conducted in accordance with the hybrid hearing procedures. In essence, those procedures limit the time available for discovery and require that an oral argument be held to determine whether any contentions must be resolved in an adjudicatory hearing. If no party to the proceeding timely requests oral argument, and if all untimely requests for oral argument are denied, then the usual procedures in 10 CFR part 2, Subpart G apply.

For further details with respect to this action, see the application for amendments dated March 23, 1999, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010 for Byron Station, and the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481 for Braidwood Station.

Dated at Rockville, Maryland, this 10th day of June 1999.

For the Nuclear Regulatory Commission.

**Stewart N. Bailey,**

*Project Manager, Section 2, Project Directorate 3, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.*

[FR Doc. 99–15244 Filed 6–15–99; 8:45 am]

**BILLING CODE 7590–01–P**

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## NUCLEAR REGULATORY COMMISSION

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

### I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from May 21, 1999, through June 4, 1999. The last biweekly notice was published on June 2, 1999 (64 FR 29707).

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

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

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

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

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

By July 19, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

*Date of amendments request:* May 23, 1997, as revised by letters dated September 27, 1998, and May 26, 1999.

*Description of amendments request:* The proposed amendments would revise Technical Specification (TS) Limiting Condition of Operation (LCO) 3.4.14 and TS Sections 5.5.9 and 5.6.8 to allow the use of steam generator (SG) tube sleeves as an alternative to plugging defective SG tubes. The May 26, 1999, letter completely revised the May 23, 1997, request for amendments, and this notice supersedes the original **Federal Register** notice dated July 30, 1997 (62 FR 40845).

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

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

The proposed change to TS LCO 3.4.14.d and e will replace the leakage limits of 1 gallon per minute (gpm) primary to secondary leakage through all SGs and 720 gallon per day (gpd) through any one SG with a new limit of 150 gpd through any one SG. This is a more restrictive change. A TS limit of 150 gpd primary to secondary Leakage through any one steam generator is significantly less than the initial conditions assumed in the safety analyses. The 150 gpd limit is based on operating experience as an indication of one or more propagating tube leak mechanisms. The Steam Generator Tube Surveillance Program described in TS Section 5.5.9 ensures that the structural integrity of the SG tubes is maintained. The leakage rate limit of 150 gpd for any one SG provides additional assurance against tube rupture at normal and faulted conditions and provides additional assurance that cracks will not propagate to burst prior to detection by leakage monitoring methods and commencement of plant shutdown. Therefore, this change to TS LCO 3.4.14.e will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to TS 5.5.9 will add inservice inspection requirements for SG tube sleeves. These requirements will ensure that all installed SG tube sleeves will be inspected prior to initial operation and routinely thereafter, to assure the capability of each sleeve to perform its design function during each operating cycle. The tube sleeves will be the Combustion Engineering, Inc. (CE or ABB–CE) Leak Tight sleeves, as described in CE report CEN–630–P, ''Repair of ¾'' O.D. Steam Generator Tubes Using Leak Tight Sleeves,'' Revision 02, dated June 1997. (This proprietary report is provided as Enclosure 4 with this submittal.) The tube sleeve dimensions, materials and joints are designed to the applicable ASME [American Society of Mechanical Engineers] Boiler and Pressure Vessel code requirements. An extensive test program was performed that demonstrated that the sleeves will fulfill their intended function as leak tight structural members. Evaluation of sleeved tubes indicates no detrimental effects on the sleeve-tube assembly resulting from reactor coolant system flow, coolant chemistries, or thermal and pressure conditions. Structural analyses of the sleeve-tube assembly have established its integrity under normal and accident conditions. Mechanical testing using ASME code stress allowables was performed to support the analyses. Also, corrosion tests were performed and revealed no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions. A sleeved tube will exhibit greater hydraulic resistance and reduced heat transfer capability than an un-sleeved tube. However, these effects are much less than would be imposed by taking the tube out of service by plugging. Section 10.0 of CE report CEN–630–P describes the analyses to determine the hydraulic and heat transfer effects. Calculations using plant-specific information will identify sleeve-to-plug equivalency ratios. The proposed changes to the SG inservice inspection program will assure that sleeved SG tubes will meet the structural requirements of tubes that are not defective. The proposed sleeve plugging limit of 35% of nominal wall will ensure that the sleeves remaining in service will perform their design function. Also, installation of sleeves will not significantly [a]ffect the primary system flow rate or the heat transfer capability of the SGs. Therefore, this change to TS section 5.5.9 will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The change to the SG reporting requirements in TS section 5.6.8 will ensure that the number of sleeved SG tubes will be reported to the NRC along with the number of plugged tubes. This is an administrative change that has no effect on the operation or maintenance of the plant and will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to TS LCO 3.4.14.d and e will replace the leakage limits of 1 gpm primary to secondary leakage through all SGs and 720 gpd through any one SG with a new limit of 150 gpd through any one SG. This is a more restrictive change that will provide added assurance against steam generator tube ruptures. Since the current allowable primary to secondary leakage is being reduced, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to TS section 5.5.9 for the SG inservice inspection program will assure that sleeved SG tubes will meet the structural requirements of tubes that are not defective. Also, installation of sleeves will not significantly [a]ffect the primary system flow rate or the heat transfer capability of the SGs. Therefore, this change to TS section 5.5.9 will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The change to the SG reporting requirements in TS section 5.6.8 will ensure that the number of sleeved SG tubes will be reported to the NRC along with the number of plugged tubes. This is an administrative change that has no effect on the operation or maintenance of the plant and will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change to TS LCO 3.4.14.d and e will replace the leakage limits of 1 gpm primary to secondary leakage through all SGs and 720 gpd through any one SG with a new limit of 150 gpd through any one SG. This is a more restrictive change that will provide added assurance against steam generator tube ruptures. Since the current allowable primary to secondary leakage is being reduced, this change will not involve a significant reduction in a margin of safety.

The proposed changes to TS section 5.5.9 for the SG inservice inspection program will assure that sleeved SG tubes will meet the structural requirements of tubes that are not defective. Also, installation of sleeves will not significantly [a]ffect the primary system flow rate or the heat transfer capability of the SGs. Therefore, this change to TS section 5.5.9 will not involve a significant reduction in a margin of safety.

The change to the SG reporting requirements in TS section 5.6.8 will ensure that the number of sleeved SG tubes will be reported to the NRC along with the number of plugged tubes. This is an administrative change that has no effect on the operation or maintenance of the plant and will not involve a significant reduction in a margin of safety.

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

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

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

*NRC Section Chief:* Stephen Dembek.

*Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois*

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

*Description of amendment request:* The proposed amendments would revise the basis for evaluation of the reactor building ventilation (VR) system exhaust plenum masonry walls. Specifically, the amendment would approve the use of different methodology and acceptance criteria for the reassessment of certain masonry walls subjected to transient pressurization loads resulting from a high energy line break.

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

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

The change involves reassessment of the VR exhaust plenum due to a transient pressurization during a Main Steam Line Break (MSLB). Since the transient pressurization is a result of the MSLB, and the block walls and the dampers are not initiators of any accident, the probability of an accident previously evaluated is not affected.

