public Document Room location:* York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

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

*NRC Project Director:* B. J. Youngblood

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

*Date of amendment request:* June 3, 1987

*Description of amendment request:* The proposed amendments would revise Technical Specification (TS) 3/4.1.3 "Movable Control Assemblies" and its associated Bases 3/4.1.3 to allow the

unit to remain at power (Modes 1 or 2) for up to 72 hours with more than one full-length rod inoperable but trippable. The existing TS does not distinguish between immovable rods and immovable but trippable rods, and with more than one full-length control rod immovable, the unit is presently required by existing Action Statements to be in hot standby within 6 hours.

The specific change would be implemented by deleting "inoperable" from Action Statement b (which presently addresses both inoperable or misaligned rods), and adding a new Action Statement d. New Action Statement d would require that, with more than one full-length rod trippable but inoperable due to causes other than being immovable as a result of excessive friction or mechanical interference or known to be untrippable, power operation may continue provided that (1) within one hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within +12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of existing TS Figure 3.1-1, and providing that (2) thermal power level is restricted pursuant to existing TS 3.1.3.6 during subsequent operation. These two provisions regarding the remainder of the rods correspond to the existing and unchanged requirements of Action Statement c which addresses no more than one full-length rod trippable but inoperable for the same causes. New Action Statement d would also include a provision that (3) the inoperable rods are to be returned to operable status within 72 hours.

Except that the "inoperable" portion of Action Statement b would be addressed separately by the proposed new Action Statement d, the proposed amendments would not otherwise change TS Action Statements a, b or c.

*Basis for proposed no significant hazards consideration determination:* By letter dated December 31, 1984, Westinghouse recommended generic revisions to TS 3/4.1.3 and its Bases regarding multiple immovable, but trippable, control rods. Westinghouse noted the experience at several different plants with its NSSS in which a group or several groups of control rods became immovable (would not step in or out) because of a rod control system failure. The licensee cited similar experiences during control rod movement periodic tests in which control malfunctions prohibited a control rod bank or group from moving when selected, including a recent occurrence at Catawba Unit 2 when a fuse blew in the rod control circuitry. In all cases, the rods were still trippable (i.e., these rods would have been inserted into the core in the event of a reactor trip signal and, thus, were fully capable of performing their intended safety function of shutting down the reactor). The NRC has previously accepted the Westinghouse recommended generic revision and has incorporated the change into the TS of several operating plants (e.g., Diablo Canyon and Joseph Farley) and into a proposed revision to the Standard Technical Specification (NUREG-0452). The additional time provided by the change to find and repair the cause of the rods' inoperability is justified because the rods are trippable.

The Commission has provided guidance concerning the application of its standards set forth in 10 CFR 50.92 for no significant hazards consideration by providing certain examples (51 FR 7744). The changes for the proposed amendments do not match those examples. However, on the basis of its previous reviews of similar Westinghouse plants and the licensee's submittal, and because trippability of the control rods is not affected by the proposed change, previous assumptions and results of accident analyses and the reliability of the reactor protection would not adversely change. Therefore, the change would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) involve a significant reduction in a margin of safety. The proposed change also would not involve any change in the design, nor any changes (other than the time allowed to find the cause and to repair the trippable rods' inability to be stepped into or out of the core) in the operation of the plant. Therefore, the proposed change would not (3) create the possibility of a new or different kind of accident from any accident previously evaluated. Accordingly, the Commission proposes to determine that the requested license amendments involve no significant hazards consideration.

*Local Public Document Room location:* Atkins Library, University of North Carolina, Charlotte (UNCC) Station, North Carolina 28223.

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

*NRC Project Director:* B. J. Youngblood

**Duquesne Light Company, Docket No. 50-334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania**

*Date of amendment request:* April 7, 1987

*Description of amendment request:* The amendment request covers a number of areas in the Technical Specifications, and License Condition 2.C.(5). Associated Updated Final Safety Analysis Report (UFSAR) pages will also be changed. The affected Technical Specifications are:

1. Section 3.3.3.6 (Fire Detection Instrumentation) would be deleted and its requirements would be incorporated in the UFSAR and plant operational procedures. For clerical clarity a note would be added to page 3/4 3-46 (last page preceding Section 3.3.3.6) to indicate the deletion. Section 6.9.2, item f would also be deleted to reflect this change. Item 18 of Technical Specification Table 3.3-10 would be replaced by Primary Auxiliary Building smoke detection instruments which are included in administrative procedures. Item 18, which provides fire detection instruments for the Unit 2 control room zones 7, 8 and 9, and Chemical Addition Building smoke detectors will be incorporated in administrative procedures.

2. Section 3.7.14.1 (Fire Suppression Water System) would be deleted and its requirements, with the exception of special reporting requirements, would be incorporated in the UFSAR and plant operational procedures. To reflect this change Section 6.9.2 item g would also be deleted.

3. Section 3.7.14.2 (Spray and/or Sprinkler Systems), 3.7.14.3 ($CO_2$ Systems), 3.7.14.4 (Fire Hose Stations), 3.7.14.5 (Halon Systems), and 3.7.15 (Fire Barrier Penetrations) would be deleted and their requirements would be incorporated in the plant operational procedures. Section 6.9.2 item g would be deleted to reflect these changes.

4. Section 6.2.2.f Site Fire Brigade Requirements and Section 6.4.2 would be deleted, and their requirements would be transferred to the plant operational procedures.

5. License condition 2.C.(5) would be replaced by the standard condition provided in section F of Generic Letter 86-10, Implementation of Fire Protection Requirements. The current license condition 2.C.(5) refers to modifications identified in Table 1 of the Fire Protection Safety Evaluation Report for BVPS, Unit 1, dated May 9, 1979. These modifications have all been completed, therefore, this condition is no longer necessary.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed

amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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 first four changes do not involve a significant increase in the probability or consequence of an accident previously evaluated because, other than certain special reporting requirements for the Fire Suppression Water System, requirements have not been changed in their relocations. Since the amendment relocates rather than changes these requirements, it does not create the possibility of a new or different kind of accident from an accident previously evaluated.

The special reporting requirements for the Fire Suppression Water System which were not incorporated in the UFSAR and the plant operating procedures are covered to the extent that they are significant by the reporting requirements of 10 CFR 50.72 and 10 CFR 50.73. Furthermore, since the first four modifications do not change requirements, no significant reduction in margin of safety results from them.

The replacement of License Condition 2.C.(5) with the standard condition from Generic Letter 86-10 is consistent with the Technical Specification changes above. This proposed amendment does not change the overall fire protection requirements, and therefore does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident from an accident previously evaluated; nor does it involve a significant reduction in a margin of safety.

Accordingly, the staff has made a proposed determination that the requested license amendment does not involve a significant hazards consideration.

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

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

*NRC Project Director:* John F. Stolz

**Duquesne Light Company, Docket No. 50-334, Beaver Valley Power Station Unit No. 1, Shippingport, Pennsylvania**

*Date of amendment request:* April 30, 1987

*Brief description of amendment request:* The proposed amendment would clarify the technical specifications governing boron dilution in the reactor coolant system (RCS) as follows:

(1)Section 3.1.1.3 would be revised to clearly define "boron dilution" as addition of water whose boron concentration is less than that required for the shutdown margin.

(2)Section 3.4.1.4 and corresponding basis would be revised so that an isolated loop must have coolant boron concentration equal to or greater than that required by Section 3.1.1.2 and 3.9.1 before the loop is unisolated.

(3)Section 4.9.8.1 would be revised to clearly define "dilution".

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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 licensee stated that the current Technical Specifications cause unnecessary delays in operation. The proposed changes would only clarify current requirements and the definition of "boron dilution". No hardware modification is involved and there is no relaxation of safety acceptance criteria. Thus the answers to the three points above are negative. The staff concurs with the licensee's assessment and proposes to determine that the requested amendment involves no significant hazards considerations.

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

*Attorney for licensee:* Gerald Charnoff, Esquire, Jay E. Silberg, Esquire, Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20036

*NRC Project Director:* John F. Stolz

**Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50-366, Edwin I. Hatch Nuclear Plant, Unit 2, Appling County, Georgia**

*Date of amendment request:* March 20, 1987

*Description of amendment request:* This amendment would modify the Technical Specifications to permit temporary adjustments to trip setpoints for the Main Steam Line Radiation Montior (MSLRM) instruments as described in Tables 3.2.1-1, 3.3.1-1, 3.3.2-1, 3.3.2-2, 3.3.6.7-1 and 3.3.6.7-2 to allow performance of tests of hydrogen injection into the primary coolant. These tests will be performed in order to evaluate Hydrogen Water Chemistry (HWC) as a potential mitigator of intergranular stress corrosion cracking (IGSCC). This amendment also would correct a typographical error in Table 3.3.6.7-1.

The temporary change proposed would permit the normal full power background level, associated with the Main Steam High Radiation scram and isolation setpoints, to be increased so as to compensate for the anticipated increase in the main steam radiation levels during hydrogen injection. This background radiation level increase when hydrogen injection is underway is caused by higher levels of short half-life N-16 carryover into the main steam.

The proposed modification would allow this temporary adjustment to the setpoints to be made only when above 20 percent of rated power and would require that it be made within 24 hours prior to planned start of hydrogen injection. It would require that normal setpoints be established within 24 hours of reestablishing normal radiation levels after completion of the hydrogen injection and prior to establishing power levels below 20 percent rated power. It also would permit adjustments to the setpoints during the test based on either calculations or measurements of actual radiation levels resulting from hydrogen injection.

A similar change was approved for the purpose of hydrogen injection tests at Hatch Unit 1 by amendment 125 to the Hatch Unit 1 license, dated May 21, 1986.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards

consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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 current setpoints are less than or equal to 3 times normal rated power background. The only design basis accident which takes credit for main steam isolation valve (MSIV) closure on Main Steam Line high radiation is the design basis control rod drop accident (CRDA). The CRDA is only of concern below 10 percent of rated power. Since the current MSLRM setpoint will not be changed when at or below 20 percent rated power, the MSLRM sensitivity to fuel failure is not impacted, and the FSAR analysis for the CRDA remains valid.

The MSLRM also performs a general function of monitoring for failed fuel. This capability to monitor fuel failures is retained with the adjusted radiation "background" setpoint. Additionally, this fuel failure monitoring capability is provided through the offgas radiation monitor, performance of primary coolant analyses, and routine radiation surveys.

If, due to a recirculation pump trip or other unanticipated power reduction event, the reactor drops below 20 percent rated power without setpoint readjustment, control rod withdrawal will be prohibited by procedures until the necessary setpoint readjustment is made. This ensures that fuel failures of the type concerning the MSLRM (specifically FSAR CRDA analysis) are unlikely.

Table 3.3.6.7.1 was added to the Technical Specifications by Amendment 71. Part of that change involved moving the requirements for the control room air inlet radiation detectors from previous TS 4.7.2.e.3.g to the new Table 3.3.6.7-1. In making that change, the new table incorrectly referenced note (d) instead of note (c) as pertaining to these detectors. This change would correct the reference to note (c) as was originally intended.

On the basis of the above, the Commission has determined that the requested amendment meets the three criteria and therefore has made a proposed determination that the amendment application does not involve a significant hazards consideration.

*Local Public Document Room location:* Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

*Attorney for licensee:* Bruce W. Churchill, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037

*NRC Project Director:* B. J. Youngblood

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

*Date of amendment request:* February 13, 1987

*Description of amendment request:* These proposed changes would modify the Unit 1 and Unit 2 Technical Specifications to: (1) incorporate the revised reporting requirements of 10 CFR Part 50, Sections 50.72 and 50.73, as directed by NRC Generic Letter 83-43, "Report Requirements of 10 CFR Part 50, Sections 50.72 and 50.73, and Standard Technical Specifications," dated December 19, 1983; and (2) incorporate the revised reporting requirements for primary coolant iodine spiking and remove existing requirements for plant shutdown if primary coolant iodine activity limits are exceeded for 800 hours within a 12-month period, as requested by NRC Generic Letter 85-19, "Reporting Requirements on Primary Coolant Iodine Spikes," dated September 27, 1985.

Reporting requirements for Hatch Units 1 and 2 were established and the Technical Specifications for these plants were issued prior to codification of reporting requirements as now specified by Sections 50.72 and 50.73 of 10 CFR Part 50, which became effective on January 1, 1974. Generic Letter 83-43 informed all licensees that they should amend the Technical Specifications for their plants to conform to the new reporting requirements. This amendment would conform the Technical Specifications for Hatch Units 1 and 2 to the reporting requirements of Sections 50.72 and 50.73 of 10 CFR Part 50.

Technical Specifications prior to the issuance of Generic Letter 85-19 on September 27, 1985, were written to require short-term reporting of spikes in iodine activity in the primary coolant, and plant shutdown in the event accumulated time at high iodine activity levels exceeded 800 hours in a 12-month period. Generic Letter 85-19 announced that the NRC had determined these requirements to be unnecessary in view of the improvements that had been made in nuclear fuel, and requested that licensees submit a request to amend the Technical Specifications for their plant to delete these requirements. This amendment would delete the previous reporting and plant shutdown requirements based on iodine activity levels from the Hatch Units 1 and 2 Technical Specifications in accordance with the guidance of Generic Letter 85-19.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The proposed changes would make modifications of an administrative nature to the Technical Specifications to incorporate current reporting requirements of 10 CFR Part 50, Sections 50.72 and 50.73, and to revise the short-term reporting requirements regarding iodine activity levels. Neither of these changes would have any effect on the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident, or involve a significant reduction in the margin of safety.

The change also would delete the requirement for plant shutdown in the event the accumulated time with high iodine activty levels in the coolant exceeded 800 hours in a 12-month period. 10 CFR 50.72 (b)(1)(ii) requires that the NRC be immediately notified of fuel cladding failures that exceed expected values or that are caused by unexpected factors. This reporting requirement and proper fuel management, coupled with the fuel improvements that have been made, assure that corrective action would be taken long before the 800 hour limit is even approached, thereby making the shutdown requirement unnecessary. Deletion of this requirement would not involve a significant increase in the probability or consequences of an accident previously evaluated, would not create the possibility of a new or different kind of accident, and would not involve a significant reduction in the margin of safety.

On the basis of the above, the Commission has determined that the requested amendments meet the three criteria and therefore has made a proposed determination that the

amendment application does not involve a significant hazards consideration.

*Local Public Document Room location:* Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

*Attorney for licensee:* Bruce W. Churchill, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037

*NRC Project Director:* B.J. Youngblood

**Indiana and Michigan Electric Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan**

*Date of amendments request:* May 28, 1987.

