ESF Ventilation System	Penetration	RH
[ ]	[ ]	[ ]

Note: The use of any standard other than ASTM D3803–1989 to test the charcoal sample may result in an overestimation of the capability of the charcoal to adsorb radioiodine. As a result, the ability of the charcoal filters to perform in a manner consistent with the licensing basis for the facility is indeterminate.

ASTM D3803–1989 is a more stringent testing standard because it does not differentiate between used and new charcoal, it has a longer equilibration period performed at a temperature of 30°C [86°F] and a relative humidity (RH) of 95% (or 70% RH with humidity control), and it has more stringent tolerances that improve repeatability of the test

$$\text{Allowable Penetration} = \frac{[100\% - \text{Methyl Iodide Efficiency For Charcoal Credited in SER}]}{\text{Safety Factor}}$$

When ASTM D3803–1989 is used with 30°C [86°F] and 95% RH (or 70% RH with humidity control) is used, the staff will accept the following:

Safety factor ≥ 2 for systems with or without humidity control.]

For Plants With Older Technical Specifications

Each engineered safety feature (ESF) ventilation system shall be demonstrated OPERABLE:

a. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

(1) Verifying, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than [see note in preceding section titled ''For Plants With Improved Standard Technical Specifications'']% when tested in accordance with ASTM D3803–1989 at a temperature of ≤ 30°C [86°F] and greater than or equal to a relative humidity of [see note in preceding section titled ''For Plants With Improved Standard Technical Specifications'']%.

b. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than [see note in preceding section titled ''For Plants With Improved Standard Technical Specifications'']% when tested in accordance with ASTM D3803–1989 at a temperature of ≤ 30°C [86°F] and greater than or equal to a relative humidity of [see note in preceding section titled ''For Plants With Improved Standard Technical Specifications'']%.

Dated at Rockville, Maryland, this 19th day of February 1998.

For the Nuclear Regulatory Commission.

Jack W. Roe,

Acting Director, Division of Reactor Program Management, Office of Nuclear Reactor Regulation.

[FR Doc. 98–4761 Filed 2–24–98; 8:45 am]

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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 February 2, 1998, through February 12, 1998. The last biweekly notice was published on February 11, 1998 (63 FR 6968).

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 March 27, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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 amendment request: October 4, 1996, as supplemented by letters dated June 6, September 19, November 7, and December 16, 1997.

Description of amendment request: The proposed amendment for each unit identified above would change the distance criterion in Action b to Limiting Condition for Operation (LCO) 3/4.1.3, ''Movable Control Assemblies,'' by which more than one full-length or part-length control element assembly (CEA) is misaligned from any other CEA in its group. Action b states, in part, that if the misalignment is greater than the specified distance criterion, the reactor core is to be placed in at least hot standby within 6 hours. The proposed amendment would reduce the distance criterion from 19 inches to 9.9 inches, and replace hot standby in 6 hours by ''open the reactor trip breakers.''

This proposed amendment is included as a ''more restrictive'' change in the conversion of the current Technical Specifications (CTS) to the Improved Technical Specifications, which was noticed in the **Federal Register** (62 FR 18153) on April 14, 1997. The proposed amendment would be included in Action F to LCO 3.1.5, ''Movable Control Assemblies,'' of the Improved Technical Specifications. This proposed amendment is a change to the current Technical Specifications and is in addition to the six proposed changes to the CTS or proposed deviations to the Improved Standard Technical Specifications (NUREG–1432) which were identified in the notice of April 14, 1997.

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

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

The proposed changes provide more stringent requirements than previously existed in the CTS. The more stringent requirements will not result in operation that will increase the probability of initiating an analyzed event. If anything, the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes discussed in the specific Discussion of Changes [for specification 3.1.5]. These changes will not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements will not alter the operation and will continue to ensure process variables, structures, systems, or components are maintained consistent with safety analyses and licensing basis [for the plant]. These changes have been reviewed to ensure that no previously evaluated accident has been adversely affected. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident evaluated.

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

Making existing requirements more restrictive and adding more restrictive requirements to the CTS will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. These changes do impose different requirements. However, they are consistent with the assumptions made in the safety analyses, licensing basis, and NUREG–1432 [for the plant]. Therefore, these changes will 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 changes provide more stringent requirements than previously existed in the CTS. An evaluation of these changes concluded that adding these more restrictive requirements either increases or has no impact on the margin of safety. The changes provide additional restrictions which may enhance plant safety. These changes maintain requirements of the safety analysis, licensing basis, and NUREG–1432 [for the plant]. As such, no question of safety is involved. Therefore, these changes will not involve a significant reduction in a margin of safety.

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

Local Public Document Room Location: 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 Project Director: William H. Bateman.

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

Date of amendment request: September 19, 1997.

Description of amendment request: The proposed amendment would relocate the Radioactive Effluent Technical Specifications (RETS) and the Radiological Environmental Monitoring Program to the Offsite Dose Calculation Manual (ODCM), in accordance with the recommendations of Generic Letter 89–01 and NUREG–1433. In addition, changes to other sections of the TSs are being proposed to align the current TSs with NUREG–1433.

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

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

Operation of PNPS in accordance with the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated because of the following:

Definitions

Definitions perform a supporting function for other sections of the TS. The proposed change to incorporate the definition for the Offsite Dose Calculations Manual (ODCM) into Section 5.0, ''Programs and Manuals'', subsection 5.5.1 of the proposed TS will carry forward the requirements contained in the DEFINITION, with minor editorial rewording to be consistent with NUREG 1433, and result in no technical changes. Since the requirements will remain, the impact on initiators of analyzed events or the assumptions assumed in the mitigation of accidents or transient events will not change. Editorial rewording (either adding or deleting) and reformatting is proposed to provide clarity and does not change any technical requirements.

The definitions being proposed for relocation do not impact reactor operation, identify a parameter which is an initial condition assumption for a DBA or transient, identify a significant abnormal degradation of the reactor coolant pressure boundary, and do not provide any mitigation of a design basis event.

RAD Effluents

All editorial rewording (either adding or deleting) and renumbering is made to restructure the section accounting for the requirements relocated in accordance with Generic Letter 89–01. During the editorial rewording and renumbering of the Improved Technical Specifications, no technical changes (either actual or interpretational) to the TS were made unless they were identified and justified.

Adding a note to clearly indicate that the first sample for noble gas activity is not required for 31 days after SJAE is placed in operation has always been considered the intent of this surveillance requirement. This allowance is consistent with the frequency for the required surveillance and allows time for concentrations of longer lived isotopes to reach equilibrium. In addition, other instrumentation continuously monitors the offgas to alert operators of significant increases in radioactivity.

The proposed change provides more stringent requirements than previously existed in the Technical Specifications. The more stringent requirements will not result in operation that will increase the probability of initiating an analyzed event. If anything, the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes discussed above. The change will not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements will not alter the operation of process variables, structures, systems, or components as described in the safety analyses.

These proposed changes relocate requirements from the Technical Specifications to the T. S. BASES, FSAR, or ODCM. The licensee controlled document containing the relocated requirements will be maintained using the provisions of 10 CFR 50.59 or a change control process in the Administrative Controls Section of the Technical Specifications. Since any changes to these licensee controlled documents will be evaluated per an NRC approved change control process, no increase in the probability or consequences of an accident previously evaluated will be allowed.

Basing the potential fission product release rate on gross gamma activity rate is more representative of the whole body dose that would be received by an individual at the site boundary should a release occur. Therefore, reasonable assurance that the potential whole body accident dose to an individual at the exclusion area boundary will not exceed a small fraction of the limits specified in 10 CFR Part 100 is maintained.

Allowing the sample to be taken from either pretreatment monitor station will have no effect on the objective of assuring that the potential whole body accident dose to an individual at the exclusion area boundary will not exceed a small fraction of the limits specified in 10 CFR Part 100, because both monitor stations are prior to treatment, adsorption, or delay of the noble gases.

RAD Material Source

The requirements for miscellaneous radioactive materials do not impact reactor operation, identify a parameter which is an initial condition assumption for a DBA or transient, identify a significant abnormal degradation of the reactor coolant pressure boundary, and do not provide any mitigation of a design basis event.