This analysis does not affect the total amount of radioactive release due to the MSLB Outside of the Primary Containment, so the total offsite dose consequences does not change. A small portion of the release, which passes the dampers prior to closure, will now be an elevated release via the plant ventilation stack instead of a ground level release. The original analysis assumed the entire release was a ground level release, and thus remains bounding for the MSLB accident.

The Control Room and Auxiliary Electric Equipment Room (AEER) dose consequences are impacted only slightly due to the small amount of steam/air mixture released from the new pressure relief damper. The steam/air mixture becomes mixed with the air volume in that area of the Auxiliary Building but was all assumed to be available for inleakage to the Control Room and AEER. The dose increase for the Control Room and AEER is less than or equal to 0.05 Rem thyroid and negligible change to the whole body dose, such that the dose due to the MSLB accident remains much less than the DBA LOCA dose and General Design Criteria 19. The MSLB accident dose consequences remain bounded by the Design Basis Loss of Coolant Accident.

The effects of the steam released by the pressure relief damper into the Auxiliary Building has been evaluated for environmental qualification impact on systems, structures and components (SSCs) in the area of the Auxiliary Building affected for both radiation and steam/temperature affects. The effect on area temperature is about 4 °F and is above initial temperature for not more than 24 hours. The change in humidity is negligible, and radiation dose impact is small and bounded by previous calculations.

These consequences assume that the VR exhaust plenum masonry walls do not rupture based on the design changes being made in conjunction with the masonry wall reevaluation for each LaSalle Unit that will prevent the failure of the VR exhaust plenum masonry walls.

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

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

The MSLB accident is previously analyzed but considered only instantaneous closure of installed dampers. The reevaluation and design changes extend the previous accident analysis to assure that structures previously considered unaffected by the MSLB will maintain their structural integrity. The block walls are static and the dampers function in response to an accident, thus the analysis method and design changes are not accident initiators. Therefore the change does not create the possibility of a new [or] different kind of accident from any accident previously evaluated.

The design changes being made in conjunction with the masonry wall reevaluation for each LaSalle Unit that will prevent the failure of the VR exhaust plenum masonry walls are as follows:

(1) Installation of a pressure relief damper,

(2) An excess-flow check damper, and

(3) Required masonry wall support improvements in the reactor building ventilation exhaust plenum for each Unit.

The reevaluation of the masonry walls uses different load factors and load combinations as well as reduced acceptance criteria than previously used for these walls. The change in the evaluation does not cause the rupture or failure of the effected masonry walls, since the evaluation shows the walls remain intact.

The installation of the above design changes, in conjunction with masonry wall analysis assure that the subject masonry walls will not rupture or fail. Therefore, SSCs that would be affected by wall rupture can fulfill their intended function, maintaining the consequences of previously evaluated accident the same.

The new pressure relief damper and excess-flow check damper are safety-related and are analyzed to function under the conditions created by the MSLB. In addition, the dampers and the duct they are installed in have been analyzed to assure no failure will occur during an Operating Basis Earthquake (OBE) or Safe Shutdown Earthquake (SSE).

Based on an analysis of potential failure modes in accordance with ANSI/ANS–58.9–1981, ''Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems,'' Paragraph 4.1, the active function of the pressure relief damper and excess flow check damper are considered exempted from consideration of single failure. The principles governing operation of the dampers are simple and direct and not subject to change or deterioration with time, similar to the function of a code safety relief valve and a swing check valve. With periodic testing of the dampers, continued reliable performance is assured.

The dampers are designed and set so that the pressures created by normal ventilation flow changes do not cycle the dampers, and thus the new dampers do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Administrative controls will be in place prior to implementation of this change to assure the testing and maintenance is periodically performed in accordance with vendor recommendations. These dampers will be included as equipment required to be monitored/maintained, because the function performed by the dampers is within the scope of the Maintenance Rule, 10 CFR 50.65.

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

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

Originally, no masonry walls were evaluated for HELB pressurization effects, because the walls were considered protected by the isolation dampers. However, the original design methodology for masonry did include load combinations including $P_a$:

Abnormal
$1.0D + 1.0L + 1.5P_a$

Abnormal/Severe Environment
$1.0D + 1.0L + 1.25P_a + 1.25E_o$

Abnormal/Extreme Environment
$1.0D + 1.0L + 1.0P_a + 1.0E_{ss}$,

Where D is Dead Load; L is Live Load; $P_a$ is pressurization due to HELB; $E_o$ is Loads generated by the Operating Basis Earthquake (OBE); and $E_{ss}$ is Loads generated by the Safe Shutdown Earthquake (SSE).

The current reevaluation was required due to determination that some block walls in the LaSalle Auxiliary Building are affected by a transient pressurization due to a MSLB. The specific changes from the original analyses involve the following for loads and load combinations.

1. Abnormal:

$1.0D + 1.0L + 1.0P_{HELB}$

2. Abnormal/severe environmental:

$1.0D + 1.0L + [(1.1E_o)^2 + 1.0P_{HELB2}]^{1/2}$

3. Abnormal/extreme environmental:

$1.0D + 1.0L + [1.0E_{ss}^2 + 1.0P_{HELB}^2]^{1/2}$

Where:

(1) $P_{HELB}$ is the short-term differential pressurization load on the VR plenum masonry walls resulting from non-instantaneous opening/closure of the protection dampers.

(2) The Load Factor on pressure due to HELB is 1.0 for all cases.

(3) The Loading Combination of pressure and seismic is the Square Root of the Sum of Squares (SRSS).

LaSalle has selected the proposed load combinations in consideration of the following:

Isolation, check, and relief dampers protect the walls; therefore the pressurization effects are not sustained, but are transient in nature.

The transient pressurization effect ($P_{HELB}$) is derived from a conservative detailed analysis of an instantaneous HELB combined with non-instantaneous damper opening/closure. Due to the precise nature and conservatism of this HELB analysis, there is little uncertainty in $P_{HELB}$ .

Therefore a load factor of 1.0 is used for all abnormal load combinations.

$P_{HELB}$ is a short duration, dynamic load. Accordingly, the seismic and transient HELB pressurization loads are combined using the Square Root of Sum of the Squares (SRSS) method because the peak effects of these dynamic loads are unlikely to occur simultaneously. This combination method is used in the analysis of other components such as component supports.

The proposed load combinations accordingly provide a conservative basis for reassessment of the VR exhaust plenum masonry wall systems.

In regards to the masonry acceptance criteria, the original acceptance criteria used for this condition are the National Concrete Masonry Associations (NCMA) ''Specification for the Design and Construction of Load Bearing Masonry—1979'' allowable stresses times a 1.67 factor. These allowable stresses correspond to stress equal to the modulus of rupture ($f_r$) of the masonry divided by a factor of safety of 3.35. During reviews to address masonry wall issues per NRC IE Bulletin 80–11, six walls did not meet this acceptance criteria. The acceptance criteria used for these walls was for $f_r$ values determined from testing at Clinton Power Station divided by a factor of safety of 2.5. This acceptance criteria was accepted by the NRC for LaSalle in Supplement 5 of NUREG 0519, Safety Evaluation Report related to the Operation of LaSalle County Station, Units 1 and 2. The VR exhaust plenum walls will use the same acceptance criteria for the transient HELB pressurization cases.