*Description of amendments request:* The proposed amendment would revise the Technical Specifications for the spent fuel pool ventilation system to update the testing standards; modify the test requirement to reflect the as-built ventilation system; allow the crane bay and drumming room rollup doors to be open but under administrative control when fuel is moved or loads carried over the fuel pool; delete air flow distribution tests across HEPA filters and charcoal absorbers; delete redundant filter bypass testing requirements; modify leak testing requirements for charcoal and HEPA filters after modification, reinstallation, and sample testing; add a footnote to the spent fuel pool ventilation section to correspond to the auxiliary building crane section which recognizes the load block in a deenergized state as not being a heavy load; and correct a number of editorial errors.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The licensee proposes to update the testing standards for the spent fuel pool ventilation system. The current standard, ANSI N500-1975, will be replaced by ANSI N500-1980. The ANSI N510-1975 included test requirements for systems built to the ANSI N509 design; however, Cook was not designed or required to be designed to ANSI N509 and some provisions of ANSI N510-1975 cannot be literally complied with. ANSI N510-1980 recognizes that plants may be non-ANSI N509 designs and provides for this in stating requirements. Since ANSI N510-1980 corresponds more closely with the Cook ventilation system design and the change is to currently accepted testing standards, the change does not involve a significant increase in the probabilities or consequences of a previously analyzed accident. Since the change is in testing procedures and does not change operations or involve plant modifications, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed. The ANSI N510-1980 is the current industry standard and its use does not result in a significant reduction in a margin of safety.

The licensee proposes to modify the technical specifications to allow the crane bay rollup door and the drumming room rollup door, which are near the spent fuel pool, to be open under administrative control when fuel is being moved or when heavy loads are carried over the spent fuel pool. The current technical specifications require the ventilation system to maintain a negative pressure on the pool when fuel is moved or loads are carried over the fuel. With the doors open, the vent system cannot maintain the negative pressure; however, under administrative control the doors could be quickly closed in the event of any mishap. With the fans operating and either door open, the likely direction for air flow would be into the spent fuel pool area and through the filters. In the event the fans failed, fuel movement and loads over the spent fuel pool would be halted and the doors could be closed. In the event of an accident in the spent fuel pool, the fans are stopped and the doors would be closed to limit any releases.

The design basis accident is the complete rupture underwater of the highest rated spent fuel element. This accident was assumed to occur both inside containment and in the spent fuel pool. The bounding accident location is the one inside containment which took no credit for filtration or containment isolation. The proposed change by the licensee for the spent fuel pool area would change the results for the spent fuel pool but would still not be as severe since the doors would be quickly closed in the spent fuel pool area and the containment location for the accident was analyzed unisolated. The bounding accident location would remain as the one inside containment. Therefore, allowing the crane bay rollup and the drumming room rollup doors to be open under administrative control during movement of fuel or crane loads over the fuel could have some increase in the consequences of a previously evaluated accident location but are within the limits of the accident previously analyzed for the containment location. The proposed change by the licensee does not significantly increase the probability or consequences of the previously analyzed accident and because the change does not affect the mechanisms for damaging fuel and is consistent with ongoing operations with the doors, the change does not create a new or different kind of accident. Further, the administrative controls to quickly close the doors might represent some reduction in a margin of safety, but the brief time a door was open would not be a significant reduction in a margin of safety when the accident assumptions did not include any isolation of containment for the design basis.

The licensee also proposed to change the surveillance requirement for the spent fuel pool ventilation system to more accurately describe the system as designed and found acceptable. The original specifications were believed to be generically applicable to cover most systems but literal interpretation for the Cook system is questionable. The Cook proposal is to change the wording to more accurately describe the required test for the as-built system. The test will continue to be performed as always; therefore, there is no significant increase in the probabilities or consequences of any previously analyzed accident nor is there any reduction in a safety margin. There is no change in operation or testing; therefore, the proposed change does not create the possibility of any new or different kind of accident.

The licensee proposes to delete the air flow distribution testing across the HEPA and charcoal filters every 18 months but will continue to perform these tests as required by ANSI N510-1980 "following original installation, modification, or repair of the air cleaning system." The current Standard Technical Specifications do not require the testing in recognition that the code requirements are adequate and sufficient. The proposed change will not adversely affect the flow distribution nor will it greatly decrease the knowledge or expectations of the filter operation; therefore, there is no significant increase in the probabilities or consequences of any previously analyzed accident nor is there a significant decrease in a margin of safety. Deletion of the 18-month test will

not create any new or different kind of accident.

The licensee proposes to delete redundant surveillance requirements to determine total filter bypass, including leakage through the system's diverting valves. The specification as written does not recognize that the Cook ventilation system has bypass dampers integral to the system and that a combination of other required tests will also allow for the determination of total bypass and leakage. The licensee also proposes to change "diverting" to "directing" in the specifications to accurately reflect system function. Because the deletion is to remove redundant tests and the description change to "directing" is administrative in nature, these proposed changes do not significantly increase the probabilities or consequences of any previously analyzed accident nor do they involve any significant decrease in a margin of safety. These changes do not involve modifications or changes in system operations and, therefore, do not create any new or different kinds of accidents.

The current specifications require leak testing of charcoal adsorber banks after reinstallation but not after taking a sample. The licensee proposes to add a specification for leak testing charcoal adsorbers after sample taking to make the specifications consistent. The licensee also proposes to remove the requirement to test HEPA filters after charcoal tray installation since this would not affect the HEPA filters. These proposed changes are to achieve consistency and delete unnecessary tests and, therefore, do not significantly increase the probability or consequences of any previously analyzed accident nor do they significantly decrease any margin of safety. These actions will not involve changes in plant operation or configuration and, therefore, do not create any new or different kind of accident.

License Amendments 93/79, issued February 24, 1986, added footnotes to some technical specifications to allow operation of the crane over the spent fuel pool with the load block deenergized and unloaded. The spent fuel pool ventilation technical specifications should have been similarly modified with the same footnote. The licensee proposes to correct this oversight. This change is consistent with the previous footnotes and safety evaluation and thus does not significantly increase the probabilities or consequences of any previously analyzed accident nor does it significantly decrease any margin of safety. This mode of operation has been found acceptable pending a load drop analysis review and approval and does not create any new or different kind of accident.

In the last change, the licensee proposes to correct a number of typographical errors and make editorial changes that are administrative in nature. The Commission has also provided guidance concerning the application of these standards by providing examples of amendments considered not likely to involve significant hazards considerations (51 FR 7744). One of these examples, (i), is a purely administrative change to technical specifications. This last change by the licensee is directly related to this example.

On the basis of the above considerations, the staff proposes to determine that the licensee's request involves no significant hazards considerations.

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

*Attorney for licensee:* Gerald Charnoff, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

*NRC Project Director:* David L. Wigginton, Acting.

**Mississippi Power & Light Company, System Energy Resources, Inc., South Mississippi Electric Power Association, Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi**

*Date of amendment request:* June 3, 1987

*Description of amendment request:* The June 3, 1987 application for license amendment requested changes to Technical Specifications (TSs) related to the definition of core alteration and the snubber sample size. This notice considers only the changes related to the definition of core alteration.

The following changes to the Technical Specifications (TSs) would be made:

1.The definition of core alteration would be modified to exclude normal movement of the source range monitors (SRMs), intermediate range monitors (IRMs), local power range monitors (LPRMs), traversing in-core probes (TIPs) or special movable detectors.

2.The "*" footnote to Specification 3.1.1 on shutdown margin would be deleted. This footnote provides an exception to the core alteration definition for movement of IRMs, SRMs or special movable detectors.

3.The "*" footnote to Surveillance Requirement 4.1.3.2.a would be modified by deleting the exception to the core alteration definition for the movement of SRMs, IRMs or special movable detectors. The exception for normal control rod movement remains and is not affected by this proposed change.

4.The "*" footnote to Table 3.3.1-1 would be modified by deleting the exceptions to the core alteration definition for IRMs, SRMs or special movable detectors. The part of the "*" footnote requiring operable SRM instrumentation for replacement of LPRM strings would be retained.

5.The "**" footnote to Specification 3.9.2 on refueling operations instrumentation would be deleted. This footnote provides an exception to the core alteration definition for movement of IRMs, SRMs or special movable detectors.

6.The "*" footnote to Specification 3.9.5 would be modified by deleting the exception to the core alteration definition for incore instrumentation. The part of the "*" footnote that allows an exception for control rod movement with their normal drive system remains and is not affected by this proposed change.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards considerations if operation of the facility in accordance with a 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 licensee has provided an analysis of significant hazards consideration in its request for a license amendment. The licensee has concluded, with appropriate bases, that the proposed amendment meets the three standards in 10 CFR 50.92 and, therefore, involves no significant hazards considerations.

The licensee's analysis is reproduced below.

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The reactivity worth of the SRMs, IRMs, LPRMs, TIPs and special movable detectors is insignificant and their movement in the core does not have any adverse impact on reactivity excursion events. This change does not involve a change to plant hardware.

plant operating procedures or plant emergency procedures. Therefore, this change cannot increase the probability or consequences of an accident.

2. The proposed change does not create the possibility of a new or different kind of accident from any previously analyzed. Movement of all detectors in and around the core is controlled by approved procedures. The movement of the small amount of reactivity in the reactor core provided by the subject detectors does not create the possibility of a new accident or different accident from any previously analyzed.

3. The proposed change does not involve a significant reduction in a margin of safety. The proposed change to the core alteration definition for SRMs, IRMs, LPRMs, and special movable detectors will provide exceptions that are currently allowed in the Technical Specifications. The proposal to exempt the TIPs from the requirements of the core alteration definition is similar to that currently in effect for the other detectors. Therefore, since the proposed changes are similar to present exceptions allowed in the Technical Specifications, no margin of safety is reduced.

The NRC staff has made a preliminary review of the licensee's analysis and agrees with the licensee's conclusions that the three standards in 10 CFR 50.92 are met for the proposed changes in TSs for Grand Gulf Nuclear Station, Unit 1.

Accordingly, the Commission proposes to determine that the requested changes to the TSs do not involve significant hazards considerations.

*Local Public Document Room location:* Hinds Junior College, McLendon Library, Raymond, Mississippi 39154

*Attorney for licensee:* Nicholas S. Reynolds, Esquire, Bishop, Liberman, Cook, Purcell and Reynolds, 1200 17th Street, NW., Washington, DC 20036

*NRC Project Director:* Lester S. Rubenstein

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

*Date of application for amendment:* May 15, 1987

*Brief description of amendment:* The proposed amendment would delete the list of hydraulic and mechanical snubbers in Tables 3.6.1.a and 3.6.1.b, reduce the number of additional snubbers to be tested for every failed snubber discovered during functional testing, add a surveillance requirement for the snubber service life program, and clarify the visual inspection acceptance criteria and functional test requirements.

*Basis for proposed no significant hazards consideration determination:* Changes to the snubber population resulted from work accomplished during the 1985 refueling outage. Subsequently, additional snubbers were placed on the safety-related snubber lists in the station procedures. Generic Letter 84-13 concluded that the inclusion of snubber listings within the Technical Specifications is not necessary provided the snubbers are required to be operable. Therefore, the licensee has proposed, consistent with the BWR Standard Technical Specifications, to delete Tables 3.6.1.a and 3.6.1.b, Technical Specifications 3.6.1.1 and 3.6.1.6; to amend Technical Specifications 3.6.1.2, 3.6.1.3, 3.6.1.4, 3.6.1.5, 4.6.1.2, 4.6.1.3, and 4.6.1.4; and to add Technical Specification 4.6.1.5.

(1)The snubber listings currently in the station procedures would be maintained and updated as appropriate, and would be subject to the more stringent surveillance requirements as outlined in the proposed technical specifications

(2)The required number of additional snubbers to be tested for every failed snubber discovered during the functional testing would be reduced from 10% to 5% of the total number of snubbers of that type.

This change is consistent with the conclusions of the ASME-OM-4 evaluation which asserted that a retest of 5.0% of the original sample (i.e., 5% sample size of that type of snubber) provided adequate conservatism associated with snubber surveillance testing. The staff has previously concurred with these conclusions.

(3)A surveillance requirement for the snubber service life program would be added.

The intent of this new requirement is to ensure that the service life of the snubber is not exceeded between surveillance inspections. This represents a more stringent surveillance requirement.

(4)Clarification of the visual inspection acceptance criteria and functional retest requirements would be added.

These clarifying statements serve to better define snubber operability and to assure the retesting of the current snubbers after failure of the functional test in prior outages.

The licensee has reviewed the proposed changes pursuant to 10 CFR 50.59 and has determined that they do not constitute an unreviewed safety question. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety (i.e., safety-related) previously evaluated in the Final Safety Analysis Report have not been increased. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. There has not been a reduction in the margin of safety as defined in the basis for any Technical Specification. These proposed changes will not result in physical changes to the plant or changes in the way the plant is operated: Hence, there is no effect on the design basis accident analyses.

The licensee has also reviewed the proposed changes in accordance with 10 CFR 50.92 and has concluded that they do not involve a significant hazards consideration in that these changes would not:

1.Involve a significant increase in the probability or consequences of an accident previously analyzed. There are no physical changes to the plant as a result of the proposed changes; therefore, previously analyzed accidents are not affected.

2.Create the possibility of a new or different kind of accident from any previously analyzed. Since there are no changes in the way the plant is operated, the potential for an unanalyzed accident is not created because no new failure modes are introduced.

3.Involve a significant reduction in a margin of safety. The proposed changes to the Technical Specifications will not significantly reduce the surveillance testing requirements.

The proposed changes have no effect on the intent of the original Technical Specifications. No changes in the intent of current surveillance testing will result. The changes will more easily accommodate future additions or deletions to the safety-related snubber listings because a formal license amendment will not be required. However, any changes in snubber quantities, types, or locations would be a change to the facility, and such changes would continue to be evaluated against the provisions of 10 CFR 50.59, and would be reflected in plant records as appropriate.

*Local Public Document Room location:* Waterford Public Library, 49 Rope Ferry Road, Waterford, Connecticut 06385.

*Attorney for licensee:* Gerald Garfield, Esquire, Day, Berry, & Howard, Counselors at Law, City Place, Hartford, Connecticut 06103-3499.

*NRC Project Director:* Cecil O. Thomas.