Major Design Features

The reformatting, renumbering, and rewording along with the other changes listed involve no technical changes to existing Technical Specifications. The proposed changes are administrative in nature and do not impact initiators or assumptions of analyzed accidents or transient events.

The proposed change provides more stringent requirements than previously existed in the Technical Specifications. The more stringent requirements will not result in operation that will increase the probability of initiating an analyzed event. If anything, the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes discussed above. The change will not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements will not alter the operation of process variables, structures, systems, or components as described in the safety analyses.

These proposed changes relocate requirements from the Technical Specifications to the FSAR. Since any changes to the FSAR must be evaluated per 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed.

Administrative Controls

The reformatting, renumbering, and rewording along with the other changes listed involves no technical changes to existing Technical Specifications. The change to the existing Technical Specifications was done in order to be consistent with the NUREG–1433. During development of NUREG–1433, certain wording preferences or English language conventions were adopted. The proposed change to this section is administrative in nature and does not impact initiators of analyzed events. It also does not impact the assumed mitigation of accidents or transient events.

The proposed change provides more stringent requirements than previously existed in the Technical Specifications. These more stringent requirements are administrative in nature (*e.g.*, specifying additional responsibilities for plant personnel, ensuring overtime control, incorporating program and manual requirements already in place, and adding details to reports). These additional requirements will not alter the plant configuration (no new or different type of equipment will be installed) or changes in methods governing normal plant operation, not alter assumptions relative to the mitigation of an accident or transient event, or alter the operation of process variables, structures, systems, or components as described in the safety analyses.

This proposed change relocates requirements from the Technical Specifications to licensee controlled documents. The licensee controlled documents containing the relocated requirements are required to meet the applicable regulation and any change process invoked by the regulation. Since any changes to the licensee controlled document must continue to meet the regulation, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed.

This change proposes to provide flexibility in meeting the minimum shift staffing for up to two hours in order to provide for unexpected absence. The proposed change does not affect the probability of an accident. The actions of an individual are not assumed to be an initiator of any analyzed event. Also, the change does not negate the requirement to have licensed individuals in the control room. This proposed change does not impact the assumptions of any design basis accident. This change will not alter assumptions relative to the mitigation of an accident or transient event.

This change proposes to relax the requirement to have an individual qualified in radiation protection procedures to be onsite when fuel is in the reactor. The proposed change will allow the position to be vacant for up to two hours in order to provide for unexpected absence.

The proposed change does not affect the probability of an accident. The actions of an individual qualified in radiation protection procedures are not assumed to be an initiator of any analyzed event. Also, the consequences of an accident are not affected by the presence of an individual qualified in radiation protection. This proposed change does not impact the assumptions of any design basis accident. This change will not alter assumptions relative to the mitigation of an accident or transient event. This change will not have any impact on the plant safety because the presence of a person qualified in radiation protection is not required for the mitigation of any accident.

This change proposes to relax the requirement for submitting the Radioactive Effluent Release Report and to relocate the report details outside the TS. The current TS require the report to be submitted semi-annually. This proposed change will allow the report to be submitted annually as required by 10 CFR 50.36a. The proposed change does not affect the probability of an accident. Neither the submittal requirements nor the contents of the Radioactive Effluent Release Report is assumed to be an initiator of any analyzed event. Also, the consequences of an accident are not affected by submittal requirements nor the contents of the Radioactive Effluent Release Report. This proposed change does not impact the assumptions of any design basis accident. This change will not alter assumptions relative to the mitigation of an accident or transient event. This change has no impact on the safe operation of the plant. The report will still be required to be submitted and does not affect any plant equipment or requirements for maintaining plant equipment. The submittal of this report is not required for the mitigation of any accident.

The proposed alternatives for control of access to high radiation areas are consistent with the intent of 10 CFR 20.1601(a) and (b). The proposed changes do not affect the probability of an accident. The controls used for access to high radiation areas are not assumed in the initiation of any analyzed event. Also, the consequences of an accident are not affected by these changes. These changes are both consistent with good radiological safety practice and will provide an adequate level of radiation protection. These proposed changes do not impact the assumptions of any design basis accident. These changes will not alter assumptions relative to the mitigation of an accident or transient event. These changes have no impact on safe operation of the plant.

Radiological Environmental Monitoring

The proposed changes only alter the format and location of procedural details and administrative controls of the radioactive effluents, radiological environmental monitoring, and solid radioactive waste programs. The changes are administrative in nature and do not involve any change to the configuration or operation of plant equipment. The Radiological Effluent Technical Specifications (RETS) procedural details are being moved to the Offsite Dose Calculation manual (ODCM). In addition, new administrative controls have been added to the Technical Specifications which will provide an equivalent level of assurance that activities involving radioactive effluents, solid radioactive waste, and radiological environmental monitoring are conducted in full compliance with regulatory requirements. Since any changes to these requirements will require NRC approval, no increase in the probability or consequences of an accident previously evaluated will be allowed.

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

Operation of PNPS in accordance with the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated because of the following:

Definitions

These proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or different requirements or eliminate any existing requirements.

Relocating these definitions will not alter the plant configuration (no new or different type of equipment will be installed) or change methods governing normal plant operation. Relocating requirements will not impose different requirements and adequate control of information will be maintained. Relocating these definitions will not alter assumptions made in the safety analysis and licensing basis.

RAD Effluents

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or different requirements or eliminate any existing requirements.

Making existing requirements more restrictive and adding more restrictive requirements to the Technical Specifications will not alter the plant configuration (no new or different type of equipment will be installed) or change methods governing normal plant operation. These changes are consistent with current design bases, licensing bases or assumptions made in the safety analysis.

These changes do not alter the plant configuration (no new or different type of equipment will be installed) or methods governing normal plant operation. These changes will not impose different requirements and adequate control of information will be maintained. These changes do not alter assumptions made in the safety analysis and licensing basis.

The proposed change will not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. Operation of the plant will not be altered by this change. This change will not place the plant in any new condition or introduce any mode of operation not previously analyzed.

The proposed change will not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. Operation of the plant will not be altered by this change. This change will not place the plant in any new condition or introduce any mode of operation not previously analyzed.

RAD Material Source

Relocating these requirements will not alter the plant configuration (no new or different type of equipment will be installed) or change methods governing normal plant operation. Relocating requirements will not impose different requirements and adequate control of information will be maintained. Relocating requirements does not alter assumptions made in the safety analysis and licensing basis.

Major Design Features

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or different requirements or eliminate any existing requirements.

Making existing requirements more restrictive and adding more restrictive requirements to the Technical Specifications will not alter the plant configuration (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The change does impose different requirements. However, the change is consistent with assumptions made in the safety analyses.

These changes relocate requirements to the FSAR. These changes do not alter the plant configuration (no new or different type of equipment will be installed) or the methods governing normal plant operation. These changes do not impose different requirements and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis.

Administrative Controls

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or different requirements or eliminate any existing requirements.

Making existing requirements more restrictive and adding new requirements to the Technical Specifications will not alter the plant configuration (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation.

This change relocates requirements to a licensee controlled document. This change will not alter the plant configuration (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. This change will not impose different requirements and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis.

This change proposes to provide flexibility in meeting the minimum shift staffing for up to two hours in order to provide for an unexpected absence. The proposed change will not create the possibility of an accident. This change will not physically alter the plant (no new or different type of equipment will be installed).

This change proposes to relax the requirement to have an individual qualified in radiation protection procedures to be onsite when fuel is in the reactor. The proposed change will allow the position to be vacant for up to two hours in order to provide for unexpected absence. The proposed change will not create the possibility of an accident. This change will not physically alter the plant (no new or different type of equipment will be installed) or the methods of operation.

This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions.

The proposed change will not create the possibility of an accident. This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions.

Radiological Environmental Monitoring

The procedural requirements of the RETS will be maintained in the ODCM. Operation of the plant will not be altered by the changes proposed to the administration of the RETS. This change will not place the plant in any new condition or introduce any mode of operation not previously analyzed.