The minimum masonry safety factor for the LaSalle Unit 2 walls affected by the HELB loads range from 2.6 to 3.1 with one wall having a safety factor of 4.9.

Masonry wall steel support members were originally designed for this condition elastically to the American Institute of Steel Construction's (AISC) ''Steel Construction Manual—Seventh Edition'' allowable stresses times a 1.6 factor. In the reassessment of these members due to the transient HELB pressurization, elasto-plastic behavior is allowed (with a ductility ratio limit of 10). It is appropriate to consider them similar to high-energy line break systems that will maintain their integrity as they absorb the energy of the incidental pressure excursion.

High-energy line breaks are discussed in Section 3.6 of the UFSAR. The discussion in this section focuses on the design of pipe whip restraints, and in Table 3.6–6 acceptance criteria are provided. This table shows that the energy absorbing portions of the pipe whip restraint are allowed to go plastic, thereby absorbing energy. While Table 3.6–6 of the UFSAR deals with energy absorbing portions of the pipe whip restraints, wide-flange shapes are not addressed. Wide-flange shapes absorb energy through flexural deformations.

Guidance on appropriate acceptance criteria for flexural members is provided in Appendix A to SRP 3.5.3, ''Barrier Design Procedures.'' This appendix indicates that for tension due to flexure in structural steel members, a ductility ratio value not to exceed 10.0 is acceptable. SRP 3.8.4, paragraph III.5 also notes that some localized points on the structure, the allowable stresses specified for ''structural steel'' may be exceeded, provided that integrity of the structure is not affected.

Note that only one of the Unit 2 walls affected by these HELB loads required the use of the elasto-plastic acceptance criteria for two structural steel members.

In summary, these alternate criteria for reassessment of the integrity of the LaSalle Reactor Building Ventilation Exhaust Plenum masonry walls in conjunction with the design changes adding a pressure relief damper, an excess flow check damper and masonry wall support steel changes, assures that the walls will maintain their integrity during a MSLB. The safety factor is reduced; however, the walls have sufficient strength and safety margin to maintain structural integrity and thus perform their intended safety function during the pressurization transient due to a MSLB accident.

Therefore, these changes do not involve a significant reduction in the margin of safety.

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

*Local Public Document Room location:* Jacobs Memorial Library, 815 North Orlando Smith Avenue, Illinois

Valley Community College, Oglesby, Illinois 61348–9692.

*Attorney for licensee:* Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

*NRC Section Chief:* Anthony J. Mendiola.

*Consumers Energy Company, Docket No. 50–155, Big Rock Point Plant, Charlevoix, County, Michigan*

*Date of amendment request:* May 11, 1999 (Accession No. 9905170189).

*Description of amendment request:* The proposed amendment would delete from the Defueled Technical Specifications (DTS) the definition for site boundary and Figure 5.1–1, Big Rock Point Site Map, and revise the description of the Big Rock Point site under subsection 5.1. The amendment also proposes editorial changes associated with the above proposed revisions.

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

In accordance with 10 CFR 50.91, Consumers Energy Company has made a determination that the proposed amendment does not involve significant hazards considerations. Consumers Energy Company has concluded that the proposed amendment will 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 change is administrative in nature and has no [e]ffect on the health and safety of the public. There is no reduction or elimination of federal regulatory requirements associated with the proposed amendment. The information being removed from the Defueled Technical Specifications is unnecessary since Site Boundary is already defined in 10 CFR Part 20, and the site map [Defueled Technical Specification Figure 5.1–1] is already provided in the Updated Final Hazards [Summary] Report. Furthermore, the proposed changes are consistent with the guidance provide in NUREG–1625 ['Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants''].

The proposed change does not:

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

The proposed amendment does not change the site boundary as it currently exists. Deleting the Site Boundary definition and changing the upper case characters to lower case throughout the DTS and the Bases where it appears, and deleting the site figure from the DTS and related references will not increase the probability or consequences of a new or different kind of accident previously evaluated. This proposed change is administrative in nature and does not involve fuel handling or affect or modify any system, structure or component.

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

The proposed amendment does not change the site boundary as it currently exists. Deleting the Site Boundary definition and changing the upper case characters to lower case throughout the DTS and the Bases where it appears, and deleting the site figure from the DTS and related references will not create the possibility of a new or different kind of accident from any accident previously evaluated. This proposed change is administrative in nature and does not involve fuel handling or affect or modify any system, structure or component.

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

The proposed changes do not involve any physical changes to the plant or plant procedures. There will be no reduction in a margin of safety.

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

*Local Public Document Room location:* North Central Michigan College, 1515 Howard Street, Petosky, MI 49770.

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

*NRC Section Chief:* Dr. Michael T. Masnik.

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

*Date of amendment request:* October 2, 1998, supplemented May 13, 1999.

*Description of amendment request:* The proposed amendments would resolve an unreviewed safety question involving use of credit for reactor building overpressure in the licensing basis for the available net positive suction head for the reactor building spray pumps and the low pressure injection pumps. If approved, the appropriate changes would be incorporated in the Oconee Updated Final Safety Analysis Report.

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

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

The reactor building spray (RBS) and low pressure injection (LPI) systems are not considered as initiators of any analyzed event, therefore, this change has no impact on the probability of an event previously analyzed.

The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the set points at which these actions are initiated. The proposed change permits limited reactor building overpressure to be credited in the calculation of available net positive suction head (NPSH) for the RBS and LPI pumps for a limited period of time during the sump recirculation phase. It is supported by calculations which demonstrate that adequate reactor building overpressure will be available to ensure the RBS and LPI systems will be capable of performing their safety functions. Thus, the proposed change does not significantly increase the consequences of an accident previously evaluated.

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

The proposed change permits limited reactor building overpressure to be credited in the calculation of available NPSH for the RBS and LPI pumps for a limited period of time during the sump recirculation phase. It does not involve a physical alteration of the plant. The proposed change is supported by calculations which demonstrate that adequate reactor building overpressure will be available to ensure the RBS and LPI systems will be capable of performing their safety functions. This change will not alter the manner in which the RBS or LPI system is initiated, nor will the function demands on the RBS or LPI system be changed. Thus, the proposed change does not create the possibility of a new or different kind of accident.

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

The proposed change permits limited reactor building overpressure to be credited in the calculation of available NPSH for the RBS and LPI pumps for a limited period of time during the sump recirculation phase. Crediting a slight amount of overpressure does not result in a significant reduction in the margin of safety, because conservative analyses demonstrate that adequate reactor building overpressure will be available to ensure the RBS and LPI systems will be capable of performing their safety functions. Thus, the proposed change does not involve a significant reduction in a margin of safety.

Duke has concluded based on the above information that there are no significant hazards involved in this LAR.