**Northeast Nuclear Energy Company, et al., Docket No. 50-423, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut**

*Date of amendment request:* Partial response to the application dated June 25, 1987

*Description of amendment request:* By application for license amendment dated June 25, 1987, Northeast Nuclear Energy Company, et al. (the licensee), requested changes to the Technical Specifications (TS) for Millstone Unit 2 regarding hydraulic seismic restraints (snubbers), TS 3/4.7.8.1, as follows: (1) the TS Table that explicitly lists the snubbers that are required to be operable and undergo surveillance would be eliminated and (2) the TS numbering system for TS 3/4.7.8.1 would be changed. The third proposed change to the snubber TS, which would allow the licensee to perform an engineering evaluation to determine a snubber supported system/component to be operable with an inoperable snubber, will be addressed in future correspondence.

*Basis for proposed no significant hazards consideration determination:* On May 3, 1984 the NRC issued Generic Letter (GL) 84-13, "Technical Specifications for Snubbers". The contents of GL 84-13 state, in part:

> During the last several years, a large number of license amendments have been required to add, delete or modify the snubber listing within the technical specifications. We have reassessed the inclusion of snubber listings within the technical specifications and conclude that such listings are not necessary provided the snubber technical specification is modified to specify which snubbers are required to be operable. You should also note that the recordkeeping requirements of paragraph 4.9.7.f of the snubber technical specification are not altered by this revision. Paragraph 4.9.7.f requires that the plant records contain a record of the service life, installation date, etc. of each snubber. Since any changes in snubber quantities, types, or locations would be a change to the facility, such changes would be subject to the provisions of 10 CFR Part 50.59 and, of course, these changes would have to be reflected in the records required by paragraph 4.7.9.f.

The licensee's June 25, 1987 application for license amendment is responsive to GL 84-13 in that it proposes the deletion of the TS snubber listing. As recommended by GL 84-13, a revised Limiting Condition for Operation, (LCO) and associated Action Statement, replaces the snubber list. The proposed LCO and Action Statement define which snubbers must be operable (and undergo surveillance) and also provides appropriate remedial actions.

The proposed change to TS 3/4.7.8.1 does not involve a significant increase in the probability or consequences of an accident previously evaluated. Since no change in the number, location or operability of snubbers will result from the proposed TS change, system/components supported by the snubbers can be expected to respond within evaluated limits under normal (thermal expansion) and abnormal (seismic) conditions. The proposed change to the TS will not create the possibility of a new or different type of accident. The function of a snubber is to move slowly to allow for thermal expansion of the supported component and to "lock-up" to restrain the component under seismic conditions. No new snubber function is created by the proposed TS change and thus no new or different types of accidents are created. Finally, the proposed change to the TS would not involve a significant reduction in a safety margin. No changes to the seismic capability of systems or components is involved and thus no decrease in the seismic safety margins will occur. Accordingly, the Commission proposes to determine that the above-described changes to TS 3/4.7.8.1 involve no significant hazards considerations.

The licensee has also proposed renumbering existing TS 3/4.7.8.1 as 3/4.7.8. This change has been proposed since there is no TS 3/4.7.8.2.

On March 6, 1986, the NRC published guidance in the Federal Register (51 FR 7751) concerning examples of amendments that are not likely to involve a significant hazards consideration. One example provided in 51 FR 7751 of amendments not likely to involve significant hazards considerations is example (i) which involves "A purely administrative change to technical specifications: for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature". The proposed renumbering of the subject TS is within the example (i) noted above. Accordingly, the Commission proposes to determine that the proposed change to the TS involves no significant hazards considerations.

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*NRC Project Director:* John F. Stolz

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

*Date of amendment request:* October 3, 1986.

*Description of amendment request:* The proposed change would revise the wording of Technical Specifications 3.9.A and 3.9.B to reflect the existence of a third source of offsite power for supplying auxiliary electrical power. The Bases for Section 3.9 would also be revised to reflect the additional source.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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.

During the 1986 refueling and maintenance outage, a major improvement was made with the addition of a third source of offsite power for the safeguards buses at Monticello. A new source of auxiliary power, independently regulated and fed from the 345KV substation, was installed. No. 11 auxiliary transformer, which was fed from the generator output bus, was removed.

The proposed amendment would revise the Technical Specifications to recognize an additional source of offsite power available to supply the safeguards buses.

The licensee has evaluated the proposed changes to the Technical Specifications and determined they do not constitute a significant hazards consideration for the following reasons:

The additional source of offsite power significantly improves the reliability and flexibility of the station auxiliary power system. The probability of a loss of voltage or degraded voltage to the station safeguards buses is significantly reduced. The proposed amendment will require the availability of at least two offsite sources prior to critical operation just as the existing Technical Specifications do. Therefore, this change has no effect on the probability of loss of offsite power or the reliability of the power supply to plant safeguards

equipment and there is no increase in the probability or consequences of an analyzed accident.

The proposed Technical Specification changes deal exclusively with the offsite auxiliary electrical power supplies for the Monticello plant. They reflect the addition of an additional full capacity offsite source of auxiliary power to the plant. The plant is designed to respond to design basis accidents assuming the coincident loss of offsite power. The analyses reported in Section 14 of the Safety Analysis Report make this assumption. The intent of the additional offsite source is to improve the reliability of offsite power. Assuming, however, that this modification and the requested Technical Specification change could in some way result in a reduction in this reliability, no new or different kind of accident from any accident previously evaluated would be created since loss of offsite power is assumed.

The proposed Technical Specification wording change will recognize the addition to the plant of another full capacity source of offsite auxiliary electrical power. This source will significantly improve the reliability and stability of the power source to the safeguards buses. The proposed change will not, therefore, involve a reduction in the margin of safety. The staff has reviewed the licensee's analysis and agrees with the conclusions.

On the basis of the above, the staff proposes to determine that the proposed technical specification changes do not involve a significant hazards consideration.

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*NRC Project Director:* David L. Wigginton, Acting.

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

*Date of amendment request:* April 9, 1987 and June 15, 1987.

*Description of amendment request:* The proposed amendment to the technical specifications would permit the interval between the first and second containment integrated leak rate tests (ILRT's) in the second 10-year service period to exceed by a few months the 40 + 10 month interval between tests specified in the Technical Specifications.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists in 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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 proposed Technical Specification changes deal exclusively with a one-time relief of approximately 3 months to perform the second overall integrated containment leakage rate tests required during the second 10-year service period. This test can be performed only during refueling outages. Due to the current outage schedule, the requirement for test intervals of 40 + 10 months and the requirement that the last test of a series be conducted during the 10-year ASME Code Section XI inservice inspection outage cannot both be met. The proposed minor increase in specified test interval to allow a test in 1989 would resolve this conflict. The staff has reviewed and agrees with the licensee's evaluation of the requested changes.

On this basis, the Commission proposes to determine that the proposed actions involve no significant hazards consideration.

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*NRC Project Director:* David L. Wigginton, Acting.

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

*Date of amendment request:* December 9, 1986

*Description of amendment request:* The proposed amendment would revise the Susquehanna Steam Electric Station, Unit 1 Technical Specifications to support modifications which improve the containment isolation function and testability of the feedwater system.

Specifically, the proposed changes consist of:

(1) a change to Table 3.6.3-1 (Primary Containment Isolation Valves) replacing two reactor water cleanup (RWCU) manual isolation valves with two new gate valves. The valves being replaced (HV-144F042 and HV-144F104) are not being removed from the plant, but will not serve as containment isolation valves.

(2) two new containment isolation valves, HV-14182A&B, are being added to the list of valves in Table 3.8.4.2-1 (Motor-Operated Valves Thermal Overload Protection) because they are equipped with thermal overload bypass circuitry.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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 staff has reviewed the licensee's request and concurs with the following basis and conclusions provided by the licensee in its December 9, 1986 submittal.

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

Final Safety Analyses Report (FSAR) Section 5.4.8.2 states that the RWCU System is classified as a primary Power Generation System and is not an Engineered Safety Feature. The FSAR describes the function of the HV-1F042 and HV-1F104 system return valves as long term leakage control. Instantaneous reverse flow isolation is provided by the G33-1F039A&B check valves, further downstream in the RWCU piping. The modification will reassign the long term leakage control function from valves HV-F1F042 and HV-1F0104 to the new valves HV-14182A&B. The location of the new valves will be downstream from the G33-1F039A&B check valves and will not alter their present function of instantaneous reverse flow isolation. The motor-operated HV-14182A&B isolation valves will function as positiveclosing containment isolation valves for the RWCU branch connections to Feedwater penetrations X-9A and X-9B and will not increase the probability of an accident or malfunction of equipment related to safety.

(II) The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. FSAR Section 5.4.6.2 describes the safety-related portions of the RWCU System. This modification will improve RWCU capability to serve these safety-related functions by reducing containment valve leakage via new containment isolation valves HV-14182A&B. FSAR Section 6.2.4.3.2.1 identifies the safety-related function of the Feedwater containment isolation valves. This modification will not alter the present function of the Feedwater valves and is not likely to create a possibility for an accident or malfunction of a different type than is already evaluated in the FSAR.

(III) The proposed changes do not involve a significant reduction in a margin of safety.

As noted above, the containment isolation for the affected feedwater penetrations will be improved by the addition of the new valves, because they will not be used for throttling purposes. Therefore the margin of safety defined by the containment isolation function will not be reduced.

Based on the above considerations, the Commission proposes to determine that the proposed changes do not involve a significant hazards consideration.

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*NRC Project Director:* Walter R. Butler

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

*Date of amendment request:* June 19, 1987

*Description of amendment request:* The proposed amendment would revise the Susquehanna Steam Electric Station, Unit 1 Technical Specifications Table 3.8.4.1-1 to include four new circuit breakers that will be installed during the forthcoming refueling and inspection outage of Unit 1. The Technical Specifications Table 3.8.4.1-1 relates to primary containment penetration conductor overcurrent protective devices which ensure that primary and backup overcurrent protection circuit breakers are demonstrated operable by the performance of periodic surveillance. The modification to incorporate new circuit breakers will enhance the overcurrent protection of the penetrations. The new breakers being installed are thermal-magnetic and will trip prior to load device failure, thereby assuring protection of the primary containment penetration seals.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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 staff has reviewed the licensee's request and concurs with the following basis and conclusions provided by the licensee in its June 19, 1987 submittal.

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

The addition of redundant overcurrent protection for the subject penetrations will increase the reliability of the detection and isolation of short circuits. Therefore, the probability of a failure of the penetration seal being a contributor to an accident scenario is decreased.

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

The addition of redundant protection enhances its reliability. The new circuit breakers are of the same high quality as the existing ones, i.e. molded case and thermal magnetic. Failure of this redundant protection to detect and isolate a short circuit current will not create a malfunction different than any one that could be postulated without it installed.

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

As stated in I. above, the detection and isolation of short circuit currents is enhanced by this modification. This action therefore increases the safety margin associated with penetration protection.

Based on the above considerations, the Commission proposes to determine that the proposed changes involve no significant hazards consideration.

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

*Date of amendment request:* June 19, 1987

*Description of amendment request:* The proposed amendment would revise the Susquehanna Steam Electric Station, Unit 1 Technical Specifications in support of the forthcoming fuel reload for Cycle 4 operation. Specifically, the licensee has requested to change the following parts of the Technical Specifications:

(1) Definitions 1.2 and 1.13, related to fuel exposure and fraction of limiting power density

(2) Specification 3/4.2.1, related to Average Planar Linear Heat Generation Rate

(3) Specification 3/4.2.2, related to Average Power Range Monitor (APRM) Setpoints

(4) Specification 3/4.2.3, related to Minimum Critical Power Ratio (MCPR)

(5) Specification 3/4.2.4, related to Linear Heat Generation Rate (LHGR)

(6) Specification 3/4.4.1.1-2, related to Recirculation Loops - Single Loop Operation

(7) Specification 5.3.1, related to Fuel Assemblies

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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 staff has reviewed the licensee's request and concurs with the following basis and conclusions provided by the licensee in its June 19, 1987 submittal.

The following three questions are addressed for each of the proposed Technical Specification changes:

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

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

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

*Definition 1.2, Average Exposure*

I. No. Currently, no definition number is provided for Average Bundle Exposure. To correct this, the definitions of Average Bundle Exposure and Average Planar Exposure are proposed to be grouped under "Average Exposure". This relabeling is administrative in nature since neither existing definition is changing.

II. No. See I. above.

III. No. See I. above.

*Definition 1.13, Fraction of Limiting Power Density*

I. No. This change is administrative in nature; the definition was altered to reflect the appropriate Linear Heat Generation Rate (LHGR) to be used in determining the Fraction of Limiting Power Density, since and LHGR curve specifically for determination of APRM setpoints has been provided in this analysis. This is justified under Specification 3/4.2.2, APRM setpoints.

II. No. See I. above.

III. No. See I. above.

*Specification 3/4.2.1, Average Planar Linear Heat Generation Rate*

The changes to this specification reflect the addition of appropriate limits for Cycle 4 ANF 9x9 (XN-3) fuel, and editorial changes to replace references to "Exxon" with "ANF".

I. No. The editorial changes to correct the vendor reference are wholly administrative and have no impact on any safety analysis. The new Figure 3.2.1-3 illustrates the MAPLHGR limits for XN-3 fuel. These limits are based on an SNF analysis of the Loss of Coolant Accident analysis, operation within the proposed MAPLHGR remains below 2200&F, local Zr-$H_2O$ reaction remains below 17%, and core-wide hydrogen production remains below 1% for the limiting LOCA as required by 10CFR50.

With respect to GE and ANF 8x8 fuel, the Reload Summary Report shows that the XN-3 fuel is hydraulically and neutronically compatible with both fuel types. Therefore the existing MAPLHGR limits provided in Figures 3.2.1-1 and 3.2.1-2 remain applicable for Unit 1 Cycle 4 operation.

II. No. Although there are obvious physical differences between the XN-3 fuel and the GE and ANF 8x8 fuel, there are no significant differences in their operating characteristics. Therefore, the addition of XN-3 fuel in the Cycle 4 core does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. No. The analyses performed were done in accordance with 10CFR50 Appendix K and did not predict a significant reduction in any safety margin. The methodology used to perform the Cycle 4 safety analyses contain similar inherent conservatism to those which supported the Cycle 3 core.

*Specification 3/4.2.2, APRM Setpoints*

This specification has been changed to explicitly define T for GE fuel and ANF fuel. Since for ANF fuel T is dependent on a transient-based LHGR, a new figure, 3.2.2-1, has been provided.

I. No. For GE fuel, the method of calculating T has not changed. Clarification was simply provided to ensure that T was properly determined. This part of the change is therefore editorial.