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

Operation of PNPS in accordance with the proposed change will not involve a significant reduction in a margin of safety because of the following:

Definitions

Definitions perform a supporting function for other sections of the TS and the proposed editing, omission or relocation of definitions associated with this change will not, by itself, reduce existing restrictions on plant operations.

The definitions to be transposed from the Technical Specifications to the ODCM are the same as the existing Technical Specifications. Future changes to the ODCM will be controlled in accordance with proposed technical specification 5.5.1 ''Offsite Dose Calculation Manual (ODCM)''.

RAD Effluents

The change is administrative in nature and does not involve any technical changes. The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. Also, because the change is administrative in nature, no question of safety is involved.

Adding these new requirements and making existing ones more restrictive does not affect any safety analysis assumptions. As such, no question of safety is involved.

The requirements to be relocated from the Technical Specifications to the FSAR T.S. BASES, or ODCM are the same as the existing Technical Specifications and any future changes to this licensee controlled document will be evaluated per an NRC approved change control process.

Specifying a release rate based only on gamma activity is more representative of the whole body dose that would be received by an individual at the site boundary should a release occur. The actual margin of safety could be increased because potential errors in converting beta activity to whole body exposures are eliminated

The sample used to determine the gaseous activity rate will continue to be taken prior to treatment, adsorption, or delay of the noble gases.

RAD Material Source

This change relocates requirements from the Technical Specifications to a licensee controlled document. This change will not reduce a margin of safety since it has no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the Technical Specifications to the licensee controlled documents are the same as the existing Technical Specifications. Since any future changes to these licensee controlled documents must be evaluated per the cited regulations or requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed.

Major Design Features

The changes are administrative in nature and do not involve any technical changes. The proposed changes do not impact initiators or assumptions of analyzed accidents or transient events.

These new or more restrictive requirements are consistent with the current design and licensing bases; therefore, a margin of safety is not affected.

These changes relocate requirements from the Technical Specifications to the FSAR. The requirements to be are the same as the existing Technical Specifications. Since any future changes to the FSAR must be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed.

Administrative Controls

The change is administrative in nature and will not involve any technical changes. The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions.

Adding these new requirements and making existing ones more restrictive does not introduce any new tests or changes in methods governing normal plant operation. Therefore, the changes do not impact any safety analysis assumptions.

This change relocates requirements from the Technical Specifications to a licensee controlled document. The licensee controlled documents containing the relocated requirements are required to meet the applicable regulation and any change process invoked by the regulation. Since any changes to a licensee controlled document must continue to meet the regulation, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed.

This change proposes to provide flexibility in meeting the minimum shift staffing for up to two hours in order to provide for unexpected absence. This proposed change has no effect on the assumptions of a design basis accident. The safety analysis assumptions will still be maintained; thus, no question of safety exists.

This change proposes to relax the requirement to have an individual qualified in radiation protection procedures to be onsite when fuel is in the reactor. The proposed change will allow the position to be vacant for up to two hours in order to provide for unexpected absence. The margin of safety is not affected by the presence or absence on site of an individual qualified in radiation protection procedures. This proposed change has no effect on the assumptions of the design basis accident. This change will not have any impact on the plant safety because the presence of a person qualified in radiation protection is not required for the mitigation of any accident. The safety analysis assumptions will still be maintained; thus, no question of safety exists.

This proposed change has no effect on the assumptions of the design basis accident. This change has no impact on the safe operation of the plant. The report will still be required to be submitted and does not affect any plant equipment or requirements for maintaining plant equipment. The safety analysis assumptions will still be maintained; thus, no question of safety exists.

The proposed alternatives for control of access to high radiation areas are consistent with the intent of 10 CFR 20.1601(a) and (b). The margin of safety is not reduced due to these proposed changes. These changes are both consistent with good radiological safety practices and have been found to provide an adequate level of radiation protection. In addition, these changes provide the benefit of ensuring radiation dose to all workers is minimized by providing the flexibility to select the best means of providing a barrier and access control to a high radiation area given the plant location and radiological conditions. These proposed changes have no impact on the safe operation of the plant. The safety analysis assumptions will still be maintained; thus, no question of safety exists.

Radiological Environmental Monitoring

The proposed changes relocate the procedural details and Bases for RETS from the Technical Specifications to the ODCM. The RETS procedural details and Bases will be maintained by these programs. In addition, new administrative controls have been added to the Technical Specifications which assure the proper control and maintenance of these documents and provide an equivalent level of assurance that activities involving radioactive effluents, solid radioactive waste, and radiological environmental monitoring are conducted in full compliance with regulatory requirements.

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: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 0236.

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

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: November 7, 1997.

Description of amendment request: The proposed amendment would revise the technical specifications and associated bases to allow the licensee to perform 10 CFR Part 50, Appendix J, Type A testing on Byron, Unit 2, and Braidwood, Unit 2, containments at least once per 10 years based on a single successful Type A test, rather than two successful Type A tests.

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

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

Performance of Type A tests at a different interval does not involve a change to any structures, systems, or components, does not affect reactor operations, is not an accident initiator, and does not change any existing safety analysis previously evaluated in the UFSAR [Updated Final Safety Analysis Report]. Therefore, there is no significant increase in the probability of an accident previously evaluated.

Several tables of UFSAR Chapter 15, ''Accident Analyses,'' provide containment leak rate values used in assessing the consequences of accidents discussed in this chapter. Although decreasing the test frequency can increase the probability that an increase in containment leakage could go undetected for an extended period of time, the risk resulting from this proposed change is inconsequential as documented in NUREG–1493, ''Performance-Based Containment Leakage Test Program''. This document indicated that given the insensitivity of reactor risk to containment leakage rate and a small fraction of leakage paths are detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk. Further, industry experience presented in this document indicated that Type A testing has had insignificant impact on uncertainties involved with containment leak rates.

Based on risk information presented in NUREG–1493, the proposed change does not increase the probability or consequences of an accident previously evaluated.

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

The proposed change does not alter the plant design, systems, components, or reactor operations, only the frequency of test performance. New conditions or parameters that contribute to the initiation of accidents would not be created as a result of this proposed change. The change does not involve new equipment and existing equipment does not have to be operated in a different manner, therefore there are no new failure modes to consider.

Changing test intervals as shown in NUREG–1493 has no impact on, nor contributes to the possibility of a new or different kind of accident as evaluated in the UFSAR. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

With the exception of the test frequency, the actual tests will not change. Quantitative risk studies documented in NUREG–1493 regarding extended testing intervals demonstrated that there was minimal impact on the public health and safety. Reducing the frequency, as stated in the NUREG resulted in an ''imperceptible'' increase in risk to public safety. Further, a table in this NUREG regarding risk impacts due to a reduction in testing frequency suggested that there was also minimal difference in risk to the public safety when the test frequency was relaxed.

The proposed change will not reduce the availability of systems and components associated with containment integrity that would be required to mitigate accident conditions nor are any containment leakage rates, parameters or accident assumptions affected by the proposed change.

The proposed change does not involve a significant reduction in a margin of safety, based on the above information.

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: December 30, 1997.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) 3.7.1.3, ''Condensate Storage Tank,'' (CST) and its associated Bases for Byron and Braidwood to raise the minimum allowable CST level to ensure that a sufficient volume of water is available to meet the design basis requirements for the auxiliary feedwater (AFW) system supply. The proposed amendment would also revise the AFW system transfer to essential service water (SX) trip setpoint and allowable value in Table 3.3–4 to ensure that the design basis requirements for the AFW system are accurately reflected in the TS.

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

The amount of water in the CST [Condensate Storage Tank] at the beginning of an accident and the setpoint for AF [auxiliary feedwater] pump suction pressure-low trip have no impact on the probability of occurrence of any accident analyzed in the UFSAR [Updated Final Safety Analysis Report]. This is due to the availability of the safety-related SX [essential service water] water supply as a backup system. Therefore, the probability of an accident previously evaluated is unchanged.