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

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

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

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

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

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

*Description of amendment request:* The proposed amendments would: (a) revise the pressure-temperature (P–T) limits of Technical Specification (TS) 3.4.3 for heatup, cooldown, and inservice test limitations for the Reactor Coolant System to a maximum of 33 Effective Full Power Years; (b) revise TS 3.4.12, Low Pressure Overpressure Protection System (LTOP), to reflect the revised P–T limits of the Unit 1, 2, and 3 reactor vessels; (c) permit operation during LTOP conditions with two reactor coolant pumps in operation in a single loop; and (d) relax the LTOP operating envelope, thereby reducing potential challenges to the reactor coolant system power operated relief valves.

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

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

No.

These proposed Technical Specification (TS) changes were developed utilizing the procedures of ASME XI, Appendix G, in conjunction with Code Cases N–514, N–588 and N–626, as described in the Technical Justification. Usage of these procedures provides compliance with the underlying intent of 10 CFR 50 Appendix G and provide safety limits and margins of safety that ensure failure of a reactor vessel will not occur.

The proposed changes do not impact the capability of the reactor coolant pressure boundary (i.e., no change in operating pressure, materials, seismic loading, etc.) and therefore do not increase the potential for the occurrence of a loss of coolant accident (LOCA). The changes do not modify the reactor coolant system pressure boundary, nor make any physical changes to the facility design, material, or construction standards. The probability of any design basis accident (DBA) is not affected by this change, nor are the consequences of any DBA affected by this change. The proposed Pressure-Temperature (P–T) limits, Low Temperature Overpressure (LTOP) limits and setpoints, and allowable operating reactor coolant pump combinations are not considered to be an initiator or contributor to any accident analysis addressed in the Oconee UFSAR.

The proposed changes do not adversely affect the integrity of the RCS such that its function in the control of radiological consequences is affected. Radiological off-site exposures from normal operation and operational transients, and faults of moderate frequency do not exceed the guidelines of 10 CFR 100. In addition, the proposed changes do not affect any fission product barrier. The revised PORV LTOP setpoint is established to protect reactor coolant pressure boundary. The changes do not degrade or prevent the response of the PORV or safety-related systems to previously evaluated accidents. In addition, the changes do not alter any assumption previously made in the mitigation of the radiological consequences of an accident previously evaluated.

Therefore, the probability or consequences of an accident previously evaluated will not be increased by approval of the requested changes.

B. Create the possibility of a new or different kind of accident from the accident previously evaluated?

No.

The proposed license amendment revises the Oconee reactor vessel P–T limits, LTOP limits and setpoints, and allowable operating reactor coolant pumps combinations. Compliance with 10 CFR 50 Appendix G, includes utilization of ASME XI, Appendix G, as modified by Code Cases N–514, N–588 and N–626 to meet the underlying intent of the regulations.

Operation of Oconee in accordance with these proposed Technical Specifications changes will not create any failure modes not bounded by previously evaluated accidents. Consequently, approval of these changes will not create the possibility of a new or different accident from any accident previously evaluated.

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

No.

The proposed Technical Specification (TS) changes were developed utilizing the procedures of ASME XI, Appendix G, in conjunction with Code Cases N–514, N–588 and N–626, as described in the Technical Justification. Usage of these procedures provides compliance with the underlying intent of 10 CFR 50 Appendix G and provides safety limits and margins of safety which ensure failure of a reactor vessel will not occur.

No plant safety limits, set points, or design parameters are adversely affected. The fuel, fuel cladding, and Reactor Coolant System are not impacted. Therefore, there will be no significant reduction in any margin of safety as a result of approval of the requested changes.

Duke has concluded based on this information there are no significant hazards considerations involved in this amendment request.

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

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

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

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

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

*Date of amendment request:* May 17, 1999.

*Description of amendment request:* The proposed amendment would revise the Technical Specifications associated with the enabling of the Oscillation Power Range Monitor (OPRM) instrumentation reactor protection system (RPS) trip function. The OPRM is designed to detect the onset of reactor core power oscillations resulting from thermal-hydraulic instability and suppresses them by initiating a reactor scram via the RPS trip logic.

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

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

The proposed change specifies limiting conditions for operations, required actions and surveillance requirements of the OPRM system and allows operation in regions of the power to flow map currently restricted by the requirements of Interim Corrective Actions (ICAs) and certain limiting conditions of operation of Technical Specifications (TS) 3.4.1. The OPRM system can automatically detect and suppress conditions necessary for thermal-hydraulic (T–H) instability. A T–H instability event has the potential to challenge the Minimum Critical Power (MCPR) safety limit. The restrictions of the ICAs and TS 3.4.1 were imposed to ensure adequate capability to detect and suppress conditions consistent with the onset of T–H oscillations that may develop into a T–H instability event. With the installation of the OPRM System, these restrictions are no longer required.

The probability of a T–H instability event is most significantly impacted by power to flow conditions such that only during operation inside specific regions of the power to flow map, in combination with power shape and inlet enthalpy conditions, can the occurrence of an instability event be postulated to occur. Operation in these regions may increase the probability that operation with conditions necessary for a T–H instability can occur.

However, when the OPRM is operable with operating limits as specified in the COLR [Core Operating Limits Report], the OPRM can automatically detect the imminent onset of local power oscillations and generate a trip signal. Actuation of an RPS trip will suppress conditions necessary for T–H instability and decrease the probability of a T–H instability event. In the event the trip capability of the OPRM is not maintained, the proposed change includes actions which limit the period of time before the effected OPRM channel (or RPS system) must be placed in the trip condition. If these actions would result in a trip function, an alternate method to detect and suppress thermal hydraulic oscillations is required. In either case the duration of this period of time is limited such that the increase in the probability of a T–H instability event is not significant. Therefore the proposed change does not result in a significant increase in the probability of an accident previously evaluated.

An unmitigated T–H instability event is postulated to cause a violation of the MCPR safety limit. The proposed change ensures mitigation of T–H instability events prior to challenging the MCPR safety limit if initiated from anticipated conditions by detection of the onset of oscillations and actuation of an RPS trip signal. The OPRM also provides the capability of an RPS trip being generated for T–H instability events initiated from unanticipated but postulated conditions. These mitigating capabilities of the OPRM system would become available as a result of the proposed change and have the potential to reduce the consequences of anticipated and postulated T–H instability events. Therefore, the proposed change does not significantly increase the consequences of an accident previously evaluated.

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

The proposed change specifies limiting conditions for operations, required actions and surveillance requirements of the OPRM system and allows operation in regions of the power to flow map currently restricted by the requirements of ICAs and TS 3.4.1. The OPRM system uses input signals shared with APRM [Average Power Range Monitor] and rod block functions to monitor core conditions and generate an RPS trip when required. Quality requirements for software design, testing, implementation and module self-testing of the OPRM system provide assurance that no new equipment malfunctions due to software errors are created. The design of the OPRM system also ensures that neither operation nor malfunction of the OPRM system will adversely impact the operation of other systems and no accident or equipment malfunction of these other systems could cause the OPRM system to malfunction or cause a different kind of accident. Therefore, operation with the OPRM system does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Operation in regions currently restricted by the requirements of ICAs and TS 3.4.1 is within the nominal operating domain and ranges of plant systems and components for which postulated equipment and accidents have been evaluated. Therefore operation within these regions does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change which specifies limiting conditions for operations, required actions and surveillance requirements of the OPRM system and allows operation in certain regions of the power to flow [map] does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change specifies limiting conditions for operations, required actions and surveillance requirements of the OPRM system and allows operation in regions of the power to flow map currently restricted by the requirements of ICAs and TS 3.4.1.