For ANF fuel, the T factor is modified by an exposure dependent LHGR which is based on ANF's Protection Against Fuel Failure (PAFF) line shown on Figure 3.4 of SN-NF-85-67, Revision 1. This LHGR is provided in new Figure 3.2.2-1, which corresponds to the ratio of PAFF/1.2. Under this limit, cladding and fuel integrity are protected during anticipated operational occurrences (AOO's), including an overpower condition for transients initiated from partial power. Therefore, this change will ensure fuel design limits are not violated.

II. No. Again, for GE fuel the change is editorial. For ANF fuel, as stated in (I) above, the new LHGR limit provides assurance that cladding and fuel integrity are protected during AOO's. Since the ANF fuel is hydraulically and neutronically compatible with GE fuel, no new events were postulated to occur.

III. No. For GE fuel, no change has occurred. For ANF fuel, the method for determining T has been shown to provide appropriate protection against 1% clad strain and fuel centerline melting; therefore, no significant reduction in safety margin has occurred.

*Specification 3/4.2.3, Minimum Critical Power Ratio*

I. The plant transient model used to evaluate the system effects of the Feedwater Controller Failure and Load Reject Without Bypass transients is ANF's CONTRANSA code... This output was utilized by the XCOBRA-T methodology... to determine Delta CPRs. The CONTRANSA code has been used in support of SSES Unit 2 Cycle 2, and has been approved by the NRC.

All core-wide transients were analyzed deterministically (i.e., using bounding values of input parameters).

Based on the above, the method used to develop operating limit MCPRs for the Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated.

II. No. The Methodology described can only be evaluated for its affect on the consequences of analyzed events; it cannot create new ones. The consequences of analyzed events were evaluated in I. above.

III. No. As stated in I. above, the methodology used to evaluate core-wide transients is consistent or more realistic than previously approved methods and meets all pertinent regulatory requirements for use in this application. Therefore, its use will not result in a significant decrease in any margin of safety.

*Specification 3/4.2.4, Linear Heat Generation Rate*

The changes to this specification reflect editorial changes to replace references to "ENC" with "ANF", to revise the limits for ANF 8x8 fuel, and to add appropriate limits for ANF 9x9 (XN-3) fuel.

I. No. The editorial change to correct the vendor reference are wholly administrative and have no impact on any safety analysis. Revised Figure 3.2.4.2-1 and new Figure 3.2.4.2-2 reflect appropriate LHGR limits for ANF fuel under steady-state conditions. These figures are based on information provided in the fuel mechanical design analysis (XN-NF-85-67, Rev. 1) and assures margin to design limits for the life of the fuel.

II. No. This change reflects appropriate limits which ensure compliance with all relevant fuel mechanical design criteria. Application of these limits will not create the possibility of a new or different accident.

III. No. the mechanical design report which forms the basis for these LHGR limits is the same analyses which was approved in support of Unit 2 cycle 2 operation. These analyses ensure appropriate safety margin to fuel mechanical design limits for all anticipated operational occurrences throughout the life of the fuel.

*Specification 3/4.4.1.1.2, Recirculation Loops-Single Loop Operation*

I. No. Three editorial changes are being proposed: a reference to Exxon is changed to ANF; a reference to the new Figure 3.2.1-3 is provided (this was justified under Specification 3/4.2.1); and a typographical error in the APRM - Flow Biased Trip Setpoint under 3.4.1.1.2.a.5.b is corrected. These changes are administrative and therefore do not impact any safety analysis (again, the new Figure 3.2.1-3 was considered previously).

II. No. See I. above.

III. No. See I. above.

*Specification 5.3.1, Fuel Assemblies*

I. No. As written, this specification provides GE and ANF 8x8 general core design information. The proposed changes provide the same information for the ANF 9x9 fuel being introduced in Cycle 4. This general information was part of a much more elaborate set of inputs used to generate the attached analyses and the Technical Specification limits discussed above. Since the Technical Specifications and associated analyses have been shown not to increase the probability or consequences of any previous evaluation, the proposed change to this section is primarily editorial and therefore will not degrade the current level of safety at Susquehanna SES Unit 1.

II. No. See I. above.

III. No. See I. above.

Based on the above considerations, the Commission proposes to determine that the proposed changes involve no significant hazards consideration.

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*NRC Project Director:* Walter R. Butler

**Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania**

*Date of amendment request:* April 13, 1987

*Description of amendment request:* The proposed amendments would revise the Susquehanna Steam Electric Station (SSES) Units 1 and 2 Technical Specifications to make the Surveillance requirements for the Standby Gas Treatment System (SGTS) exhaust high radiation isolation instrumentation consistent with its applicability requirements.

Table 4.3.2.1-1 currently requires valid surveillance prior to entry into Operational Conditions 4 or 5 regardless of whether venting or purging is planned. The proposed change would modify the applicability requirements for Operational Conditions 4 and 5 such that the SGTS exhaust high radiation instrumentation be operable only when venting or purging the drywell in accordance with the Technical Specification Section 3.11.2.8.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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 staff has reviewed the licensee's request and concurs with the following basis and conclusions provided by the licensee in its April 13, 1987 submittal.

The proposed changes do not:

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

The safety function of the subject instrumentation during Operational Conditions 4 and 5 is to isolate the containment purge and nitrogen makeup valves upon sensing a predetermined radiation level at the SGTS exhaust vent during venting or purging operations. The subject change will ensure that the instrumentation will have valid surveillances on record prior to performing venting or purging, thereby ensuring a high reliability of the isolation function. It is unnecessary to require surveillance requirements to be met when the instrumentation is not required to be operable. Since the proposal is not affecting the reliability of the safety function but is simply deleting an unnecessary restriction, no previous accident evaluation is impacted.

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

Since no changes to the design or operation of the subject instrumentation is proposed, no new event requiring evaluation is required.

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

As explained in (1) above, the period of operation when the safety function of the instrumentation is required is unaffected by the proposed change; therefore safety margin is not impacted.

Based on the above consideration, the Commission proposes to determine that the proposed changes do not involve a significant hazards consideration.

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**Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania**

*Date of amendment request:* April 13, 1987

*Description of amendment request:* The proposed amendments would revise the Susquehanna Steam Electric Station (SSES) Units 1 and 2 Technical Specifications Section 3.6.5.3 related to the Standby Gas Treatment System (SGTS). Technical Specification 3.6.5.3 presently requires that upon losing both trains of SGTS, the Technical Specifications Section 3.0.3 be invoked and within one hour the licensee is to proceed to plant shutdown. The proposed change to Section 3.6.5.3 would allow the licensee 4 hours to restore at least one inoperable SGTS subsystem to operable status.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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 staff has reviewed the licensee's request and concurs with the following basis and conclusions provided by the licensee in its April 13, 1987 submittal.

The proposed change does not:

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

The proposed change affects the allowed restoration time of the SGTS, and this can impact the existing accident analyses only if the time is increased to the point where the system cannot be assumed to perform its safety function when required. This is not the case, here, since the proposed increase (from 1 to 4 hours) is consistent with the currently allowed restoration time for the safety function that SGTS is supporting.

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

Neither the design nor the operation of the SGTS have been affected by the proposed change; therefore, no new events are required to be evaluated.

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

This change, in terms of impact on safety margin has been qualitatively assessed as an insignificant degradation if it is one at all. This degradation is representative of the "extra" safety margin provided by shutting the unit down sooner when SGTS is the reason Secondary Containment Integrity has been violated. This is insignificant for two main reasons:

(a) It has been shown that Secondary Containment Integrity can be violated for up to 4 hours and that this should be independent of the reason for the violation, and

(b) Some positive safety margin is incurred through fewer shutdowns due to SGTS inoperability, coupled with the fact that less restoration work will have to be performed while the unit is proceeding to shutdown and is therefore in a transient condition.

Based on the above considerations, the Commission proposes to determine that the proposed change does not involve a significant hazards consideration.

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**Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania**

*Date of amendment request:* June 10, 1987

*Description of amendment request:* The proposed amendments would revise the Susquehanna Steam Electric Station

Units 1 and 2 Technical Specifications related to load profiles for batteries 1D612 and 1D622. The proposed changes are intended to accommodate the installation of Alternate Rod Injection solenoid valves in compliance with Anticipated Transients Without Scram (ATWS) rule.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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 staff has reviewed the licensee's request and concurs with the following basis and conclusions provided by the licensee in its June 10, 1987 submittal.

The proposed changes do not:

(1) Involve an increase in the probability or consequences of an accident previously evaluated. FSAR Section 8.3.2.1.1.4 states the station batteries have sufficient capacity without the charger, to independently supply the required loads for 4 hours. The Technical Specifications require the batteries be surveilled to a dummy load which is greater than the design loads. To demonstrate the batteries have adequate capacity to supply the proposed load profile, a calculation was performed by our Engineering department. This calculation - in part - is based on IEEE 485. In particular, cell sizing worksheets were prepared for the new loads. These worksheets determine the number of positive plates the specific battery must have to supply the given load profile. For the proposed profile, and considering worst case which is a battery electrolyte temperature of 60&F, the 1D612 and 1D622 batteries were demonstrated to have adequate capacity.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. As stated in Part (1), the batteries have sufficient capacity for the proposed load profile, thus enabling them to perform their intended function. Any postulated accident resulting from this change is bound by previous analysis.

(3) Involve a reduction in the margin of safety. In accordance with IEEE 450, the related battery capacity is 25 percent greater than required. This margin allows replacement of the battery when its capacity has decreased to 80 percent of its rated capacity (100 percent of design load). This margin has been maintained and included as part of the calculation referred to in Part (1).

Based on the above considerations, the Commission proposes to determine that the proposed changes involve no significant hazards consideration.

*Local Public Document Room location:* Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701

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

*NRC Project Director:* Walter R. Butler

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

*Date of amendment request:* June 30, 1987, supplementing previous amendment request dated May 22, 1987

*Description of amendment request:* This submittal augments the supporting information provided in the May 22, 1987 submittal which was noticed in the Federal Register on July 1, 1987 (52 FR 24557). This submital provides additional details concerning the supporting information related to the proposal to change the Technical Specification Steam Line Radiation High-High trip function setpoint in order to permit hydrogen injection testing. Specifically, this submittal states that the licensee will follow the separation distance for the hydrogen tube trailer that is specified in the draft January 1987 EPRI "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations." These EPRI guidelines specify a more conservative (greater) separation distance for the hydrogen tube trailer than specified by the code and committed to by the licensee in its May 22, 1987 submittal (NFPA Code No. 50A). The submittal also provides additional discussion of actions taken to minimize personnel exposure during the test. It further states that chlorine is not stored on site.

*Basis for proposed no significant hazards consideration determination:* This submittal does not change the amendment requested by the licensee in its May 22, 1987 submittal which the staff proposed to determine does not involve a significant hazards consideration (52 FR 24557). It makes a commitment to separation distance for the hydrogen tube trailer that is more conservative than proposed in the previously noticed submittal and notes that there is no chlorine stored on the site (another conservatism). Accordingly, the staff proposes to determine that the proposed amendment, as supplemented by this June 30, 1987 submittal, does not involve a significant hazards consideration.

*Local Public Document Room location:* Pennsville Public library, 190 S. Broadway, Pennsville, New Jersey 08070

*Attorney for licensee:* Troy B. Conner, Jr., Esquire, Conner and Wetterhahn, 1747 Pennsylvania Avenue, NW., Washington, DC 20006

*NRC Project Director:* Walter R. Butler

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

*Date of amendment request:* March 10, 1987

*Description of amendment request:* The proposed amendment consists of increasing the boron concentration limits in the Refueling Water Storage Tanks and Accumulators. These changes are being requested to accommodate the use of high energy, low leakage cores for future Unit 1 and 2 reloads. Also included are several changes to various related specifications which are being made to provide uniformity between Salem Unit 1 and Unit 2 Technical Specifications.

Specifically, the following changes are being requested:

(1) Increase of the required RWST boron concentration range from 2000-2200 ppm to 2300-2500 ppm - Units 1 and 2

(2) Increase of the required Accumulator boron concentration range from 1900-2200 ppm to 2200-2500 ppm - Units 1 and 2

(3) Increase Units 1 and 2 RWST Mode 5 and 6 required volume from 9,690 gallons to 12,500 gallons

(4) Increase Units 1 and 2 maximum expected boration capability requirement from 75,000 gallons to 85,000 gallons (Basis only)

(5) Change Unit 1 Mode 1 thru 3 accumulator volume range from 6380-6657 to 6223-6500

(6) Change Unit 1 Boric Acid Tank (BAT) boron concentration range from 20,100-21,800 to 20,000-22,500 ppm Boron

(7) Change Unit 1 Boron Injection Tank (BIT) Boron concentration range from 20,100-21,800 to 20,000-22,500 ppm Boron

(8) Change Unit RWST Mode 1 thru 4 required volume range from 364,000-400,000 to 364,500-400,000 gallons

(9) Changes on LCOs related to heat tracing requirements to make units consistent internally and with the Westinghouse Standard Technical Specifications.

*Basis for proposed no significant hazards consideration determination:* The licensee has determined that the proposed changes involve no significant

hazards consideration under the provisions of 10 CFR 50.92. A safety analysis was performed to assess the impact of increasing the boron concentration ranges for the RWST and Accumulators. Those incidents which were found to be sensitive to the above changes were evaluated and/or analyzed. These incidents include the following:

1. Non-LOCA FSAR transients
2. Small and Large Break LOCA
3. Hot Leg Switchover Time Calculation
4. Sump and Spray pH, Hydrogen Production, Stress Corrosion, and Radiological Consequences

The results of the safety analysis show that for every incident analyzed, the safety criteria continue to be fully met with ample margin to the applicable FSAR limits. Thus, the current safety analysis design bases continue to be met, and operations of Salem Units 1 and 2 in accordance with the proposed increase of RWST and Accumulator Boron Concentration ranges amendment:

(1) would not involve a significant increase in the probability or consequences of an accident previously evaluated for Salem Units 1 and 2, since the proposed increase in boron concentration ranges showed that the current core safety limits were still met.

(2) in no way creates the possibility of a new or different kind of accident from any accident previously evaluated for Salem Units 1 and 2, since no plant modifications resulted from this change.

(3) does not involve a significant reduction in the margin of safety since the Technical Specification Boron Concentration limits are being increased to reflect the requirements of the extended life core analysis, thus at least maintaining current safety margins.