The loss of the Safety Category II CST under accident conditions has already been evaluated in the UFSAR. The SX system is the emergency source of water supply to the AF system under accident conditions. The design basis analysis for the essential service water (SX) system and the Limiting Condition for Operation requirements for the ultimate heat sink ensure that a sufficient supply of water is available to plant operators to mitigate the consequences of all analyzed accidents. None of the proposed changes to the CST minimum level or the setpoints documented in TS Table 3.3–4, functional unit 6.g. has any negative impact on the assumptions or results of these analyzed accidents. To the contrary, the proposed changes will ensure that the CST remains available as the primary supply of water to the AF system and that automatic suction transfer will occur for circumstances where the Safety Category II CST becomes unavailable (*e.g.*, seismic event or tornado).

The level in the CST and the associated instrumentation and setpoints help ensure that sufficient water is available to plant operators to mitigate the consequences of accidents that are analyzed in the UFSAR. The SX system is the emergency source of water credited in the UFSAR. However, the proposed Technical Specification Bases require that sufficient water be maintained in the CST to respond to postulated events where the CST remains available (*e.g.*, non-seismic related events and events with no tornado assumed). The proposed CST levels ensure that this requirement is met. The water level requirement for the CST provides additional assurance that plant operators remain capable of responding to postulated events as described in the UFSAR. Therefore, the proposed changes do not increase the consequences of an accident previously evaluated.

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

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

The proposed changes are being implemented to account for instrument accuracy and AF system suction requirements that affect the volume of useable water in the CST. The amendment request incorporates the full design requirements of the AF System and components to ensure that sufficient water is maintained in the CST. The changes reduce the probability of an undesirable introduction of lower quality essential service (SX) system water into the steam generators unless required due to the unavailability of the CST during emergency conditions (*e.g.*, seismic event or tornado). Although the SX system is the safety-related water supply to AF, the water contains high levels of impurities and sediment that could eventually degrade the steam generators. The CST contains demineralized water. Therefore, the long term reliability and availability of the steam generators is enhanced by precluding introduction of SX water into the steam generators unless required under emergency conditions. The proposed CST levels account for the incremental increase in CST water volume required due to the larger metal mass and primary volume of the replacement steam generators for Byron Unit 1 and Braidwood Unit 1. Finally, the trip setpoint and allowable values in Table 3.3–4 of the TS are being updated to reflect the current design basis of the AF system. The required CST level changes when plant modifications are completed. Each configuration has been evaluated and the associated CST level maintains a sufficient water volume to perform its design function.

The modification to the suction pressure circuitry involves the addition of an electronic ''lead-lag'' circuit card for the motor-driven AF pump, which experiences the most severe startup suction pressure transients. This circuit card will be set up for ''lag'' only operation and will filter the suction pressure signal during transients associated with pump startup or other sudden changes in flow or pressure. This will prevent an inadvertent trip during transient conditions when the CST is available. In situations where the CST is unavailable, the suction pressure will decrease with no recovery until switchover. Under this condition, the output of the lead-lag card will continue to decrease as well until the switchover setpoint is reached. The time constant of the lead-lag card was selected such that the resulting time delays in actuating SX switchover and pump trip are consistent with pump protection requirements.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated. This conclusion is also valid when considering the planned modifications to the AF suction pressure transient circuitry.

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

The proposed change is made in the conservative direction with respect to the current TS requirements for minimum CST level and AF pump CST to SX switchover setpoints. Increasing the volume of water contained in the CST level provides redundancy to the safety-related source of water to the AF supply, which is the SX system. In combination, the CST and the SX system ensure that sufficient water is available to feed the steam generators under all anticipated normal and emergency conditions to cool a unit from full power conditions down to 350 degrees Fahrenheit, when the residual heat removal system can be placed into service. The proposed changes ensure the CST will have sufficient water to meet all normal operating conditions and mitigate the consequences of all analyzed accidents except those that result in CST unavailability. In addition, automatic switchover of the AF water supply from the CSTs to SX will occur as assumed in the current safety analyses for events where the CST becomes unavailable. The SX system remains capable of supplying the emergency source of water to the AF supply.

Therefore, the proposed 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: 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.

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: January 28, 1998 (NRC–98–0002).

Description of amendment request: The proposed amendment would revise technical specification (TS) surveillance requirements 4.8.2.1.a.2, 4.8.2.1.b, and 4.8.2.1.c.4 to accommodate differences in the monitored parameters between the existing batteries and the batteries that will be installed for Division II during the sixth refueling outage.

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

The proposed changes do not involve a change in the manner in which the plant is operated. TS Section 4.8.2.1 is being revised to reflect the new Division II battery cell/system characteristics and associated requirements. The new battery will have an increased capacity over the present battery, while maintaining the existing battery system voltage requirements. This is possible because the present and new battery specific gravity (1.215) and type (lead calcium) are the same. Also, the end of battery system discharge voltage remains the same as 210 VDC. The Division II batteries will continue to furnish power to redundant essential loads as required and as designed. The new surveillance requirement voltages are based on the same volts/cell criteria used for the existing batteries. Furthermore, failure or malfunction of the station batteries does not initiate any of the analyzed accidents previously evaluated in the UFSAR [updated final safety analysis report]. The changes described will therefore not involve an increase in the probability or consequences of an accident previously evaluated.

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

The new battery is Class 1E qualified equipment and is being maintained within the same overall design parameters as the existing battery. That is, the battery terminal voltage on float voltage conditions (2.167 volt[s]/cell), overvoltage conditions (2.5 volts/cell) and charger capability (2.15 volts/cell) are the same as the original design. Furthermore, the end of system discharge voltage of the battery system is maintained the same; therefore, there is no negative impact to plant loads supplied by the batteries. Failures of the batteries and chargers have been considered in both the existing and modified configurations. The proposed changes will not change performance or reliability nor introduce any new or different failure modes or common mode failure and will therefore not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The changes act to increase overall battery capacity from 560 ampere-hours to 1200 ampere-hours with the minimum battery discharge voltage remaining at 210 VDC (or 105 VDC per battery). The battery terminal voltage on float voltage conditions (2.167 volt[s]/cell), overvoltage conditions (2.5 volts/cell) and charger capability (2.15 volts/cell) are the same as the original design. The new surveillance requirement voltages are based on the same volts/cell criteria used for the existing batteries. The batteries' ability to satisfy the design requirements (battery duty cycle) of the dc system will not be reduced from original plant design and will therefore not have any negative impact to plant loads the battery supplies. The proposed changes therefore do not involve a reduction in the margin of safety.

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

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161.

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Project Director: Cynthia A. Carpenter.

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

Date of amendment request: January 28, 1998 (NRC–98–0003).

Description of amendment request: The proposed amendment would revise technical specification (TS) 3.4.10, TS Figure 3.4.10–1 and the associated bases by changing the prohibited and restricted operating regions associated with core thermal-hydraulic stability. TS 3.4.1.4, TS Figure 3.4.1.4–1, and the associated bases would also be revised to reflect stability-related improvements in operating restrictions for idle recirculation loop startup. Finally, in an unrelated change, TS Tables 3.3.7.5–1 and 4.3.7.5–1 would be revised to delete neutron flux from the parameters the licensee is required to monitor by TS 3.3.7.5, Accident Monitoring Instrumentation.

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:

Thermal Hydraulic Stability and Idle Recirculation Loop Startup

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

These changes act to prohibit operations which have been found to carry a significant potential for the formation of core thermal-hydraulic instabilities and eliminates inappropriate technical specifications for maintaining <50% recirculation loop flow before starting the idle recirculation pump. As such, operation in compliance with the proposed provisions does not affect any initiating mechanism for previously evaluated accidents or the response of the plant to a previously evaluated accident. The actions taken lead to placing the plant in a safe condition and are not themselves associated with an initiator for a previously evaluated accident. Therefore, the change does not represent a significant increase in the probability or consequences of any previously evaluated accident.

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

As discussed above, the change acts to restrict operations previously allowed. The change also provides remedial actions that act to place the plant in a safe condition. The actions specified are within the analyzed domain of plant operations. Unless an instability event is in progress, the new allowance to use a core flow increase to leave the Exit Region is no different than normal plant maneuvering. If an instability event is in progress, the new ACTION 3.4.10.c to scram the reactor takes precedence. The allowance to start an idle loop with the active loop flow <50% of rated flow has been shown to have no adverse [e]ffect on scram avoidance or jet pump riser brace vibration. Therefore, the proposed changes do not create a new or different type of accident.