The OPRM system monitors small groups of LPRM signals for indication of local variations of core power consistent with T-H oscillations and generates an RPS trip when conditions consistent with the onset of oscillations are detected. An unmitigated T-H instability event has the potential to result in a challenge to the MCPR safety limit. The OPRM system provides the capability to automatically detect and suppress conditions which might result in a T-H instability event and thereby maintains the margin of safety by providing automatic protection for the MCPR safety limit while significantly reducing the burden on the control room operators. In the event the trip capability of the OPRM is not maintained, the proposed change includes actions which limit the period of time before the effected OPRM channel (or RPS system) must be placed in the trip condition. If these actions would result in a trip function, an alternate method to detect and suppress thermal hydraulic oscillations is required. Since, in either case, the duration of this period of time is limited so that the increase in the probability of a T-H instability event is not significant. Operation with the OPRM system does not involve a significant reduction in a margin of safety.

Operation in regions currently restricted by the requirements of ICAs and TS 3.4.1 is within the nominal operating domain assumed for identifying the range of initial conditions considered in the analysis of anticipated operational occurrences and postulated accidents. Therefore, operation in these regions does not involve a significant reduction in the margin of safety.

The proposed change, which specifies limiting conditions for operations, required actions and surveillance requirements of the OPRM system and allows operation in certain regions of the power to flow map, does not involve a significant reduction in a margin of safety.

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

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

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

*NRC Section Chief:* James W. Clifford.

*South Carolina Electric & Gas Company (SCE&G), South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station (VCSNS), Unit No. 1, Fairfield County, South Carolina*

*Date of amendment request:* May 17, 1999.

*Description of amendment request:* The proposed amendment would change VCSNS Technical Specification 3.7.1.3 ''Condensate Storage Tank—Limiting Conditions for Operation'' to revise the tank minimum contained water volume from 172,000 gallons to 179,850 gallons.

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

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

FSAR [Final Safety Analysis Report] 10.4.9.1 states that minimum required usable volume for the Condensate Storage Tank (CST) is 158,570 gallons based on maintaining the plant at HOT STANDBY conditions for eleven hours. This volume has already been adjusted for both plant uprate conditions and replacement steam generator requirements. This change to LCO [Limiting Condition for Operation] 3.7.1.3 will ensure that 160,054 gallons is maintained in the CST, being available and dedicated to the Emergency Feedwater (EFW) System. Thus, this change will ensure that the EFW System has an adequate water supply to perform its design basis function in regard to maintaining the plant in HOT STANDBY condition.

2. This request does not create the possibility of a new or different kind of accident from any accident previously evaluated.

This change increases the minimum required volume of water in the CST, thus ensuring that the EFW System can perform its required safety function. The maximum and normal water levels in the CST are not being changed. Therefore, no new failure modes of the CST, or flooding concerns are created.

3. This request does not involve a significant reduction in a margin to safety[.]

This change does not reduce any margin associated with the CST inventory available to the EFW. In fact, a small gain in margin (less than 1%) is realized by specifying the minimum required volume based on the maximum volume available due to nozzle locations and other physical characteristics of the tank instead of the minimum required to maintain HOT STANDBY for 11 hours. Additionally, the requirement for sufficient CST volume to maintain HOT STANDBY for 11 hours is still met and the Service Water System still provides the long term supply of safety grade cooling water to the EFW System. The Service Water supply is not affected by this change, and thus the margin for safety grade cooling water to the EFW System (or safety grade cooling of the RCS) is not affected.

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

*Local Public Document Room location:* Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180.

*Attorney for licensee:* Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

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

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

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

*Description of amendment request:* The proposed amendment would add an additional condition to the Farley Nuclear Plant (FNP), Unit 1 license. This condition would allow cycle 16 operation based on a risk-informed approach to evaluate steam generator tube structural integrity.

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

The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated in the FSAR [Final Safety Analysis Report]. The probability of tube burst is slightly increased as a result of this proposed amendment but is within current industry guidance. Therefore, the probability of a previously evaluated accident are not significantly increased. There is no change in the FNP design basis as a result of this change and, as a result, this change does not involve a significant increase in the consequences of an accident previously evaluated.

The proposed changes to the TSs [technical specifications] do not increase the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new limiting single failure or accident scenario has been created or identified due to the proposed changes. Safety-related systems will continue to perform as designed. The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes do not involve a significant reduction in the margin of safety. There is no impact in the accident analyses. These proposed changes are technically consistent with the requirements of NEI [Nuclear Energy Institute] 97–06, ''Steam Generator Program Guidelines,'' Draft Regulatory Guide DG 1074, ''Steam Generator Tube Integrity,'' and Regulatory Guide (RG) 1.174, ''An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.'' Thus the proposed changes do not involve a significant reduction in the margin of safety.

Accordingly, SNC [Southern Nuclear Operating Company] has determined that the proposed amendment to the Facility Operating License NPF–2 does not involve a significant hazards consideration.

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

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

*Attorney for licensee:* M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama.

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

*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 3, 1999.

*Description of amendment request:* The proposed changes will modify the Technical Specifications to ensure the emergency ventilation system is maintained operable consistent with the assumptions in the radiological dose consequence reanalysis from a Large Break Loss-of-Coolant Accident and to clearly identify that the ventilation system is a shared system between the two units.

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

1. There is no significant change in the probability or consequences of an accident previously evaluated. There are no system changes which would increase the probability of occurrence of an accident. The dose consequences of the accidents have been reviewed, and in some cases the doses at the EAB [exclusion area boundary] * * * and the doses to the control room personnel were found to increase. However, this increase is not significant because the revised doses remain below the limits of 10 CFR 100 and below the limits of GDC [General Design Criterion]—19 of Appendix A of 10 CFR 50.

2. No new accident types or equipment malfunction scenarios have been introduced. Therefore, the possibility of an accident of a different type than any evaluated previously in the UFSAR [Updated Final Safety Analysis Report] is not created.

3. There is no significant reduction in the margin of safety, as the revised dose calculations for all accidents continue to meet the appropriate GDC–19 limits.

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

*Local Public Document Room location:* The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

*Attorney for licensee:* Mr. Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

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

*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 6, 1999.

*Description of amendment request:* The proposed changes will modify the Technical Specifications, revising the surveillance frequency for the Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) analog instrumentation channels and also revising the allowed outage time and action times for the RTS and ESFAS analog instrumentation channels and the actuation logic.

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

Virginia Electric and Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) Technical Specification changes for the North Anna Units 1 and 2 and determined that a significant hazards consideration is not involved. In support of this conclusion, the following evaluation is provided.