Additionally, the increase in the RWST minimum contained volume from 9690 gallons to 12,500 gallons in modes 1 through 4 and the increase in maximum boration capability requirement from 75000 gallons to 85000 gallons resulted due to a revision in Westinghouse design policy. This revision requires that the limiting boron addition to achieve cold shutdown be 1300 ppm in taking the unit from the End of Life, Hot Full Power condition to cold shutdown. The new limit offers more core design and operational flexibility. These changes continue to meet the design bases in the current safety analysis. It does not involve a significant increase in the probability or consequences of an accident previously evaluated for Salem Units 1 and 2 as the current core safety limits continue to be met; it does not create a possibility of a new or different kind of accident since no plant modifications are made and does not involve a significant reduction in the margin of safety since the margin of safety is being maintained by the increased boron requirements.

Based on the above evaluation, we have determined that the proposed amendment, revising the RWST and Accumulator boron concentration ranges and the boration capability requirement technical specifications and bases do not constitute a significant hazards consideration. In addition, the Commission has provided guidance and examples of amendments that are considered not likely to involve significant hazards considerations. The proposed changes correspond to example (vi) as a change which in some way may result in some increase in the probability or consequences of a previously analyzed accident, but are within the acceptable criteria for the system.

The various other changes in heat tracing requirements, volumes and/or concentrations, and bases discussed above were administrative in nature and either corrected typographical errors or corrected inconsistencies between Units 1 and 2 Technical Specifications in accordance with existing analyses since the analyses are applicable to both units. These changes do not create a significant increase in the probability or consequence of a previously evaluated accident, create a new or different kind of accident, or involve a significant reduction in the margin of safety due to their administrative nature. These changes correspond to Example (i), (51 FR 7744), as a purely administrative change.

Therefore, on the basis of the licensee's analysis with which the staff agrees, and because the proposed changes fit the examples cited above, the Commission proposes that the amendment will not involve a significant hazards consideration.

*Local Public Document Room location:* Salem Free Public library, 112 West Broadway, Salem, New Jersey 08079

*Attorney for licensee:* Mark J. Wetterhahn, Esquire, Conner and Wetterhahn, Suite 1050, 1747 Pennsylvania Avenue, NW., Washington, DC 20006

*NRC Project Director:* Walter R. Butler

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

*Date of amendment requests:* February 27, 1987 (TS 87-01)

*Description of amendment requests:* Tennessee Valley Authority proposes to amend the technical specifications of Sequoyah Nuclear Plant Units 1 and 2 to ensure that directions given by the technical specifications regarding submittals to the NRC are consistent with those determined in the 10 CFR Parts 50 and 51 Final Rule as published in the Federal Register on November 6, 1986, and made effective January 5, 1987.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards determination exists as stated in 10 CFR 50.92(c). 10 CFR 50.91 requires that at the time a licensee requests an amendment, it must provide to the Commission its analyses, using the standards in Section 50.92 about the issue of no significant hazards consideration. Therefore, in accordance with 10 CFR 50.91 and 10 CFR 50.92, the licensee has performed and provided the following analysis.

Because of varying and sometimes conflicting requirements for the submittal of information by applicants and licensees, confusion has arisen with regard to copy requirements and proper submittal procedures. In an effort to clarify these matters, the NRC issued Regulatory Guide 10.1 (Revision 4), "Compilation of Reporting Requirements for Persons Subject to NRC Regulations," and on August 8, 1982, the Director, Division of Licensing, Office of Nuclear Reactor Regulation, issued Generic Letter 82-14, "Submittal of Documents to the Nuclear Regulatory Commission." While these efforts at clarification resolved much of the confusion, these guidance documents contain outdated information and, in many cases, conflict with requirements in regulations or individual licenses. Therefore, NRC has issued the 10 CFR Parts 50 and 51 Final Rule to specify copy requirements and provide mailing instructions for submittals to NRC.

Licensees whose technical specifications contain submittal directions that conflict with those determined in the 10 CFR Parts 50 and 51 Final Rule were authorized by the Final Rule to delete the conflicting directions by pen and ink changes to the technical specifications. The distribution of these pen and ink changes for all controlled copies of Sequoyah Nuclear Plant units 1 and 2 technical specifications has been initiated with the subject changes noted as a "Special Revision." For the purpose of complete documentation and to formally incorporate the pen and ink changes into the licenses of the respective units, these changes are also being submitted to NRC.

In its conclusion, the licensee addressed the issue of no significant hazards consideration as follows:

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

No. This amendment, already authorized by NRC in their determination of the 10 CFR Parts 50 and 51 Final Rule, deletes conflicting submittal directions contained in the technical specifications. In that this amendment affects only submittal directions, it follows that the amendment results in no changes in either the probability or consequences of an accident previously evaluated.

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

No. Again, as the amendment only deletes conflicting submittal directions contained in the technical specifications, the amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. Again, as the amendment only deletes conflicting submittal directions contained in the technical specifications, the amendment involves no reduction in margin of safety.

The staff has reviewed the licensee's no significant hazards consideration determination and agrees with the licensee's analysis. Therefore, the staff proposes to determine that the application for amendments involves no significant hazards consideration.

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

*Attorney for licensee:* General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, E11 B33, Knoxville, Tennessee 37902.

*NRC Assistant Director:* John A. Zwolinski

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

*Date of amendment requests:* April 16, 1987 (TSC 87-21)

*Description of amendment requests:* Tennessee Valley Authority (TVA) proposes an amendment to the Sequoyah Nuclear Plant Units 1 and 2 Technical Specifications to correct the line items alignment for functional units 20, reactor trip breakers, and 21, automatic trip logic, of Table 3.3-1, "Reactor Trip System Instrumentation." Additionally, the channels to trip, minimum channels operable, and applicable mode requirements for the same functional units would be revised to be consistent with the Standard Technical Specifications (NUREG-0452).

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards determination exists as stated in 10 CFR 50.92(c). 10 CFR 50.91 requires at the time a licensee requests an amendment, it must provide to the Commission its analyses, using the standards in Section 50.92 about the issue of no significant hazards consideration. TVA has proposed amendments which: (1) do not involve a significant increase in the probability or consequences of an accident previously evaluated because the correction to line item alignment is strictly an administrative change. The removal of the "*" mode for startup and power operation is also an administrative change because the change only involves removing a redundant entry. The proposed changes to correct number requirements for channels to trip and minimum channels operable must be made to reflect trip capabilities in the shutdown modes. The proposed amendments involve no modifications nor changes to the way in which plant operation is conducted. (2) Do not create the possibility of a new or different kind of accident from any previously evaluated because it is administrative in nature and does not create any new situation to those previously analyzed. The purpose of the line requirements for the reactor trip breakers and automatic trip logic has not been changed. And (3). do not affect the margin of safety. The proposed changes are administrative in nature and will not require any modifications or alter any testing, action, or operating requirement of the plant.

Therefore, the staff proposes to determine that the application for amendments involves no significant hazards consideration.

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

*Attorney for licensee:* General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, E11 B33, Knoxville, Tennessee 37902.

*NRC Assistant Director:* John A. Zwolinski

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

*Date of amendment requests:* April 17, 1987 (TS 87-10)

*Description of amendment requests:* The proposed amendments would increase the surveillance requirements (SR) in specifications 4.6.1.8.d.1, 4.7.7.e.1, 4.7.8.d.1, and 4.9.12.d.1 of both the Sequoyah Units 1 and 2 Technical Specifications. The amendments decrease the maximum allowable pressure drop across the high efficiency particulate air (HEPA) and charcoal filters in the emergency gas treatment system, auxiliary building gas treatment system, and the control room emergency ventilation system.

*Basis for proposed no significant hazards consideration determination:* The Tennessee Valley Authority's (TVA) Technical Specification Verification Program at Sequoyah identified no traceable documented basis for the limiting pressure drop values currently provided in SR 4.6.1.8.d.1, 4.7.7.e.1, 4.7.8.d.1, and 4.9.12.d.1. Consequently, TVA is revising these pressure drop values to reflect the values used in the engineering design calculations. These design calculations were performed to determine the fan sizing requirements based on the desired airflow rate and the pressure losses due to the duct configurations, dampers, grills, and filters. The proposed limiting pressure drop values are the pressure drops solely attributed to the HEPA filters and charcoal banks in the filter train of each system. This process would result in a traceable documented basis for each of the subject pressure drop values.

The proposed amendments decrease the maximum allowable pressure drop across the HEPA filters and charcoal banks in three ventilation filter assemblies. This would require the HEPA filters to be replaced more frequently than the current TS. This provides more assurance that these filters would be available during an accident.

The Commission has provided standards for determining whether a significant hazards determination exists as stated in 10 CFR 50.92(c). 10 CFR 50.91 requires that at the time a licensee requests an amendment, it must provide the Commission its analyses, using the standards in Section 50.92 about the issue of no significant hazards consideration. Therefore, in accordance with 10 CFR 50.91 and 10 CFR 50.92, the licensee has performed and provided the following analysis.

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

No. The proposed change is conservative. The maximum allowable pressure drops due to dirty HEPA and charcoal filters in the various air cleanup systems are lowered. This change adds additional assurance that these filters will be replaced before their function or the system's flow capacity is significantly degraded.

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

No. The proposed change will not result in any new maintenance procedures, testing, or

different operational parameters. Thus, the possibility of a new or different kind of accident is not created.

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

No. The margin of safety is not reduced. The function of these air cleanup systems is not compromised.

The staff has reviewed the licensee's no significant hazards consideration determination and agrees with the licensee's analysis. Therefore, the staff proposes to determine that the application for amendments involves no significant hazards consideration.

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

*Attorney for licensee:* General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, E11 B33, Knoxville, Tennessee 37902.

*NRC Assistant Director:* John A. Zwolinski

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

*Date of amendment requests:* May 22, 1987 (TS 87-11)

*Description of amendment requests:* Tennessee Valley Authority proposes to amend the Sequoyah Nuclear Plant Units 1 and 2 Technical Specifications to revise Table 4.11-2, "Radioactive Gaseous Waste Monitoring Sampling and Analysis Program." Specifically, footnote b of Table 4.11-2 would be modified to require sampling to be performed using upper and lower containment noble gas activity monitors rather than the Shield Building stack monitor as is currently called for in the subject footnote.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards determination exists as stated in 10 CFR 50.92(c). 10 CFR 50.91 requires at the time a licensee requests an amendment, it must provide to the Commission its analyses, using the standards in Section 50.92 about the issue of no significant hazards consideration. Therefore, in accordance with 10 CFR 50.91 and 10 CFR 50.92 the licensee has performed and provided the following analysis.

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

No. The proposed amendment would provide for a more accurate and timely evaluation of the need to perform a sampling and analysis of the containment atmosphere following shutdown, startup, or a thermal power change exceeding 15 percent of rated thermal power. The basis for the evaluation would be obtained using existing radiation monitoring instrumentation. The proposed amendment does not change plant hardware, plant operating setpoints or limits, or plant operating procedures. Thus, the proposed amendment involves no increase in the probability or consequences of an accident previously evaluated.

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

No. As previously stated, the proposed amendment does not change plant hardware, plant operating setpoints or limits, or plant operating procedures. The proposed amendment would only utilize existing instrumentation that will provide both a representative and timely evaluation of containment atmosphere activity. Thus, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. Again, as previously stated, the proposed amendment does not change plant hardware, plant operating setpoints or limits, or plant operating procedures. The proposed amendment would, however, result in the utilization of existing instrumentation that will provide both a representative and timely evaluation of containment atmosphere activity. Thus, the proposed amendment would involve no reduction in the margin of safety, but rather increases the margin of safety.

The staff has reviewed the licensee's no significant hazards consideration determination and agrees with the licensee's analysis. Therefore, the staff proposes to determine that the application for amendments involves no significant hazards consideration.

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

*Attorney for licensee:* General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, E11 B33, Knoxville, Tennessee 37902.

*NRC Assistant Director:* John A. Zwolinski

**Virginia Electric and Power Company, Docket Nos. 50-338 and 50-339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia**

*Date of amendment request:* May 27, 1987

*Description of amendment request:* The proposed change would modify the North Anna 1 and 2 (NA-1&2) Technical Specifications (TS) 3/4.6.3, Table 3.6-1 as follows:

1. Correct an inconsistency between the licensee's response to NUREG-0737 and the NA-2 TS with regard to the isolation signal for the containment instrument air system. At present, instrument air isolates on a Phase B signal in Unit 1 and a Phase A signal in Unit 2. The proposed change would change the containment isolation signal in NA-2 from Phase A to Phase B to be in compliance with NUREG-0737.

2. Place the hydrogen recombiner containment isolation valves for NA-1&2 under administrative control in order to permit functional testing in Modes 1 through 4.

A discussion of both proposed changes follows:

1. Containment isolation at NA-1&2 is accomplished in two phases. Containment isolation Phase A is initiated by the Safety Injection Actuation System. Containment isolation Phase B is initiated on the containment high-high pressure setpoint.

TS 3/4.6.3, Table 3.6-1 for NA-2 lists the containment isolation valves for containment instrument air supply (TV-IA 202A and TV-IA 202B) as shutting on receipt of a Phase A containment isolation signal. Although the licensee's letter dated October 24, 1979 designated NA-1 containment instrument air as an essential Level 2 system which isolates on a Phase B signal, the similar letter for NA-2 dated October 25, 1979 makes no mention of the containment instrument air system. The licensee's response to NUREG-0737, Rev. 0 dated December 10, 1980 lists containment instrument air as a Phase B isolation system for both units, indicating that the omission of mention in the NA-2 letter was an oversight. The licensee proposes to change the containment isolation signal from Phase A to Phase B to correct this oversight and bring NA-2 instrument air containment isolation signal into compliance with NUREG-0737 and provide consistency with the homologous valves in NA-1.

2. Technical Specification surveillance requirements as listed in TS 4.6.4.2.a for NA-1&2 require a functional test of the hydrogen recombiners every six months. TS 3.6.1.1, on the other hand, requires that containment integrity be maintained in Modes 1-4 and specifies that the valves listed in Table 3.6-1 which do not receive automatic closure signals be maintained closed unless otherwise noted in Table 3.6-1. At present, the valves not receiving automatic closure signals include the hydrogen recombiner remote-manual isolation valves (Unit 1 - TV-HC-104A and B, 105A and B, 106A and B, and 107A and B; Unit 2 - TV-HC-204A and B, 205A and B, 206A and B, and 207A and B). It is proposed that Table 3.6-1 be changed by adding asterisks to these

valves to provide for taking administrative control in order to permit surveillance testing at power. During these periods (approximately four hours every six months), the operation of the affected valves would be administratively controlled via a Periodic Test (PT) procedure.