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

Consistent with the latest BWROG [Boiling Water Reactor Owners Group] guidance, the changes act to expand the Exit region compared to the current TS for core thermal-hydraulic instability and provide improved remedial actions which promptly terminate the potential for instability. These changes therefore do not involve a significant reduction in a margin of safety.

Post-Accident Monitoring

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

The proposed change does not involve a change in plant design or a change in the manner in which the plant is operated. The long term post-accident design requirements of the Neutron Monitoring System (NMS) are not based on operator use for transients with scram, accidents with scram, and other occurrences without scram (Reference 6 [of January 28, 1998, application]). For lesser events such as transients without scram, the NMS enhances the operator actions, since successful verification that power is below approximately 3% power can avoid non-routine operator actions (Reference 6). These lesser events establish design requirements for the NMS. The failure of this instrumentation during post-accident conditions will not prevent the operator from determining reactor power levels. Alternate parameter status will be available from which reactor power may be inferred. Based on the multiple inputs available to the operator, sufficient information will be available upon which to base operational decisions and to conclude that reactivity control has been accomplished. This change will therefore not represent a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not introduce a new mode of plant operation and does not involve the installation of any new equipment or modifications to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change eliminates a TS listing of a function to reflect the actual safety significance. As such it has no effect on actual plant operation and thus no impact on any margin of safety.

Based on the above, Detroit Edison has determined that the proposed amendment does not involve a significant hazards consideration.

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

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161.

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Project Director: Cynthia A. Carpenter.

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

Date of amendment request: January 28, 1998 (NRC–98–0006).

Description of amendment request: The proposed amendment would revise technical specification (TS) surveillance requirement 4.4.3.2.2.a to extend the interval for leak rate testing of pressure isolation valves from 18 months to 24 months.

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

The proposed change revises the periodicity of TS Surveillance Requirement (SR) 4.4.3.2.2.a from ''At least once per 18 months'' to ''At least once per 24 months.'' This change revises the testing periodicity only; no other testing methodology is being affected. The testing periodicity is being revised to be consistent with other Category ''A'' valves since the Pressure Isolation Valves (PIVs) are classified as Category ''A'' valves. Both ASME [American Society of Mechanical Engineers] [Code] Section XI and NUREG–1482 require Category ''A'' valves to be leak tested on a periodicity of at least once every 2 years.

The function of the PIVs is to protect the low pressure portions of safety systems from the RCS [reactor coolant system] pressure. Periodic valve leak rate testing is performed on the PIVs to assure system integrity is maintained and to prevent the design pressure of the low pressure systems from being exceeded. The frequency of the inservice test could increase the probability that an increase in PIV seat leakage may occur. If this were to occur and the leakage was significant (assuming leakage through both the inboard and outboard valves of the same penetration), the excess leakage would be detected by the system leakage detection instrumentation which would require corrective actions to be taken to assure that leakage remained within allowable limits. Considering that past test results show very minimal seat leakage changes over years of service, the consequences and probabilities resulting from the proposed change is considered minimal.

The proposed change does not impose or eliminate any testing requirements. This change is only a change to the frequency (testing interval) for measuring the seat leakage through the PIVs. The PIVs will continue to be tested in accordance with ASME Code Section XI. This change does not affect any of the parameters or conditions that could contribute to the initiation of any accidents previously evaluated and therefore cannot increase the consequences or probabilities of any accident previously evaluated.

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

The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to the initiation of any accidents. This change only involves the lengthening of the PIVs' testing frequency from 18 months to 24 months. The method for performing the actual tests are not changed. No new accident scenarios are created by extending the testing intervals. No safety-related equipment or safety functions are altered as a result of this change. Therefore, extending the test frequency does not create the possibility of a new or different kind of accident or malfunction from those previously analyzed.

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

The proposed change only affects the frequency of the PIVs' seat leakage tests. The frequency is proposed to be extended to reflect the ASME Section XI, 1980 Edition, Winter 1980 Addenda, Section IWV–3422 seat leakage testing periodicity requirement of 24 months. No other testing methodology is being changed. The allowable leakage limits will not be affected by this change. The margin of safety as defined in the bases of any Technical Specification will, therefore, not be reduced by extending the testing periodicity of the subject valves.

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: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161.

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Project Director: Cynthia A. Carpenter.

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

Date of amendment request: January 28, 1998 (NRC–98–0008).

Description of amendment request: The proposed amendment would revise the technical specifications (TSs) by modifying the ''#'' footnote to Table 1.2 and the ''*'' footnote to surveillance requirements 4.9.1.2 and 4.9.1.3 to permit the Reactor Mode Switch to be placed in the Run or Startup/Hot Standby positions to test switch interlock functions provided that all control rods are verified to remain fully inserted in core cells containing one or more fuel assemblies.

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

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

The proposed change would permit the Reactor Mode Switch to be placed in the Run or Startup/Hot Standby positions to test the switch interlock functions provided that all control rods are verified to remain inserted in core cells containing one or more fuel assemblies. The existing TS requires that all control rods be verified to remain inserted regardless of whether core cells are defueled. The reactor mode switch refuel position interlocks restrict the operation of refueling equipment or withdrawal of control rods to reinforce unit procedures that prevent the reactor from achieving criticality during refueling operations. As such, the refueling equipment interlocks preserve the assumptions for the analyses of a control rod withdrawal event or loading of a fuel assembly into an uncontrolled cell during refueling operations. The reactor mode switch refuel position interlocks are not initiators of any previously evaluated accident. The revised footnote requires that all control rods remain fully inserted in core cells containing one or more fuel assemblies while the mode switch is moved to support interlock testing. Additionally, when the reactor mode switch is unlocked to support interlock testing, TS 3.9.1 prohibits core alterations. With all control rods fully inserted in core cells containing one or more fuel assemblies and no core alterations in progress, there are no credible mechanisms to initiate a reactivity excursion during the interlock testing. Therefore, the proposed change does not involve a significant increase in the probability of a previously evaluated accident.

The proposed change accommodates reactor mode switch refuel position interlock testing with one or more control rods removed as permitted by TS 3.9.10.1 and 3.9.10.2. In addition to requiring all fuel assemblies to be removed from core cells associated with removed control rods, TS 3.9.10.1 and 3.9.10.2 require minimum shutdown margin to be maintained in accordance with TS 3/4.1.1. Under these conditions, it is not possible for criticality to occur in the event of a withdrawal of a single control rod or loading of fuel assemblies into a single core cell with no control rod inserted. Therefore, the proposed change does not involve a significant increase in the consequences of a previously evaluated accident.

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

Repositioning of the reactor mode switch to test refueling position interlocks is permitted by both the existing and proposed TS. The proposed change affects only the conditions under which the mode switch can be repositioned. The proposed changes do not change underlying principles affecting the way in which the plant is operated and no new or different failure modes are introduced by the proposed change for any plant system or component. No new limiting single failure has been identified as a result of the proposed changes. Therefore, no new or different types of failures or accident initiators are introduced by the proposed changes.

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

The proposed change described above affects the conditions under which the reactor mode switch can be repositioned to accommodate refuel position interlock testing. The proposed change in combination with existing restrictions within the TS provide assurance that there is no credible mechanism to initiate a reactivity excursion during interlock testing. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161.

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Project Director: Cynthia A. Carpenter.

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

Date of amendment request: January 28, 1998 (NRC–98–0011).

Description of amendment request: The proposed amendment would revise technical specification (TS) 3.4.2.1 by changing the tolerance for the as-found setpoints of the safety/relief valves (SRVs) from [plus or minus] 1 percent to [plus or minus] 3 percent of the nominal setpoint. The revised tolerance would be used when evaluating whether setpoint test results were acceptable. However, after initial testing, the as-left setpoints of the SRVs would be adjusted to within [plus or minus] 1 percent of the nominal setpoint.