*Criterion 1*—Operation of North Anna Units 1 and 2 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The determination that the results of the proposed changes remain within acceptable criteria was established in the SER(s) [Safety Evaluation Reports] prepared for WCAP–10271, WCAP–10271 Supplement 1, WCAP–10271 Supplement 2, WCAP–10271 Supplement 2, Revision 1 and WCAP–14333 issued by letters dated February 21, 1985, February 22, 1989, April 30, 1998, and July 15, 1998.

Implementation of the proposed changes is expected to result in an increase in total RTS and ESFAS yearly unavailability. The proposed changes have been shown to result in a small increase in the core damage frequency (CDF) due to the combined effects of increased RTS and ESFAS unavailability and reduced inadvertent reactor trips.

The values determined by the WOG [Westinghouse Owners Group] and presented in the WCAP for the increase in CDF were verified by Brookhaven National Laboratory (BNL) as part of an audit and sensitivity analyses for the NRC [Nuclear Regulatory Commission] Staff. Based on the small value of the increase compared to the range of uncertainty in the CDF, the increase is considered acceptable. The analysis performed by the WOG and presented in the WCAP included changes to the surveillance frequencies for the automatic actuation logic and actuation relays and the reactor trip and bypass breakers. The overall increase in the CDF, including the changes to the surveillance frequencies for the automatic actuation logic and actuation relays and the reactor trip and bypass breakers, was approximately 6 percent. However, even with this increase, the overall CDF remains lower than the NRC safety goal of $10^{-4}$/reactor year.

Changes to surveillance test frequencies for the RTS and ESFAS interlocks do not represent a significant reduction in testing. The currently specified test interval for interlock channels allows the surveillance requirement to be satisfied by verifying that the permissive logic is in its required state using the annunciator status light. The surveillance as currently required only verifies the status of the permissive logic and does not address verification of channel setpoint or operability. The setpoint verification and channel operability is verified after a refueling shutdown. The definition of the channel check includes comparison of the channel status with other channels for the same parameter. The requirement to routinely verify permissive status is a different consideration than the availability of trip or actuation channels which are required to change state on the occurrence of an event and for which the function availability is more dependent on the surveillance interval. Therefore, the change in the interlock surveillance requirement to at least once every 18 months does not represent a significant change in channel surveillance and does not involve a significant increase in unavailability of the RTS and ESFAS.

For the additional relaxations in WCAP–14333, the WOG evaluated the impact of the additional relaxation of allowed outage times and completion times, and action statements on core damage frequency. The change in core damage frequency is 3.1 percent for those plants with two out of three logic schemes that have not implemented the proposed surveillance test interval, allowed outage times, and completion times evaluated in WCAP–10271 and its supplements. This analysis calculates a significantly lower increase in core damage frequency than the WCAP–10271 analysis calculated. This can be attributed to more realistic maintenance intervals used in the current analysis and crediting the AMSAC [ATWS (anticipated transient without scram) mitigating system actuation circuitry] system as an alternative method of initiating the auxiliary feedwater pumps. Therefore, the overall increase in CDF is estimated to be 3.1% for the proposed changes per the generic Westinghouse analysis.

The NRC performed an independent evaluation of the impact on core damage frequency (CDF) and large early release fraction (LERF). The results of the staff's review indicate that the increase in core damage frequency is small (approximately 3.2%) and the large early release fraction would increase by only 4 percent for 2 out of 3 logic schemes that have not implemented the proposed surveillance test interval, allowed outage times, and completion times evaluated in WCAP–10271 and its supplements. Further, the absolute values for CDF still remain within NRC safety goals.

Therefore, the proposed changes do not result in a significant increase in the severity or consequences of an accident previously evaluated. Implementation of the proposed changes affects the probability of failure of the RTS and ESFAS but does not alter the manner in which protection is afforded or the manner in which limiting criteria are established.

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

The proposed changes do not result in a change in the manner in which the RTS or ESFAS provide plant protection. No change is being made which alters the functioning of the RTS or ESFAS (other than in a test mode). Rather the likelihood or probability of the RTS or ESFAS functioning properly is affected as described above. Therefore, the proposed changes do not create the possibility of a new or different kind of accident as defined in the Safety Analysis Report.

The proposed changes do not involve hardware changes. 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 [Institute of Electrical and Electronics Engineers] Standards. Therefore, testing in bypass has been previously approved and implementation of the proposed changes for testing in bypass does not create the possibility of a new or different kind of accident from any previously evaluated. Furthermore since the other proposed changes do not alter the physical operation or functioning of the RTS or ESFAS the possibility of a new or different kind of accident from any previously evaluated has not been created.

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

The proposed changes do not alter the safety limits, limiting safety system setpoints or limiting conditions for operation. The RTS and ESFAS analog instrumentation remain operable to mitigate as assumed in the accident analysis. 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.

Implementation of the proposed changes is expected to result in an overall improvement in safety by less frequent testing of the RTS and ESFAS analog instruments will result in less inadvertent reactor trips and actuation of Engineered Safety Features components.

This analysis demonstrates that the proposed amendment to The North Anna Unit 1 and 2 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.

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

*Local Public Document Room location:* The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

*Attorney for licensee:* Mr. Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

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

**Notice of Issuance of Amendments to Facility Operating Licenses**

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

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

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

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

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

*Date of application for amendments:* March 22, 1999.

*Brief description of amendments:* The amendments modify the technical specifications to permit the use of the Gamma-Metrics Post Accident Neutron Monitors source range neutron flux detectors in addition to the Westinghouse source range neutron flux monitors to satisfy the requirement that two source range neutron flux monitors be operable during Mode 6 operations (refueling).

*Date of issuance:* June 2, 1999.

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

*Amendment Nos.:* 109 & 109, 102 & 102.

*Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77:* The amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: March 29, 1999 (64 FR 14944). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 2, 1999.

No significant hazards consideration comments received: No.

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

*Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois*

*Date of application for amendments:* December 2, 1996, as supplemented on May 27, 1999.

*Brief description of amendments:* The amendments revised Technical Specification 3/4.4.2 to reduce the number of required Safety/Relief valves (SRVs). This change supports a modification to remove five of the currently installed SRVs due to excess capacity and to reduce the amount of valve maintenance and associated worker radiation dose. The revised TS requires that 12 of the remaining installed 13 SRVs be operable.

*Date of issuance:* June 3, 1999.

*Effective date:* Immediately, to be implemented prior to startup of L1C10 for Unit 1 and prior to startup of L2C9 for Unit 2.

*Amendment Nos.:* 133 & 118.

*Facility Operating License Nos. NPF–11 and NPF–18:* The amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: January 29, 1997 (62 FR 4343). The May 27, 1999, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 3, 1999.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* Jacobs Memorial Library, 815 North Orlando Smith Avenue, Illinois Valley Community College, Oglesby, Illinois 61348–9692.

*Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan*

*Date of application for amendment:* March 23, 1999 (NRC–99–0025).

Brief description of amendment: The amendment revises Technical Specification Surveillance Requirement (SR) 4.4.1.1.1 to require each recirculation pump discharge valve be demonstrated operable at least once every 18 months, deletes the "*" footnote from the SR, and revises the footnote itself to read "Not used."