In a letter from the NRC dated January 12, 1984, a safety evaluation was provided which accepted NA-1&2's action taken in response to NUREG-0737 Item II.E.4.1 (Dedicated Hydrogen Penetrations). The safety evaluation found the licensee's request to add the remotely operated (HC Series) valves to Table 3.6-1 acceptable and that "opening of these valves will take place only under specific administrative control as specified in post-accident procedures." During the testing period, the recombiner would be connected directly to the containment atmosphere and the recombiner and associated piping could possibly be pressurized to 45 psig if an accident were to occur. For normal operation, the recombiner is designed for an operating pressure limit of 10 psig.

A review of the hydrogen recombiner system design has been conducted to evaluate conducting the test at power. The hydrogen recombiners and their associated piping systems are seismically designed and missile protected. The piping and components are designed to ASME III, Class 2 through Summer 1973 Addendum. The recombiner system piping is 150 lb. carbon steel. The recombiner pressure-retaining boundary is 304 stainless steel, is designed to withstand 50 psig and was originally tested to 75 psig.

The recombiner technical manual specifies a 10 psig rating for operating conditions (1300-1400&F) and 50 psig for non-operating conditions. Rockwell International was contacted and stated that the 10 psig operating pressure limit is imposed to limit the mass flow through the recombiner to provide assurance that effluent hydrogen concentration meets specifications, and that there is no overstress concern with a pressure excursion to 45 psig at operating temperatures.

Further analysis and evaluation has concluded that the recombiner could withstand a DBA since sufficient design margin existed to preclude a breech of the hydrogen recombiner components and associated piping. This conclusion was based upon a review of the Rockwell Stress Report (TI-019-120-003) and application of design information through calculations to show that sufficient design margin existed for the postulated test configuration with a temperature of 1250&F and a pressure excursion to 50 psig.

The alignment of the recombiner to the containment during functional testing provides verification of design flow capability for the actual flow path used during accident conditions. Alternative testing schemes to keep the recombiner separated from containment atmosphere would require some degree of piping reconfiguration which could potentially affect the integrity of the piping system following restoration after testing. In addition, the capability exists to isolate the containment from the recombiner by taking operator action from the control room in the event containment isolation is required while surveillance testing is in progress. Prior to implementation of this TS change, which would allow functional testing at power, precautions will be added to the two affected PTs (68.1.1 and 68.1.2) to instruct the control room operator to secure the recombiner and shut any open HC valves if a containment isolation signal is generated while the test is in progress.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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 proposed changes do not involve a significant hazards consideration because operation of NA-1&2 in accordance with these changes would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes would:

(a) change NA-2 containment instrument air supply valves from Phase A Containment Isolation to Phase B Containment Isolation. The revised designation brings NA-2 into compliance with Virginia Electric and Power Company's response to NUREG-0737, Item II.E.4.2 and provides consistency with the homologous valves in NA-1;

(b) place the NA-1&2 hydrogen recombiner suction and discharge containment isolation valves under administrative control to allow for functional testing which is required by the NA-1&2 TS. The opening of the valves under administrative control (by procedure) for testing will not affect containment conditions or the operation of any other equipment which directly communicates with containment. The containment boundary will be extended to include additional piping and components for approximately four hours every six months. However, the design standards used for construction of the recombiner system ensure that containment conditions will not be degraded and integrity will be maintained for existing accident analysis.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes:

(a) enhance safe operation at NA-2 by assuring the availability of containment instrument air on a Phase A containment isolation signal;

(b) continue to provide containment isolation protection for NA-1&2 even if the valves are open during an accident since it has been demonstrated by analysis and testing that the hydrogen recombiner and associated piping can withstand a DBA pressure excursion and prevent release of radioactivity to the environment. Further, upon receipt of a containment isolation signal the affected valves can be shut remotely from the control room.

(3) involve a significant reduction in a margin of safety. The proposed changes:

(a) assure NA-2 availability of containment instrument air on a Phase A containment isolation signal;

(b) ensure NA-1&2 containment integrity by system design during test conditions and permit required functional testing.

Accordingly, the Commission proposes to determine that these changes do not involve a significant hazards consideration.

*Local Public Document Room location:* Board of Supervisors Office, Louisa County Courthouse, Louisa, Virginia 23093 and the Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901.

*Attorney for licensee:* Michael W. Maupin, Esq., Hunton, Williams, Gay and Gibson, P.O. Box 1535, Richmond, Virginia 23212.

*NRC Project Director:* Lester S. Rubenstein

**Virginia Electric and Power Company, Docket Nos. 50-338 and 50-339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia**

*Date of amendment request:* June 1, 1987

*Description of amendment request:* The proposed changes would revise the North Anna 1 and 2 (NA-1&2) Technical Specifications (TS) associated with primary coolant specific activity limits. The changes are necessary in order to conform with the recommendations of NRC Generic Letter 85-19, "Reporting Requirements on Primary Coolant Iodine Activity."

Specifically, the proposed changes would revise TS 3.4.8 regarding reactor coolant specific activity limits to delete the requirement to shut down the plant if coolant iodine activity limits are exceeded for more than approximately 800 hours in any 12 month period (action "a" for modes 1, 2 and 3); and the requirement for a special report each time coolant iodine activity limits are exceeded (action "a" for modes 1, 2, 3, 4 and 5). These changes are in response to NRC Generic Letter 85-19, "Reporting Requirements on Primary Coolant Iodine Spikes." The special report is to be replaced by a requirement to report on an annual basis the results of any specific activity analysis in which the primary coolant activity exceeds the limits of TS 3.4.8. As discussed in the Generic Letter, the quality of nuclear fuel has been greatly improved over the past decade with the result that coolant iodine activity is normally well below the TS limit. In addition, 10 CFR 50.72(b)(1)(ii) requires the licensee to immediately notify the NRC if fuel cladding failures exceed expected values or are caused by unexpected factors. Thus the 800 hour limit is no longer considered necessary on the basis that proper fuel management and existing reporting requirements should preclude exceeding the limit.

The licensee also proposed to add to TS 6.9.1.5 a requirement for reporting on an annual basis the results of any specific activity analysis in which the primary coolant exceeds the limits of TS 3.4.8, and delete from TS 6.9.2 the requirement for a special report on the same subject. This change is in response to NRC Generic Letter 85-19 and would replace the requirement for a 30-day special report with an annual reporting requirement. The report content would also be changed to include the information requested by the Generic Letter.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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 bases for these criteria with respect to the proposed changes follows:

The licensee provided an analysis which shows that the proposed changes do not involve a significant hazards consideration because operation of NA-1&2 in accordance with these changes would not:

(1) involve a significant increase in the probability or consequence of an accident previously evaluated. The changes involve administrative changes specified in Generic Letter 85-19. The deletion of the requirement to shutdown if the coolant activity limit is exceeded for more than 800 hours in any 12 month period is not considered necessary because of the increased quality of nuclear fuel production and management, and the requirement of 10 CFR 50.72(b)(1)(ii) for immediate notification (if fuel clad failures exceed expected values) should preclude approaching the limit.

(2) create the possibility of a new or different kind of accident from any accident previously identified. The changes involve administrative changes specified in Generic Letter 85-19. The deletion of the requirement to shutdown if the coolant activity limit is exceeded for more than 800 hours in any 12 month period is not considered necessary because of the increased quality of nuclear fuel production and management, and the requirement of 10 CFR 50.72(b)(1)(ii) for immediate notification (if fuel clad failures exceed expected values) should preclude approaching the limit.

(3) involve a significant reduction in a margin of safety. The changes involve administrative changes specified in Generic Letter 85-19. The deletion of the requirement to shutdown if the coolant activity limit is exceeded for more than 800 hours in any 12 month period is not considered necessary because of the increased quality of nuclear fuel production and management, and the requirement of 10 CFR 50.72(b)(1)(ii) for immediate notification (if fuel clad failures exceed expected values) should preclude approaching the limit.

While the staff does not agree that these changes in reporting and surveillance requirements are administrative, the staff agrees with the licensee's conclusions. Accordingly, the Commission proposes to determine that these changes do not involve a significant hazards consideration.

*Local Public Document Room location:* Board of Supervisors Office, Louisa County Courthouse, Louisa, Virginia 23093 and the Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901.

*Attorney for licensee:* Michael W. Maupin, Esq., Hunton, Williams, Gay and Gibson, P.O. Box 1535, Richmond, Virginia 23212.

*NRC Project Director:* Lester S. Rubenstein

**Virginia Electric and Power Company, Docket Nos. 50-280 and 50-281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia**

*Date of amendment requests:* February 23, 1987

*Description of amendment requests:* The proposed amendments would modify the Technical Specifications to allow the accumulator water volume to vary between 975 and 1025 $ft^3$ per accumulator instead of 975 and 989 $ft^3$ per accumulator, and the Height Dependent Heat Flux Hot Channel Factor ($F_Q(z)$) to be as high as 2.32 instead of 2.18 as presently stated in the Technical Specifications.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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 margin of safety. The evaluation of these standards follows:

(1) Since the proposed changes involve parameters which are not accident initiators they will not increase the probability of occurrence of any malfunction or accident previously evaluated. The reanalyzed large break Loss of Coolant Accident (LOCA) analysis verifies that operation under the revised specifications would also not result in a significant increase in

accident consequences and that the acceptance criteria for the emergency core cooling system (ECCS) delineated in 10 CFR 50.46 are met. The proposed changes have also been shown to have no significant impact upon the non-LOCA transients. Therefore, the probability or consequences of an accident previously evaluated will not significantly increase.

(2) The proposed changes involve changes in assumptions made for previously evaluated LOCA accidents. The revised analysis included these parameter changes and demonstrated that they would not cause a new accident. In addition, the increase in steam generator tube plugging was evaluated for impact upon RCS flow and RCS coolant volume. It has been demonstrated that the non-LOCA accidents for which these parameters are significant remain bounded by the existing analyses. Thus, the proposed changes will not create the possibility of a new or different kind of accident.

(3) The water volumes in the range of 975 $ft^3$ to 1025 $ft^3$ per accumulator have no significant impact on the results of the LOCA analysis and therefore, would not significantly reduce the margin of safety. Additionally, the calculated LOCA peak clad temperature was reduced for the revised analysis which used the new parameters. The revised analysis meets the acceptance criteria delineated in 10 CFR 50.46. The non-LOCA accidents are unaffected by the proposed changes, hence the existing analysis remains applicable. Thus the proposed changes do not involve a significant reduction in the margin of safety during LOCA and non-LOCA accidents.

Therefore, the staff proposes to determine that the proposed amendments do not involve a significant hazards consideration.

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

*Attorney for licensee:* Mr. Michael W. Maupin, Hunton and Williams, Post Office Box 1535, Richmond, Virginia 23213.

*NRC Project Director:* Lester S. Rubenstein

Virginia Electric and Power Company, Docket Nos. 50-280 and 50-281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia

*Date of amendment requests:* May 26, 1987

*Description of amendment requests:* The proposed amendments would permit the operation of Surry Units 1 and 2 with 15x15 Surry Improved Fuel (SIF) Assemblies (modified Westinghouse 15x15 Optimized Fuel Assemblies which include two additional features from VANTAGE 5 assemblies) in addition to the Westinghouse Low Parasitic 15x15 (LOPAR) Fuel Assemblies during Cycle 10. The LOPAR fuel assemblies will eventually be replaced by the SIF assemblies.

The new assembly design utilizes Zircaloy grids and smaller diameter thimble tubes. As a consequence of smaller diameter thimble tubes, a change to the Technical Specification governing the maximum allowable control rod drop time is being requested. In addition, a change is being requested to the design Departure from Nucleate Boiling Ratio (DNBR) limit to accommodate the use of the WRB-1 DNB correlation with the new fuel assemblies. In addition to the fuel assembly modifications, thimble plug assemblies will be removed from Surry Units 1 and 2 for Cycle 10 and subsequent cycles. Also, a few editorial changes to the Technical Specifications have been proposed to correct several grammatical errors and to make the Technical Specifications consistent with reactor operating condition.

*Basis for proposed no significant hazards consideration determination:* The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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 licensee has evaluated the changes against the standards provided above and has determined that the changes would involve no significant hazards consideration because:

(1) The revised Loss of Coolant Accident (LOCA) analysis which supports these changes demonstrated that the ECCS acceptance criteria of 10 CFR 50.46 were met. Also the non-LOCA analyses were re-evaluated and met the limits based on the WRB-1 DNB correlation. The WRB-1 correlation limit, like the W-3 limit, was developed on the basis that if the calculated DNBR for a rod was greater than or equal to the design limit there was 95% probability with 95% confidence that the rod was not in DNB. Therefore, the proposed changes would not increase the probability or consequence of an accident previously evaluated.

(2) The proposed fuel assembly design meets all the design criteria applied to the current 15x15 LOPAR design. The 15x15 SIF assembly incorporates some of the features from the 15x15 optimized fuel assembly (OFA) and the Westinghouse 15x15 VANTAGE 5 assembly, both of which have been previously found acceptable by the NRC. In addition, the changes in the assembly design do not change plant systems or procedures. Therefore the changes do not create the possibility of a new or different kind of accident from any accident previously identified.

(3) As stated above, the WRB-1 and W-3 correlation limits were developed on the same basis. Therefore, the changes should not significantly reduce the margin of safety. However, the thimble plug removal proposed by licensee will increase the core-bypass flow slightly from 4.5% to 6%. Per licensee's analysis, this results in an approximate loss of 2% in DNBR margin due to the decreased flow in the flow channels. However, the licensee's analysis shows that the revised ECCS analysis meets the requirements of 10 CFR 50.46. The reanalysis of the non-LOCA accidents also meet the analysis limits using the WRB-1 correlation. Therefore, the changes do not involve a significant reduction in margin of safety.

Also, the Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (51 FR 7751) of actions not likely to involve a significant hazards consideration. Example (iii) of this guidance states: "For a nuclear power reactor, a change resulting from a nuclear reactor core reloading if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable."

The proposed license amendments are directly related to the above example in that the amendments are related to core reload and the new fuel is similar to that found acceptable by the NRC, and the analytical methods used by the licensee in the required reload analyses have

been previously found acceptable by the NRC.

In addition, Example (i) was found to be applicable to the proposed editorial changes. Example (i) states: "A purely administrative change to technical specifications: for example, a change to achieve consistency throughout the technical specifications, correction of error, or a change in nomenclature." The editorial changes reflect an effort to make the Technical Specifications Reactor Operating Conditions consistent and correct several grammatical errors.