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

The proposed change allows an increase in the SRV setpoint tolerance, determined by test after the valves have been removed from service, from [plus or minus] 1% to [plus or minus] 3%. The proposed change does not alter the SRV lift setpoints, the SRV lift setpoint test frequency, or the number of SRVs required to be operable. This change does not involve physical changes to the SRVs, nor does it change the operating characteristics or safety function of the SRVs. This change requires that the SRVs be adjusted to within [plus or minus] 1% of their nominal lift setpoints following testing and prior to installation in the plant.

The only change, other than the change in setpoint tolerance, will be to increase the maximum rated speed of the RCIC [reactor core isolation cooling] turbine and pump. The increased speed is within the design limits of the system and the overspeed trip function retains adequate margin; therefore, RCIC operability is not affected by this change. Additionally, SRV actuation is not a precursor to any design basis accident analyzed for the Fermi 2 plant. Therefore, this change will not significantly increase the probability of an accident previously evaluated.

Generic considerations related to the change in setpoint tolerance were addressed in NEDC–31753P, ''BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report,'' and were reviewed and approved by the NRC. The plant specific evaluations identified in the NRC[']s Safety Evaluation for NEDC–31753P were performed in order to support the proposed change (Cycle 6 reload licensing report, Power Uprate Safety Analysis, and NEDC–32788P, ''Safety Review for Enrico Fermi Energy Center Unit 2 Safety/Relief Valve Setpoint Tolerance Relaxation Analyses''). These evaluations included transient analysis of the anticipated operational occurrences (AOOs); analysis of the design basis overpressurization event; evaluation of the performance of high pressure systems, motor operated valves, and vessel instrumentation and associated piping; and evaluation of the containment response during LOCA [loss of coolant accident] and the hydrodynamic loads on the SRV discharge lines and containment. Although not specified in the generic topical report NEDC–31753P, an analysis of the short term pressurization phase of an ATWS [anticipated transient without scram] event was also performed. These analyses show that there is adequate margin to the design core thermal limits and to the reactor vessel pressure limits using a [plus or minus] 3% SRV setpoint tolerance. They also show that operation of the high pressure injection systems will not be adversely affected; and the containment response during LOCA will be acceptable. Therefore, this change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change to allow an increase in the SRV setpoint tolerance from [plus or minus] 1% to [plus or minus] 3% does not alter the SRV lift setpoints, the minimum SRV lift setpoint test frequency, or the number of SRVs required to be operable. This change does not involve physical changes to the SRVs, nor does it change the operating characteristics or the safety function of the SRVs. The only change to plant equipment will be to increase the RCIC turbine/pump maximum rated speed from 4550 rpm to 4600 rpm. The RCIC pump and turbine have been verified to be capable of operating at the increased speed, pressure and temperature associated with this increase in maximum rated speed. These changes do not result in any changed component interactions. The SRVs and the RCIC System will continue to function as designed. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

While the calculated peak vessel pressures for the ASME [American Society of Mechanical Engineers] overpressure event and the ATWS MSIVC [main steam isolation valve closure] event are higher than those calculated without the setpoint tolerance relaxation, both are still within the respective licensing acceptance limits associated with these events. Similarly, although the loads associated with SRV blowdown could increase slightly, containment loadings have been determined to remain within acceptance limits. These licensing acceptance limits have been determined by the NRC to provide a sufficient margin of safety. Additionally, the increased setpoint tolerances have been determined to have a negligible effect on the other accidents and transients analyzed. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161.

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Project Director: Cynthia A. Carpenter.

Duquesne Light Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of amendment request: January 17, 1998.

Description of amendment request: The proposed amendment would revise the waste gas system line break accident analysis. The proposed changes would affect Beaver Valley Power Station, Unit No. 1 Updated Final Safety Analysis Report (UFSAR) Tables 11.3–7, ''Postulated Control Room Accident Dose,'' and 14.2–8, ''Parameters Used In Control Room Habitability Analysis Of The Waste Gas System Failure Analysis.'' The analysis references on Tables 11.3–7 and 14.2–8 would be revised due to the reanalysis of the waste gas system line break accident. In Table 11.3–7, the waste gas system line break accident gamma dose value would be revised from 0.0031 Rem to less than 0.01 Rem and the beta dose value would be revised from 0.013 Rem to less than 1.0 Rem.

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

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

The proposed change has no effect on the probability of an accident previously evaluated. The proposed change results from the correction of values and change to assumptions utilized in the original calculation to address resultant dose to Control Room operators in the event of the postulated Waste Gas System line break.

The proposed change also corrects an error in UFSAR Table 14.2–8 whereby the fraction of fuel with defects was assumed to be one percent, not 0.0026. This correction reflects the value used in the calculation and does not alter the results.

The proposed change does not significantly increase the consequences of an accident previously analyzed. Although the correction to the calculation and revision to the assumptions used result in an insignificant increase to the postulated dose to the Control Room operators, the results remain below the acceptance limit of other postulated accidents presented in the UFSAR (Table 11.3–7) and the acceptance approved by the NRC in the NRC Safety Evaluation Report, Section 15.1, dated October 1974. The proposed change does not alter the currently approved Technical Specification. The proposed change does not affect the dose to the public.

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

The proposed change does not alter the physical plant or modify the modes of operation. The proposed change does not involve modifications to plant equipment nor does it alter operation of plant systems. Therefore operation of the facility with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not reduce the margin of safety. The proposed change does not affect any plant systems or equipment. Therefore, the response of the plant to any actual events will not be affected, and the change does not involve a reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: November 13, 1997.

Description of amendment request: The proposed change will modify Technical Specification (TS) 6.8.4.a, ''Primary Coolant Sources Outside Containment,'' to add portions of the containment vacuum relief (CVR) system and the primary sampling system to the program at Waterford 3.

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. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change adds the containment vacuum relief (CVR) system and the primary sampling system to the Primary Coolant Sources Outside Containment Program in the Technical Specifications. The program will require preventative maintenance and periodic visual inspection, and leak rate testing on appropriate portions of these systems to ensure leakage of radioactive fluids are as low as practicable. The addition of these two systems to the program will not affect the probability of an accident. Neither the CVR system nor the primary sampling system are initiators of any analyzed event. The consequences of an accident are not affected by this change. The maximum allowed leakage limits are not being increased due to the addition of these two systems. Any leakage from the CVR system will be factored into the overall leakage limits and any leakage from the primary sampling system will be kept to a minimum by performing required maintenance. This change does not affect the mitigation capabilities of any component or system nor does it affect the assumptions relative to the mitigation of accidents or transients. The addition of these systems to the program also helps ensure that the systems will perform their intended function. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

Response: No.

The proposed change adds the CVR system and the primary sampling system to the Primary Coolant Sources Outside Containment Program in the Technical Specifications. The program will require preventative maintenance and periodic visual inspection, and leak rate testing on appropriate portions of these systems to ensure leakage of radioactive fluids are as low as practical. Neither the design nor configuration of the plant is being changed due to the addition of the CVR system to the program. Also, as a result of the CVR system being added to the program, there has been no physical change to plant systems, structures or components nor will the addition of the CVR system reduce the ability of any of the safety-related equipment required to mitigate anticipated operational occurrences (AOOs) or accidents.

Although the addition of the primary sampling system to the program was a result of a change to the configuration of the plant, it does not reduce the ability of any safety-related equipment required to mitigate AOOs or accidents. Any leakage from the primary sampling system will be kept to a minimum by performing required maintenance.

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

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change adds the CVR system and the primary sampling system to the Primary Coolant Sources Outside Containment Program in the Technical Specifications. The program will require preventative maintenance and periodic visual inspection, and leak rate testing on appropriate portions of these systems to ensure leakage of radioactive fluids are as low as practical. This change will not affect the maximum containment leakage allowed in the Technical Specifications. The leakage from the CVR system will be added to the overall containment leakage rate. Any leakage from the primary sampling system will be kept to a minimum by performing required maintenance. The overall containment leakage requirement is required to be met and therefore, this change will not result in an increase in the analyzed dose consequences assumed in the Waterford 3 safety analysis. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: December 31, 1997.