*Date of issuance:* May 25, 1999.

*Effective date:* May 25, 1999, with full implementation within 90 days.

*Amendment No.:* 133.

*Facility Operating License No. NPF–43:* Amendment revises the Technical Specifications.

*Date of initial notice in* **Federal Register**: April 21, 1999 (64 FR 19555)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 25, 1999.

*No significant hazards consideration comments received:* No.

*Local Public Document Room location:* Monroe County Library System, Ellis Reference and Information Center, 3700 South Custer Road, Monroe, Michigan 48161.

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

*Date of application for amendments:* July 9, 1998, as supplemented March 31, 1999.

*Brief description of amendments:* These amendments revised Technical Specification (TS) 3/4.7.1.1 and associated Bases for both units. This amendment specifies maximum allowable reactor power level based on the number of operable main steam safety valves (MSSVs) rather than requiring reduction in reactor trip setpoint. This change is consistent with the Nuclear Regulatory Commission's improved Standard Technical Specifications for Westinghouse plants (NUREG–1431, Revision 1). The maximum allowable reactor power level with inoperable MSSVs will be calculated based on the recommendations of Westinghouse Nuclear Safety Advisory Letter 94–01. The change to the Unit 1 TS 3.7.1.1 also deletes reference to 2 loop operation since 2 loop operation is not a licensed condition for either unit. Unit 1 TS Table 3.7–3 is then renumbered to be Table 3.7–2.

The March, 31, 1999 letter withdrew a portion of the amendment which would have removed the values of the orifice diameter of each MSSV from the TSs. This information will be maintained in the TSs.

*Date of issuance:* June 3, 1999.

*Effective date:* Units 1 and 2 as of date of issuance and shall be implemented within 60 days.

*Amendment Nos.:* 223 and 99.

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

*Date of initial notice in* **Federal Register**: August 12, 1998 (63 FR 43203). The March 31, 1999 letter did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 3, 1999.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* August 31, 1998.

*Brief description of amendment:* Changes the Crystal River Unit 3 Technical Specifications to add additional instrumentation variables to Improved Technical Specification Table 3.3.17–1, Post-Accident Monitoring Instrumentation.

*Date of issuance:* June 3, 1999.

*Effective date:* As of date of issuance, to be implemented prior to commencing cycle 12 operation.

*Amendment No.:* 177.

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

*Date of initial notice in* **Federal Register**: October 21, 1998 (63 FR 56250).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 3, 1999.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* November 23, 1998, as supplemented January 29 and May 7, 1999.

*Brief description of amendment:* The amendment changes the Improved Technical Specifications for several reactor protection system and engineered safeguards actuation system setpoint values, and changes the surveillance requirement to verify valve position for valves in the high pressure injection system flowpath.

*Date of issuance:* May 21, 1999.

*Effective date:* As of date of issuance, to be implemented prior to commencing Cycle 12 operation.

*Amendment No.:* 178.

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

*Date of initial notice in* **Federal Register**: December 30, 1998 (63 FR 71966). The supplemental letters dated January 29 and May 7, 1999, did not change the original proposed no significant hazards consideration determination, or expand the scope of the amendment request as originally noticed.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 21, 1999.

No significant hazards consideration comments received: No.

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

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

*Date of amendment request:* December 16, 1998.

*Description of amendment request:* These amendments consist of changes to the Technical Specifications (TS) in response to Florida Power & Light's (FPL) application dated December 16, 1998, regarding facility staff qualifications for multi-discipline supervisor (MDS) positions at Lucie Units 1 and 2. The amendments revise the administrative controls in TS Section 6.3, ''Unit Staff Qualifications,'' by modifying FPL's commitment to ANSI/ANS 3.1–1978, ''Selection and Training of Nuclear Power Plant Personnel,'' to incorporate specific staff qualifications for the position of MDS.

*Date of Issuance:* May 25, 1999.

*Effective Date:* May 25, 1999.

*Amendment Nos.:* 161 and 102.

*Facility Operating License Nos. DPR–67 and NPF–16:* Amendments revised the TS.

*Date of Initial Notice in* **Federal Register**: February 10, 1999 (64 FR 6698).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 25, 1999.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* November 5, 1998, as supplemented February 18, 1999.

*Brief description of amendment:* The amendment modifies the safety limits and surveillances of the LPRM and APRM systems and related Bases pages to ensure the APRM channels respond within the necessary range and accuracy and to verify channel operability. In addition, an unrelated change to the Bases of Specification 2.3 is included to clarify some ambiguous language.

*Date of Issuance:* June 2, 1999.

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

*Amendment No.:* 208.

*Facility Operating License No. DPR–16.* Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: December 16, 1998 (63 FR 69342). The February 18, 1999, supplemental letter provided clarifying information, was within the scope of the original application, and did not change the staff's original no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated June 2, 1999.

No significant hazards consideration comments received: No.

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

*Northeast Nuclear Energy Company, et al., Docket Nos. 50–245, 50–336, and 50–423, Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, New London County, Connecticut*

*Date of application for amendment:* December 22, 1998, as supplemented March 19, 1999.

*Brief description of amendment:* The amendment replaces specific titles in Section 6.0 of the Technical

Specifications of all three Millstone units with generic titles.

*Date of issuance:* June 3, 1999.

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

*Amendment No.:* 105, 235, and 171.

*Facility Operating License Nos. DPR–21, DPR–65, and NPF–49:* Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: January 27, 1999 (64 FR 4158). The March 19, 1999 letter provided clarifying information that did not change the scope of the December 22, 1998, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 3, 1999.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendments:* April 20, 1999.

*Brief description of amendments:* The amendments revised the implementation date for the relocation of the requirements specified in Technical Specification Sections 3.1.E and 5.1 to the Updated Final Safety Analyis Report. On December 7, 1998, the NRC had previously issued license amendments 141 and 132 for Units 1 and 2, respectively, approving the relocation of aforementioned requirements by June 1, 1999. The proposed amendments would postpone the implementation date to September 1, 1999.

*Date of issuance:* June 2, 1999.

*Effective date:* June 2, 1999, with full implementation within 30 days .

*Amendment Nos.:* 145 and 136.

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

*Date of initial notice in* **Federal Register**: April 29, 1999 (64 FR 23131) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 2, 1999.

No significant hazards consideration comments received: No.

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

*PECO Energy Company, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania*

*Date of application for amendments:* January 4, 1999.

*Brief description of amendments:* These amendments revise the administrative section of the Technical Specification pertaining to controlled access to high radiation areas, and the reporting dates for the annual occupational radiation exposure report and the annual radioactive effluent release report.

*Date of issuance:* May 24, 1999.

*Effective date:* Units 1 and 2, as of date of issuance and shall be implemented within 30 days.

*Amendment Nos.:* 135 and 100.

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

*Date of initial notice in* **Federal Register**: February 10, 1999 (64 FR 6706) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 24, 1999.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* January 25, 1999.

*Brief description of amendment:* The amendment changes the Technical Specifications (TSs) by relocating certain requirements from the TSs to the Final Safety Analysis Report.