Based on the above evaluation, the staff proposes to determine that the requested amendments do not involve a significant hazard consideration.

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

*Attorney for licensee:* Mr. Michael W. Maupin, Hunton and Williams, Post Office Box 1535, Richmond, Virginia 23213.

*NRC Project Director:* Lester S. Rubenstein.

**Wolf Creek Nuclear Operating Corporation, Kansas Gas and Electric Company, Kansas City Power & Light Company, Kansas Electric Power Cooperative, Inc., Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas**

*Date of amendment request:* June 16, 1987.

*Description of amendment request:* The proposed amendment request revises Wolf Creek Generating Station (WCGS) Technical Specification 3/4.3.1, Reactor Trip System Instrumentation, in the area of specified surveillance intervals and out-of-service times for Reactor Protection System Instrumentation. The requested revisions are based on changes approved generically as a result of the Nuclear Regulatory Commission's review of WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System" and WCAP-10271, Supplement 1.

The proposed changes are as follows:

1. Increase the surveillance interval for RPS analog channel operational tests from once per month to once per quarter,

2. Increase the time during which an inoperable RPS analog channel may be maintained in an untripped condition from one hour to six hours, and

3. Increase the time an inoperable RPS analog channel may be bypassed to allow testing of another channel in the same function from two hours to four hours.

The Wolf Creek design does not currently include the capability for bypass testing. Therefore, the portions of WCAP-10271 and WCAP-10271 Supplement 1, which concerns bypass testing, are not applicable to this Significant Hazard Evaluation.

*Basis for proposed no significant hazards determination:* In accordance with the requirements of 10 CFR 50.92, the licensee has submitted the following no significant hazards determination:

Wolf Creek Nuclear Operating Corporation has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed RPS technical specification changes for the Wolf Creek Generating Station and determined that a significant hazards consideration is not involved. In support of this conclusion, the following analysis is provided.

*Criterion 1* Operation of Wolf Creek Generating Station in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Implementation of the proposed changes is expected to result in an acceptable increase in total Reactor Protection System yearly unavailability. This increase, which is primarily due to less frequent surveillance, results in an increase of similar magnitude in the probability of an Anticipated Transient Without Trip (ATWT) and in the probability of core melt resulting from an ATWT.

Implementation of the proposed changes is expected to result in a significant reduction in the probability of core melt from inadvertent reactor trips. This is a result of a reduction in the number of inadvertent reactor trips occurring during testing of RPS Instrumentation. This reduction is primarily attributable to less frequent surveillance testing.

The reduction in inadvertent core melt probability is sufficiently large to counter the increase in ATWT core melt probability resulting in an overall reduction in total core melt probability.

The proposed changes do not result in an increase in the severity or consequences of an accident previously evaluated. Implementation of the proposed changes affects the probability of failure of the RPS but does not alter the manner in which protection is afforded nor the manner in which limiting criteria are established.

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

The proposed changes do not result in a change in the manner in which the Reactor Protection System provides plant protection. No change is being made which alters the functioning of the Reactor Protection System (other than in a test mode). Rather, the likelihood or probability of the Reactor Protection System functioning properly is affected as described above. Therefore, the proposed changes do not create the possibility of a new or different kind of accident nor involve a reduction in a margin of safety as defined in the Safety Analysis Report.

The proposed changes do not involve hardware changes. The Wolf Creek design does not currently have the capability for full bypass testing. Some existing instrumentation is designed to be tested in bypass and current technical specifications allow testing in bypass. Testing in bypass is also recognized by IEEE Standards.

*Criterion 3* The proposed license amendment does not involve a significant reduction in a margin of safety.

The proposed changes do not alter the manner in which safety limits, limiting safety system setpoints or limiting conditions for operation are determined. The impact of reduced testing other than as addressed above is to allow a longer time interval over which instrument uncertainties (e.g., drift) may act. Experience at two Westinghouse plants with extended surveillance intervals has shown the initial uncertainty assumptions to be valid for reduced testing.

Implementation of the proposed changes is expected to result in an overall improvement in safety by:

a. Fewer inadvertent reactor trips due to less frequent testing which minimizes the time spent in a partial trip condition.

b. Higher quality repairs leading to improved equipment reliability due to longer repair times.

c. Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation. This is due to less frequent distraction of the operator and shift supervisor to attend to instrumentation testing.

*Example*

10 CFR Part 50 - Statements of Consideration contains, "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Consideration." One of the examples provided is:

(vi) A change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational mode or design method.

As previously stated, implementation of the proposed changes results in an acceptable increase in the probability of ATWT and ATWT core melt. Overall core melt probability decreases. Implementation of the proposed changes does not increase the consequences of a previously analyzed accident nor reduce a margin of safety. Functioning of the RPS and the manner in which limiting criteria is established is unaffected. The stated example of a change which is likely not to involve a significant hazards consideration is applicable therefore to the proposed changes.

*Conclusions*

The foregoing analysis demonstrates that the proposed amendment to Wolf Creek Generating Station technical specifications does not involve a significant increase in the probability or consequences of a previously evaluated accident, does not create the possibility of a new or different kind of accident and does not involve a significant reduction in a margin of safety. Additionally, fewer inadvertent reactor trips are expected, equipment reliability is expected to increase and operator effectiveness is expected to improve.

Based upon the preceding analysis, Wolf Creek Nuclear Operating Corporation concludes that the proposed amendment does not involve a significant hazards consideration.

Based on the previous discussion, the licensee concluded that the proposed amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated; nor create the possibility of a new or different kind of accident from any accident previously evaluated; nor involve a significant reduction in the required margin of safety. The NRC staff has reviewed the licensee's no significant hazards considerations determination and agrees with the licensee's analysis. The staff has, therefore, made a proposed determination that the licensee's request does not involve a significant hazards consideration.

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

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

*NRC Project Director:* Jose A. Calvo

**Wolf Creek Nuclear Operating Corporation, Kansas Gas and Electric Company, Kansas City Power & Light Company, Kansas Electric Power Cooperative, Inc., Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas**

*Date of amendment request:* June 19, 1987.

*Description of amendment request:* The proposed amendment request revises Wolf Creek Generating Station (WCGS) Technical Specification 3.5.1, Accumulators, to allow the Unit to remain in Hot Standby with Reactor Coolant System pressure less than or equal to 1000 psig with one accumulator inoperable.

*Basis for proposed no significant hazards consideration determination:* In accordance with the requirements of 10 CFR 50.92, the licensee has submitted the following no significant hazards determination:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. This amendment is being proposed to make the technical specification Limiting Condition for Operation (LCO) applicability and the technical specification Action Statements consistent. This change does not affect the performance of the Emergency Core Cooling System accumulators.

2. The proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment only affects operation at a RCS pressure of less than or equal to 1000 psig. Postulated design basis accidents that take credit for automatic actuation of the accumulators begin above 1000 psig.

3. The proposed changes does not involve a significant reduction in a margin of safety. This change does not affect any technical specification margin of safety. The proposed change does not affect the capability of the accumulators to perform their safety function. This change does not alter any safety limits or limiting safety system setpoints.

Based on the above discussions it has been determined that the requested Technical Specification revision does not involve a significant increase in the probability or consequences of an accident or other adverse condition over previous evaluations; or create the possibility of a new or different kind of accident or condition over previous evalautions; or involve a significant reduction in a margin of safety. Therefore, the requested license amendment does not involve a significant hazards consideration.

Based on the previous discussion, the licensee concluded that the proposed amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated; nor create the possibility of a new or different kind of accident from any accident previously evaluated; nor involve a significant reduction in the required margin of safety. The NRC staff has reviewed the licensee's no significant hazards considerations determination and agrees with the licensee's analysis. The staff has, therefore, made a proposed determination that the licensee's request does not involve a significant hazards consideration.

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

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

*NRC Project Director:* Jose A. Calvo

**Wolf Creek Nuclear Operating Corporation, Kansas Gas and Electric Company, Kansas City Power & Light Company, Kansas Electric Power Cooperative, Inc., Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas**

*Date of amendment request:* June 19, 1987

*Description of amendment request:* The proposed amendment request revises Wolf Creek Generating Station (WCGS) Technical Specification 3/4.3.3, Radiation Monitoring for Plant Operation. The proposed revision changes the required number of minimum channels operable for Table 3.3-6 Functional Unit 1.a., Containment Atmosphere - Gaseous Radioactivity High (GT-RE 31 & 32). The requested revision also modifies Actions 27 and 30 of Table 3.3-6 to permit an allowed outage time of 72 hours with the number of operable channels one less than the minimum channels operable requirement.

*Basis for proposed no significant hazards consideration determination:* In accordance with the requirements of 10 CFR 50.92, the licensee has submitted the following no significant hazards determination:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. Reducing the required minimum channels operable for radiation monitors GT-RE-31 and 32 does not significantly increase the probability or consequences on an accident previously evaluated. One channel will be operable or Action 28 will be followed. This action requires containment purge valve closure. This assures the fulfillment of the safety function of Table 3.3-6 Functional Unit 1.a. Increasing the allowed outage times for Actions 27 and 30 will not significantly increase the probability or consequences of an accident previously evaluated. This change allows an extended time period for the diagnosis and repair of inoperable radiation monitors to which these action statements are applicable. During the allowed outage time, redundant radiation monitors remain operable. If no channels are operable only one hour is allowed until initial operation of the Control Room Emergency Ventilation System or the Emergency Exhaust System, as appropriate, is established.

2. The proposed changes do not create the possibility of a new or different kind of accident from any previously evalauted. The proposed changes do not involve any change in radiation monitor setpoints or physical design of the involved systems. There are no new failure mechanisms associated with the proposed changes. The proposed changes will allow additional time for the diagnosis and repair of inoperable radiation monitors. The proposed allowed outage times remain

consistent with the requirements for overall system availability.

3. The proposed changes do not involve a significant reduction in a margin of safety. These changes do not affect any Technical Specification margin of safety. The proposed changes do not alter the manner in which safety limits and limiting safety system setpoints are determined. Redundant radiation monitors remain operable during the extended allowed outage time and under the revised minimum operable channels Requirement.

Based on the above discussions it has been determined that the requested Technical Specification revisions do not involve a significant increase in the probability of consequences of an accident or other adverse condition over previous evaluations; or create the possibility of a new or different kind of accident or condition over previous evaluations; or involve a significant reduction in a margin of safety. Therefore, the requested license amendment does not involve a signficant hazards consideration.

Based on the previous discussion, the licensee concluded that the proposed amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated; nor create the possibility of a new or different kind of accident from any accident previously evaluated; nor involve a significant reduction in the required margin of safety. The NRC staff has reviewed the licensee's no significant hazards considerations determination and agrees with the licensee's analysis. The staff has, therefore, made a proposed determination that the licensee's request does not involve a significant hazards consideration.

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

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

*NRC Project Director:* Jose A. Calvo

**Wolf Creek Nuclear Operating Corporation, Kansas Gas and Electric Company, Kansas City Power & Light Company, Kansas Electric Power Cooperative, Inc., Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas**

*Date of amendment request:* June 19, 1987.

*Description of amendment request:* The proposed amendment request revises Wolf Creek Generating Station (WCGS) Technical Specification 3/4.10.4, Reactor Coolant Loops, replacing references to the Reactor Protective System P-7 permissive interlock to the P-10 permissive interlock.

*Basis for proposed no significant hazards consideration determination:* In accordance with the requirements of 10 CFR 50.92, the licensee has submitted the following no significant hazards determination:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed amendment would only eliminate surveillance requirements for P-13 which have no effect on plant operation while using this Technical Specification. With the turbine off line, P-13 does not provide an input into P-7. This leaves P-10 as the only input which can trip the P-7 bistable.

2. The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated. There are no new failure modes or mechanisms associated with the proposed changes. No additional plant equipment is being added nor will any existing requirements. The proposed change does not change any setpoints related to nuclear safety.

3. The proposed change does not involve a significant reduction in a margin of safety. The proposed amendment does not affect any technical specification margin of safety. This amendment only eliminates the need to perform the surveillance requirements for P-13 while the plant is operating under this Technical Specification. The proposed change does not alter the manner in which safety limits and limiting safety system setpoints are determined.

Based on the above discussions it has been determined that the requested Technical Specification revisions do not involve a significant increase in the probability or consequences of an accident or other adverse condition over previous evaluations; or create the possibility of a new or different kind of accident or condition over previous evaluations; or involve a significant reduction in a margin of safety. Therefore, the requested license amendment does not involve a signficant hazards consideration.

Based on the previous discussion, the licensee concluded that the proposed amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated; nor create the possibility of a new or different kind of accident from any accident previously evaluated; nor involve a significant reduction in the required margin of safety. The NRC staff has reviewed the licensee's no significant hazards considerations determination and agrees with the licensee's analysis. The staff has, therefore, made a proposed determination that the licensee's request does not involve a significant hazards consideration.

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

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

*NRC Project Director:* Jose A. Calvo

## PREVIOUSLY PUBLISHED NOTICES OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO OPERATING LICENSES AND PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION AND OPPORTUNITY FOR HEARING

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

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

**Gulf States Utilities Company, Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana**

*Date of amendment request:* March 10, 1987.

*Brief description of amendment request:* The proposed amendment would authorize a one-time schedular extension of up to 43 days to perform the surveillance requirements of Section 4.6.6.2.c of the Technical Specifications regarding the testing of the flow rate of each of the primary containment/drywell hydrogen mixing trains.

*Date of publication of individual notice in* Federal Register: June 18, 1987 (52 FR 23218).

*Expiration date of individual notice:* July 17, 1987

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

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

*Date of amendment request:* May 21, 1987

*Description of amendment request:* The amendment would revise the Technical Specifications to allow a reduced integrated leak rate test (ILRT) duration in accordance with NRC approved methodology.

*Date of publication of individual notice in* Federal Register: June 10, 1987 (52 FR 22013).

*Expiration date of individual notice:* July 10, 1987.

*Local Public Document Room location:* White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601

## NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

During the period since publication of the last bi-weekly 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 and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing in connection with these actions was published in the Federal Register as indicated. No request for a hearing or petition for leave to intervene was filed following this notice.

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 amendments, (2) the amendments, and (3) the Commission's related letters, Safety Evaluations and/or Environmental Assessments as indicated. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Licensing.

**Alabama Power Company, Docket No. 50-348, Joseph M. Farley Nuclear Plant, Unit No. 1, Houston County, Alabama.**

*Date of application for amendment:* October 25, 1985, as supplemented September 29, 1986

*Brief description of amendment:* Technical Specification figures of heatup and cooldown limitations are revised based on results of analysis of Capsule "U" Reactor Vessel Material Radiation Surveillance program.