Description of amendment request: The proposed amendment will revise Technical Specification 5.6.1 and associated Figure 5.6–1, and Specification 5.6.3, to permit an increase in the allowed Spent Fuel Pool (SFP) storage capacity. The analyses supporting this request, in part, assume credit for up to 1266 ppm boron concentration existing in the SFP.

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

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

Analyses to support the proposed fuel pool capacity increase have been developed using conservative methodology. The analysis of the potential accidents summarized below has shown that there is no significant increase in the consequences of any accident previously analyzed. A review of relevant plant operations has also demonstrated that there is no significant increase in the probability of occurrence of any accident previously analyzed. This conclusion is also discussed below.

Previously evaluated accidents that were examined for this proposed license amendment include: Fuel Handling Accident, Spent Fuel Cask Drop Accident, and Loss of all Fuel Pool Cooling.

There will be no change in the mode of plant operation or in the availability of plant systems as a result of this proposed change; the systems interfacing with the spent fuel pool have previously encountered borated pool water and are designed to interact with irradiated spent fuel and remove the residual heat load generated by isotopic decay. The proposed amendment does not require a change in the maintenance interval or maintenance scope for the fuel pool cooling system or for the spent fuel cask crane. The frequency of cask handling operations and the maximum weight carried by the crane is not increased as a result of the proposed license amendment. Thus, there will be no increase in the probability of a loss of fuel pool cooling or in the probability of a failure of the cask crane as a result of the proposed amendment.

There will not be a significant increase in the frequency of handling discharged assemblies in the fuel pool as a result of this change; any handling of fuel in the spent fuel pool will continue to be performed in borated water. If the license amendment is approved, there will be a one-time repositioning of certain discharged assemblies stored in the fuel pool to comply with the revised positioning requirements, but the increased pool storage capacity will permit the deferral of spent fuel handling associated with cask loading operations. Fuel manipulation during the repositioning activity will be performed in the same manner as for fuel placed in the spent fuel pool during refueling outages. There will be no changes in the manner of handling fuel discharged from the core as a result of refueling; administrative controls will continue to be used to specify fuel assembly placement requirements. The relative positions of Region I and Region II storage locations will remain the same within the fuel pool. Therefore, the probability of a fuel handling accident has not been significantly increased.

The consequences of a fuel handling accident have been evaluated. The radioactive release consequences of a dropped fuel assembly are not affected by the proposed increase in fuel pool storage capacity. They remain bounded by the results of calculations performed to justify the existing St. Lucie Unit 2 fuel storage racks and burnup limits. At the limiting fuel assembly burnup, radioactive releases from a dropped assembly would be only a small fraction of NRC guidelines. The input parameters employed in analyzing this event are consistent with the current values of fuel enrichment, discharge burnup and uranium content used at St. Lucie Unit 2 and with future use of the ‘‘value-added’’ fuel pellet design. Thus, the consequences of the fuel assembly drop accident would not be significantly increased from those previously evaluated.

The capability of the fuel pool cooling system to handle the increased number of discharged assemblies has been examined. The impact of a total loss of spent fuel pool cooling flow on available equipment recovery time and on fuel cladding integrity has also been evaluated. For the limiting full core discharge, sufficient time remains available to restore cooling flow or to provide an alternate makeup source before boiloff results in a fuel pool water level less than that needed to maintain acceptable radiation dose levels. Analysis has shown that in the event of a total loss of fuel pool cooling fuel cladding integrity is maintained. Therefore, the consequences of a loss of fuel pool cooling event, including the effect of the proposed increase in fuel pool storage capacity, have not been significantly increased from previously analyzed results for this type of accident.

The analysis of record pertaining to the radiological consequences of the hypothetical drop of a loaded spent fuel cask just outside the Fuel Handling Building was examined to determine the impact of the increased fuel storage capacity on this accident's results. The results of the previously performed analysis were determined to bound the conditions described by the proposed license amendment, thus the consequences of the cask drop accident would not be significantly increased as a result of this change.

It is concluded that the proposed amendment to increase the storage capacity of the St. Lucie Unit 2 spent fuel pool will not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

In this license amendment FPL proposes to credit the negative reactivity associated with a portion of the soluble boron present in the spent fuel pool. Soluble boron has always been present in the St. Lucie Unit 2 spent fuel pool; as such the possibility of an inadvertent fuel pool dilution has always existed. However, the spent fuel pool dilution analysis demonstrates that a dilution of the Unit 2 spent fuel pool which could increase the pool k_{eff} to greater than 0.95 is not a credible event. Neither implementation of credit for the reactivity of fuel pool soluble boron nor the proposed increase in the fuel pool storage capacity will create the possibility of a new or different type of accident at St. Lucie Unit 2.

An examination of the limiting fuel assembly misload has determined that this would not represent a new or different type of accident. None of the other accidents examined as a part of this license submittal represent a new or different type of accident; each of these situations has been previously analyzed and determined to produce acceptable results.

The proposed license amendment will not result in any other changes in the mode of spent fuel pool operation at St. Lucie Unit 2 or in the method of handling irradiated nuclear fuel. The spatial relationship between the fuel storage racks and the cask crane range of motion is not affected by the proposed change.

As a result of the evaluation and supporting analyses, FPL has determined that the proposed fuel pool capacity increase does not create the possibility of a new or different type of accident from any accident previously evaluated.

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

FPL has determined, based on the nature of the proposed license amendment that the issue of margin of safety, when applied to this fuel pool capacity increase, should address the following areas:

(1) Fuel Pool reactivity considerations
(2) Fuel Pool boron dilution considerations
(3) Thermal-Hydraulic considerations
(4) Structural loading and seismic considerations

The Technical Specification changes proposed by this license amendment, the proposed spent fuel pool storage configuration and the existing Technical Specification limits on fuel pool soluble boron concentration provide sufficient safety margin to ensure that the array of fuel assemblies stored in the spent fuel pool will always remain subcritical. The revised spent fuel storage configuration is based on a Unit 2 specific criticality analysis performed using methodology consistent with that approved by the NRC. Additionally, the soluble boron concentration required by current Technical Specifications ensures that the fuel pool k_{eff} will always be maintained substantially less than 0.95.

The Unit 2 criticality analysis established that the k_{eff} of the spent fuel pool storage racks will be less than 1.0 with no soluble boron in the fuel pool water, including the effect of all uncertainties and tolerances. Credit for the soluble boron actually present is used to offset uncertainties, tolerances, off-normal conditions and to provide margin such that the spent fuel pool k_{eff} is maintained less than or equal to 0.95. FPL has also demonstrated that a decrease in the fuel pool boron concentration such that k_{eff} exceeds 0.95 is not a credible event.

Current Technical Specifications require that the fuel pool boron concentration be maintained greater than or equal to 1720 ppm. This boron value is substantially in excess of the 520 ppm required by the uncertainty and reactivity equivalencing analyses discussed in this evaluation and the 1266 ppm value required to maintain k_{eff} less than or equal to 0.95 in the presence of the most adverse mispositioned fuel assembly.

The St. Lucie Unit 2 fuel pool boron concentration will continue to be maintained significantly in excess of 1266 ppm; the proposed license amendment will not result in changes in the mode of operation of the refueling water tank (RWT) or in its use for makeup to the fuel pool. Thus, operation of the spent fuel pool following the proposed change, combined with the existing fuel pool boron concentration Technical Specification limit of 1720 ppm, will continue to ensure that k_{eff} of the fuel pool will be substantially less than 0.95.

Even if this not-credible dilution event was to occur, no radiation would be released; the only consequence would be a reduction of shutdown margin in the fuel pool. The volume of unborated water required to dilute the fuel pool to a k_{eff} of 0.95 is so large (in excess of 358,900 gallons to dilute the fuel pool to 520 ppm boron) that only a limited number of water sources could be considered potential dilution sources. The likelihood that this level of water use could remain undetected by plant personnel is extremely remote.

In meeting the acceptance criteria for fuel pool reactivity, the proposed amendment to increase the storage capacity of the existing fuel pool racks does not involve a significant reduction in the margin of safety for nuclear criticality.