*Date of issuance:* May 24, 1999.

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

*Amendment No.:* 189.

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

*Date of initial notice in* **Federal Register**: April 21, 1999 (64 FR 19562).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 24, 1999.

No significant hazards consideration comments received: No.

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

*PP&L, Inc., Docket No. 50–387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania*

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

*Brief description of amendment:* This amendment would change the allowable values for both the core spray system and the low pressure coolant injection system reactor steam dome pressure-low functions.

*Date of issuance:* May 25, 1999.

*Effective date:* As of date of issuance, and shall be implemented within 30 days after startup from the Unit 1 eleventh refueling and inspection outage currently scheduled for spring 2000.

*Amendment No.:* 181.

*Facility Operating License No. NPF–14:* This amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: April 7, 1999 (64 FR 17028).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 25, 1999.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* October 27, 1998, as supplemented by letters in 1999 dated January 11, January 29, February 25, and April 7 (two letters), and May 17.

*Brief description of amendment:* The amendment revised Technical Specification 4.4.5.4, Table 4.4–3 and the associated Bases to allow the repair of the steam generator tubes with the Electrosleeve tube repair method.

*Date of issuance:* May 21, 1999.

*Effective date:* May 21, 1999, to be implemented within 30 days from the date of issuance. The amendment includes a two cycle operating limit that requires all steam generator tubes repaired with Electrosleeves to be removed from service at the end of two operating cycles following installation of the first Electrosleeve in the steam generators.

*Amendment No.:* 132.

*Facility Operating License No. NPF–30:* The amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: December 2, 1998 (63 FR 66604). The supplemental letters in 1999 dated January 11, January 29, February 25, and April 7 (two letters) provided additional clarifying information that did not expand the staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 21, 1999.

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 9th day of June 1999.

For the Nuclear Regulatory Commission.

**John A. Zwolinski,**

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

[FR Doc. 99–15098 Filed 6–15–99; 8:45 am]

BILLING CODE 7590–01–P

## OFFICE OF PERSONNEL MANAGEMENT

### The National Partnership Council; Notice of Meeting

**AGENCY:** Office of Personnel Management.

**ACTION:** Notice of meeting.

**TIME AND DATE:** 1:30 p.m., June 16, 1999.

**PLACE:** OPM Conference Center, Room 1350, U.S. Office of Personnel Management, Theodore Roosevelt Building, 1900 E Street, NW., Washington, DC. The conference center is located on the first floor.

**STATUS:** This meeting will be open to the public. Seating will be available on a first-come, first-served basis. Individuals with special access needs wishing to attend should contact OPM at the number shown below to obtain appropriate accommodations.

**MATTERS TO BE CONSIDERED:** The National Partnership Council will receive its first Interim Report and hear from Dr. Marick Masters, Research Director for the NPC Research Project, on the status and progress of the Project. The Council will also hear a review of its May skills-building conference and a status report on the John N. Sturdivant National Partnership Awards process.

**CONTACT PERSON FOR MORE INFORMATION:** Jeff Sumberg, Director, Center for Partnership and Labor-Management Relations, Office of Personnel Management, Theodore Roosevelt Building, 1900 E Street, NW., Room 7H28, Washington, DC 20415–2000, (202) 606–2930.

Office of Personnel Management.

**Janice R. Lachance,**

*Director.*

[FR Doc. 99–15250 Filed 6–15–99; 8:45 am]

BILLING CODE 6325–01–P

## POSTAL SERVICE BOARD OF GOVERNORS

### Sunshine Act Meeting

### Board Votes To Close June 20–22, 1999, Meeting

At its meeting on June 7, 1999, the Board of Governors of the United States Postal Service voted unanimously to close to public observation its meeting scheduled for June 20-22, 1999, in Potomac, Maryland.

**MATTER TO BE CONSIDERED:** 1. Strategic Planning.

**PERSONS EXPECTED TO ATTEND:** Governors, Ballard, Daniels, del Junco, Dyhrkopp, Fineman, McWherter, Rider and Winters; Postmaster General Henderson, Deputy Postmaster General Coughlin, Secretary to the Board Koerber, and General Counsel Elcano.

**GENERAL COUNSEL CERTIFICATION:** The General Counsel of the United States Postal Service has certified that the meeting may be closed under the Government in the Sunshine Act.

**CONTACT PERSON FOR MORE INFORMATION:** Requests for information about the meeting should be addressed to the Secretary of the Board Thomas J. Koerber, at (202) 268–4800.

**Thomas J. Koerber,**

*Secretary.*

[FR Doc. 99–15434 Filed 6–14–99; 2:40 pm]

BILLING CODE 7710–12–M

## SECURITIES AND EXCHANGE COMMISSION

**[Rel. No. IC–23865; 812–11268]**

### Global TeleSystems Group, Inc.; Notice of Application

June 9, 1999.

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

**ACTION:** Notice of application for exemption under section 3(b)(2) of the Investment Company Act of 1940 (the ''Act'').

**SUMMARY OF APPLICATION:** Global TeleSystems Group, Inc. (''GTS'') requests an order under section 3(b)(2) of the Act declaring that it is engaged primarily in a business other than that of investing, reinvesting, owning, holding, or trading in securities.

*Filing Dates:* The application was filed on August 24, 1998. Applicant has agreed to file an amendment during the notice period, the substance of which is reflected in this notice.

*Hearing or Notification of Hearing:* An order granting the requested relief will be issued unless the SEC orders a hearing. Interested persons may request a hearing by writing to the SEC's Secretary and serving applicants with a copy of the request, personally or by mail. Hearing requests should be received by the SEC by 5:30 p.m. on July 6, 1999, and should be accompanied by proof of service on applicant, in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons who wish to be notified of a hearing may request notification by writing to the SEC's Secretary.

**ADDRESSES:** Secretary, SEC, 450 5th Street, NW, Washington, DC 20549–0609. Global TeleSystems Group, Inc., 1751 Pinnacle Drive, North Tower 12th Floor McLean, Virginia 22102.

**FOR FURTHER INFORMATION CONTACT:** J. Amanda Machen, Senior Counsel, (202) 942–7120, or Nadya B. Roytblat, Assistant Director, (202) 942–0564 (Office of Investment Company Regulation, Division of Investment Management).

**SUPPLEMENTARY INFORMATION:** The following is a summary of the application. The complete application may be obtained for a fee at the SEC's Public Reference Branch, 450 5th Street, NW, Washington, DC 20549–0102 (tel. 202–942–8090).

### Applicant's Representations

1. GTS, a Delaware corporation, provides telecommunications services to businesses, other telecommunications service providers, and consumers. Through its wholly- and majority-owned subsidiaries (together with GTS, the ''GTS Group''), GTS operates voice and data networks, international gateways, local access and cellular networks, and various value-added services in Western Europe, Central Europe, and the Commonwealth of Independent States, primarily Russia.

2. GTS's management has extensive experience in the development and operation of telecommunications businesses outside the United States. GTS actively participates in the operations and management of its subsidiaries by providing most of the funding for the subsidiaries' operations, selecting key members of the local management team, developing business