*Date of issuance:* June 23, 1987

*Effective date:* June 23, 1987

*Amendment No.* 71

*Facility Operating License No. NPF-2:* Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: December 4, 1985 (50 FR 49780), renoticed May 20, 1987 (52 FR 18971) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 23, 1987.

The September 29, 1986 letter provided information that did not change the initial determination published in the Federal Register.

No significant hazards consideration comments received: No

*Local Public Document Room location:* George S. Houston Memorial Library, 212 W. Burdeshaw Street, Dothan, Alabama 36303

**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:* March 11, 1987.

*Brief description of amendment:* The amendment revises the Technical Specifications to incorporate station batteries nos. 23 and 24 under the same provisions as batteries nos. 21 and 22. The actual modification to Indian Point 2 to add the 2 batteries to the battery system was reviewed and accepted in NRC Safety Evaluation dated May 2, 1980.

*Date of issuance:* June 17, 1987

*Effective date:* June 17, 1987

*Amendment No.:* 120

*Facilities Operating License No. DPR-26:* Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: April 22, 1987 (52 FR 13335). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 17, 1987.

No significant hazards consideration comments received: No

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

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

*Date of application for amendments:* April 9, 1987, as supplemented May 18 and June 15, 1987

*Brief description of amendments:* The amendments change the Technical Specifications to reflect the third of several refueling stages involved in the transition to the use of optimized fuel assemblies in Unit 2.

*Date of Issuance:* June 22, 1987

*Effective Date:* June 22, 1987

*Amendent Nos.:* 73 and 54

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

*Date of initial notice in* Federal Register: May 20, 1987 (52 FR 18977) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 22, 1987.

No significant hazards consideration comments received: No

*Local Public Document Room location:* Atkins Library, University of North Carolina, Charlotte (UNCC) Station, North Carolina 28223

**Duquesne Light Company, Docket No. 50-334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania**

*Date of application for amendment:* January 15, 1986 as supplemented by letters dated May 8, June 30, 1986, January 15, March 5, March 17 and April 6, 1987.

*Brief description of amendment:* The amendment changed the Technical Specifications for Beaver Valley Unit 1 to (1) provide clear, retyped pages for the Index, (2) reflect the new titles and responsibilities of the new Duquesne Light Company organization and (3) eliminate blank pages whose contents have been deleted by previous amendments.

*Date of issuance:* June 23, 1987

*Effective date:* June 23, 1987

*Amendment No.* 110

*Facility Operating License No. DPR-66.* Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: March 12, 1986 (51 FR 8590) and May 6, 1987 (52 FR 16943) The supplemental letter dated May 8, 1986, provided additional clarifying information and did not change the findings of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 23, 1987.

No significant hazards consideration comments received: No

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

General Public Utilities Nuclear Corporation, Docket No. 50-320, Three Mile Island Nuclear Station Unit 2 (TMI-2), Dauphin County, Pennsylvania

*Date of application for amendment:* January 27, 1987

*Brief description of amendment:* The amendment revises the Technical Specifications by redefining the scope of licensee procedures and changes thereto that require NRC staff approval prior to implementation. Only those procedures and changes thereto which alter the distribution or processing of a quantity of radiological material, the release of which could cause the magnitude of radiological releases to exceed the design objectives of 10 CFR Part 50 Appendix I.

*Date of Issuance:* June 25, 1987

*Effective Date:* June 25, 1987

*Amendment No.:* 28

*Facility Operating Licensing No. DPR-73:* Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: May 6, 1987 (52 FR 16947) The Commission's related evaluation is contained in a Safety Evaluation dated June 25, 1987.

No significant hazards consideration comments received: No.

*Local Public Document Room Location:* State Library of Pennsylvania, Government Publications Section, Education Building, Commonwealth and Walnut Street, Harrisburg, Pennsylvania 17126.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50-424, Vogtle Electric Generating Plant, Unit 1, Burke County Georgia

*Date of application for amendment:* March 30, 1987

*Brief description of amendment:* The amendment modifies the Technical Specifications to increase the shutdown margin requirements shown in Figure 3.1-2 and changes the title of that figure.

*Date of issuance:* June 23, 1987

*Effective date:* June 23, 1987

*Amendment No.:* 1

*Facility Operating License No. NPF-68.* Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: May 20, 1987 (52 FR 18981) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 23, 1987.

No significant hazards consideration comments received: No

*Local Public Document Room location:* Burke County Public Library, 4th Street, Waynesboro, Georgia 30830

Iowa Electric Light and Power Company, Docket No. 50-331, Duane Arnold Energy Center, Linn County, Iowa

*Date of application for amendment:* August 29, 1978 as revised by letters dated November 5, 1981 and March 16, 1984 and finally supplemented by a letter dated August 15, 1985.

*Brief description of amendment:* The amendment revised the Duane Arnold Energy Center Technical Specifications (TS) to include containment isolation valves MO-4423, CV-2410, MO-2400, MO-2238 and CV-2211 in TS Table 3.7-2.

*Date of issuance:* June 25, 1987

*Effective date:* June 25, 1987

*Amendment No.:* 144

*Facility Operating License No. DPR-49.* Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: October 26, 1983 (48 FR 49588) and May 23, 1984 (49 FR 21831). The information in the August 15, 1985 submittal is covered by the May 23, 1984 notice.

The Commission's related evaluation of the amendment is contained in a letter dated June 25, 1987.

No significant hazards consideration comments received: No

*Local Public Document Room location:* Cedar Rapids Public Library, 500 First Street, S.E., Cedar Rapids, Iowa 52401.

Maine Yankee Atomic Power Company, Docket No. 50-309, Maine Yankee Atomic Power Station, Lincoln County, Maine

*Date of application for amendment:* February 24, 1987

*Brief description of amendment:* The changes to the Technical Specifications (TS) concerns the Semiannual Radioactive Effluent Release Report (SRERR) and the Special Report (SR). The change to TS concerning SRERR allows additional time for the filing of supplemental dose and meteorological summary reports. The change to the TS concerning SRERR ensures that the specification references are consistent with the current TS.

The change to the TS concerning SRERR provides the added time necessary for preparing the annual summaries of meteorological history and estimated dose commitments resulting from plant operation. These reports are required to be submitted within 89 days after January 1 of each year. This has been accomplished by dividing the TS concerning SRERR into two sections. Section one contains the requirements for the SRERR and Section two outlines the requirements for submitting annual reports of the Estimated Dose and Meteorological Summary. This change will not affect the submittal of plant radioactive effluent data required by 10 CFR 50.36a paragraph (a)(2).

The change to TS concerning SR updates the references for the Excessive Radioactive Release Report. Presently the TS concerning SR refers to an incorrect TS section. This TS change corrects that situation.

*Date of issuance:* June 25, 1987

*Effective date:* 60 days from date of issuance

*Amendment No.:* 99

*Facility Operating License No. DPR-36.* Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: April 8, 1987 (52 FR 11366) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 25, 1987.

No significant hazards consideration comments received: No

*Local Public Document Room location:* Wiscasset Public Library, High Street, Wiscasset, Maine 04578

Maine Yankee Atomic Power Company, Docket No. 50-309, Maine Yankee Atomic Power Station, Lincoln County, Maine

*Date of application for amendment:* March 13, 1985 as clarified by letters dated January 15, 1986 and January 13, 1987.

*Brief description of amendment:* This amendment brings Maine Yankee Technical Specifications into conformance with the requirements set forth in USNRC Generic Letter 83-37 pertaining to the Reactor Coolant System (RCS) Vent System. Generic Letter 83-37 requires at least one RCS vent path to be operable and closed at all times. For Maine Yankee, which uses a pressure operated relief valve (PORV) as a RCS vent, the block valve is not required to be closed if the PORV is operable.

*Date of issuance:* June 25, 1987

*Effective date:* 60 days from date of issuance

*Amendment No.:* 109

*Facility Operating License No. DPR-36.* Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: May 21, 1985 (50 FR 20928) and May 20, 1987 (52 FR 18982). The Commission's related evaluation of the

amendment is contained in a Safety Evaluation dated June 25, 1987.

No significant hazards consideration comments received: No

*Local Public Document Room location:* Wiscasset Public Library, High Street, Wiscasset, Maine 04578.

**Public Service Company of Colorado, Docket No. 50-267, Fort St. Vrain Nuclear Generating Station, Platteville, Colorado**

*Date of amendment request:* January 15, 1987

*Brief description of amendment:* The amendment modified the Fort St. Vrain (FSV) Technical Specification to require both sections of both steam generators to be available for decay heat removal in power operation. It also assures that water sources are available to the appropriate steam generator sections.

*Date of issuance:* June 29, 1987

*Effective date:* This amendment becomes effective when the Commission approves operation of FSV at power levels higher than the 35 percent power level currently authorized.

*Amendment No.:* 55

*Facility Operating License No. DPR-34.* Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: February 26, 1987 (52 FR 5866) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 29, 1987.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* Greeley Public Library, City Complex Building, Greeley, Colorado

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

*Date of application for amendments:* October 3, 1986 and supplemented by letter dated October 10, 1986

*Brief description of amendments:* These amendments delete the specified maximum fuel rod weight limit to allow current fuel to be in compliance with the Salem 1 and 2 Technical Specifications. The supplementing submittal corrected the wording of Section 5.3.1, and did not contain substantive changes.

*Date of issuance:* June 19, 1987

*Effective date:* June 19, 1987

*Amendment Nos.* 80 and 54

*Facility Operating License Nos. DPR-70 and DPR-75.* These amendments revised the Technical Specifications.

*Date of initial notice in* Federal Register: (52 FR 11372) April 8, 1987. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 19, 1987.

No significant hazards consideration comments received: No

*Local Public Document Room location:* Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079

**Washington Public Power Supply System, Docket No. 50-397, WNP-2, Richland, Washington**

*Date of amendment request:* March 27, 1987, as supplemented April 22, 1987.

*Brief description of amendment:* This amendment revised the technical specifications to (1) authorize operation of WNP-2 for Cycle 3 with a fuel reload using Advanced Nuclear Fuels Corporation fuel assemblies, analyses and methodologies and (2) establish operating limits for Cycle 3 operation.

*Date of issuance:* June 2, 1987

*Effective date:* June 2, 1987

*Amendment number:* 45

*Facility Operating License No. NPF-21:* Amendment revises the license.

*Date of initial notice in the* Federal Register: April 22, 1987 (52 FR 13352) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 2, 1987.

No significant hazards consideration comments received: No

*Local Public Document Room location:* Richland Public Library, Swift and Northgate Streets, Richland, Washington 99352

**Yankee Atomic Electric Company, Docket No. 50-029, Yankee Nuclear Power Station, Franklin County, Massachusetts**

*Date of application for amendment:* January 29, 1987

*Brief description of amendment:* The amendment modifies the Technical Specifications requirement on the minimum number of operable neutron detectors from twelve, with two per quadrant, to nine, with one per quadrant for Cycle 19 operation.

*Date of issuance:* June 26, 1987

*Effective date:* June 26, 1987

*Amendment No.:* 106

*Facility Operating License No. DPR-3.* Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: March 12, 1987 (52 FR 7703) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 26, 1987.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* Greenfield Community College, 1 College Drive, Greenfield, Massachusetts 01301.

## NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND FINAL DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION AND OPPORTUNITY FOR HEARING (EXIGENT OR EMERGENCY CIRCUMSTANCES)

During the period since publication of the last bi-weekly 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 and Proposed No Significant Hazards Consideration Determination and Opportunity for 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 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, 1717 H Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Licensing.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendments. By August 14, 1987, 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 petition for leave to intervene. Requests for a hearing and petitions 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. 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 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 fifteen (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 fifteen (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, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. 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, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, NW., Washington, DC, by the above date. Where petitions are filed during the last ten (10) days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to (*Project Director*): petitioner's name and telephone number; date petition was mailed; plant name; and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel-Bethesda, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for 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).

**Arizona Public Service Company, et al., Docket No. STN 50-528, Palo Verde Nuclear Generating Station (PVNGS), Unit 1, Maricopa County, Arizona**

*Date of application for amendment:* May 10, 1987, as supplemented by letter dated May 14, 1987.

*Brief description of amendment:* The amendment revises Technical Specification 3/4.11.1 for a period not to exceed March 31, 1988, to allow the release of secondary system liquid waste to the onsite evaporation pond, while the concentration of Antimony-124 exceeds $5 \times 10^{-7}$ micro Ci/ml, provided that the concentration does not exceed the limits of 10 CFR Part 20, Appendix B, Table II, Column 2.

*Date of issuance:* June 3, 1987

*Effective date:* June 3, 1987

*Amendment No.:* 18

*Facility Operating License No. NPF-41:* Amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes (52 FR 18763, May 19, 1987). No comments were received by the due date. A request for an extension of the comment period was received after the amendment was issued.

The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of Arizona, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated June 3, 1987.

*Attorney for licensees:* Mr. Arthur C. Gehr, Snell & Wilmer, 3100 Valley Center, Phoenix, Arizona 85007.

*Local Public Document Room location:* Phoenix Public Library, Business, Science and Technology Department, 12 East McDowell Road, Phoenix, Arizona 85004.

*NRC Project Director:* George W. Knighton

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

*Date of application for amendment:* June 1, 1987, as supplemented June 2 and 4, 1987

*Brief description of amendment:* The amendment revised the Hope Creek Technical Specifications to permit the plant to continue operation until September 21, 1987, or until the first forced outage of sufficient duration to repair the monitor, whichever first occurs, with the acoustic monitor for one of the safety relief valve tailpipes inoperable.

*Date of Issuance:* June 17, 1987

*Effective Date:* June 4, 1987

*Amendment No.:* 5

*Facility Operating License No. NPF-57:* Amendment revised the Technical Specifications.

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

The Commission's related evaluation of the amendment, consultation with the State of New Jersey and final no significant hazards considerations determination are contained in a Safety Evaluation dated June 17, 1987.

*Attorney for licensee:* Conner and Wetterhahn, 1747 Pennsylvania Avenue Washington, DC 20006

*Local Public Document Room location:* Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070.

*NRC Project Director:* Walter R. Butler

Dated at Bethesda, Maryland this 9th day of July, 1987.

For the Nuclear Regulatory Commission

Steven A. Varga,

*Director, Division of Reactor Projects I/II, Office of Nuclear Reactor Regulation*

[FR Doc. 87-15905 Filed 7-14-87; 8:45 am]

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