Calculations of the spent fuel pool heat load with an increased fuel pool inventory were performed using ANSI/ANS–5.1–1979 methodology. This method was demonstrated to produce conservative results through benchmarking to actual St. Lucie Unit 2 fuel pool conditions and by comparison of its results to those generated by a calculation using Auxiliary Systems Branch Technical Position 9–2 methodology. Conservative methods were also used to demonstrate fuel cladding integrity is maintained in the absence of cooling system forced flow. The results of these calculations demonstrate that, for the limiting case, the existing fuel pool cooling system can maintain fuel pool conditions within acceptable limits with the increased inventory of discharged assemblies. Therefore, the proposed change does not result in a significant reduction in the margin of safety with respect to thermal-hydraulic or spent fuel cooling considerations.

The primary safety function of the spent fuel pool and the fuel storage racks is to maintain discharged fuel assemblies in a safe configuration for all environments and abnormal loadings, such as an earthquake, a loss of pool cooling or a drop of a spent fuel assembly during routine spent fuel handling. The proposed increase in spent fuel inventory on the fuel pool and the existing storage racks have been evaluated and show that relevant criteria for fuel rack stresses and floor loadings have been met and that there has been no significant reduction in the margin of safety for these criteria.

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

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

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

NRC Project Director: Frederick J. Hebdon.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of amendment request: November 22, 1996, as revised and replaced on February 2, 1998.

Description of amendment request: The licensee proposed to change the Technical Specifications (TS) to allow the use of a temporary fuel oil storage system for up to 10 days in order to perform a surveillance requirement on the Unit 3 fuel oil storage tank with Unit 3 in Modes 5, 6, or defueled.

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:

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

The proposed amendment will allow the installation of a temporary fuel oil storage and transfer system for up to 10 days, once every 10 years. EDGs [emergency diesel generators] are designed as backup AC power sources for essential safety systems in the event of a loss of offsite power. Since the EDGs are not accident initiators, the probability of occurrence of accidents previously analyzed has not been increased.

The temporary fuel oil storage tanks will be located greater than fifty (50) feet from safety related or safe shutdown components or circuits. This does not produce any threat to fire protection or safe shutdown capability and therefore represents a configuration that is bounded by existing fire hazards analysis.

The proposed amendment will not change the condition or minimum amount of operating equipment assumed in the plant safety analyses for accident mitigation. The temporary fuel storage and transfer system provides a reliable means of performing the required delivery support function for the Unit 3 EDGs.

An insignificant increase in the consequences of an accident previously evaluated is possible since the temporary storage and transfer system will not meet requirements for Seismic Category I or Class 1E. However, the probability of a seismic event will be very low due to the limited time that the temporary storage system will be in use.

The increase in the consequences of an accident previously evaluated is insignificant due to the following:

Manual actions required to provide a 7 day supply of fuel to the EDGs can easily be accomplished in the 17 hours of EDG operation provided by the 3880 gallon capacity of a single EDG day and skid tank. The location of the temporary fuel oil supply inside the protected area security fence by the Central Receiving Facility provides multiple access routes to transfer fuel to the Unit 3 EDGs and is in close proximity to a severe weather shelter for the mobile tanker.

Additionally, more than 17 hours will be available to manually transfer fuel from the temporary fuel storage tanks located inside the protected area, by filling the Unit 4 EDG storage tanks with approximately 8600 gallons of fuel oil above that required for Unit 4 EDG operability. This extra capacity will be available to the Unit 3 EDGs prior to taking the permanent Unit 3 storage tank out of service. This will be done by filling the Unit 4 fuel tanks to 39,000 gallons, which is just below the high level alarm. This gives a capacity of 4300 gallons in each tank above the Unit 4 Technical Specification minimum required volume of 34,700 gallons. The Unit 4 tanks are contained within a Seismic Class 1 structure and protected by installed fire protection equipment.

Combining the excess available fuel from the Unit 4 storage tanks and the nominal volume of the Unit 3 day and skid tanks gives a total of 12,480 gallons (4300×2+3880) of available fuel to either of the Unit 3 EDGs. This allows a run time for a Unit 3 EDG of 55 hours (assuming fuel oil transfer from Unit 4) prior to reaching the Technical Specification minimum volume for the Unit 4 fuel oil storage tanks. Manual actions to replenish the Unit 4 or Unit 3 fuel oil storage tanks from the temporary storage tanks, via the mobile tanker, can easily be accomplished within the 55 hours. Procedures currently exist for the transfer of fuel from (1) the mobile tanker to the auxiliary fill station at the Unit 3 EDGs, and (2) from the Unit 4 EDG storage tanks to the Unit 3 day tanks by using either of the Unit 4 transfer pumps. The

Unit 4 transfer pumps are powered from redundant Class 1E power supplies.

The temporary storage tanks will be located inside the protected area in the vicinity of the Nuclear Plant Central Receiving Facility. The temporary tanks will be located greater than fifty (50) feet from safety related or safe shutdown components or circuits. This does not produce any threat to fire protection or safe shutdown capability and therefore represents a configuration that is bounded by existing fire hazards analysis.

A dedicated mobile tanker staged inside the protected area to transfer fuel from the temporary storage tanks to the permanent day/skid tank system. The mobile tanker will have an integral transfer pump to facilitate movement of fuel to either of the two truck fills at the Unit 4 EDG building or day tank truck fills (auxiliary fill station) at the Unit 3 EDGs. One truck fill at the Unit 4 EDG building supplies fuel to the 4A and 4B storage tanks, the other truck fill at the Unit 4 EDG building can provide fuel directly to the Unit 3 day tanks. This fuel supply will provide continued operation for 7 days. The temporary storage and transfer system will not meet requirements for Seismic Category I or Class 1E.

The capability to operate an Unit 3 EDG for 7 days during the tank cleaning evolution will be assured by an approved plant procedure that controls the following:

- A minimum fuel supply of 3880 gallons from the Unit 3 day and skid tank. This provides 17 hours of operation.
- The extra fuel supply of 8600 gallons in the Unit 4 EDG tanks which will be transferred by using one of the installed Unit 4 transfer pumps. This provides an additional 38 hours of operation.
- Three temporary tanks containing a minimum fuel supply of 38,000 gallons. This fuel supply will provide continued operation for 7 days.

Consequently, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment will not change the physical plant or modes of plant operation defined in the Turkey Point Units 3 and 4 operating license. The change will not involve addition or modification of equipment for Unit 3 EDG fuel storage and transfer. The temporary fuel supply system provides a reliable means of performing the required fuel delivery support function for the Unit 3 EDGs.

Consequently, operation of either unit in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendment is designed to provide flexibility to schedule and perform required surveillance activities. Surveillance intervals or operating requirements are not changed by the proposal; only the method of fuel oil storage on a temporary basis for a single operable EDG is addressed. The proposed change will not alter the basis for any Technical Specification that is related to the establishment of, or maintenance of, a nuclear safety margin.

Consequently, operation of Turkey Point Units 3 and 4 in accordance with this proposed amendment would not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

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

NRC Project Director: Frederick J. Hebdon.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of amendment request: January 9, 1998.

Description of amendment request: The licensee proposed to change the Technical Specifications (TS) to allow the use of ZIRLO^{tm} fuel rod clad material.

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:

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

Implementation of ZIRLO^{tm} fuel rod cladding will have no impact on the probability or consequences of any Design Basis Event occurrences which were previously evaluated. The determination that fuel design limits are met will continue to be performed using NRC approved fuel performance analysis methodology. Changing to ZIRLO^{tm} fuel rod cladding poses no significant increase in the probability or consequences of any accident previously evaluated.

No new performance requirements are being imposed on any system or component in order to support implementation of ZIRLO^{tm} fuel rod cladding. Since the LOCA and Non-LOCA analysis results will remain within design limits, the inputs to the radiation dose analysis do not change. Therefore, the consequences to the public resulting from any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR) is not increased.

Fuel rod design criteria will be evaluated every cycle to ensure proper compliance with fuel rod design limits and therefore the UFSAR. The evaluation of the fuel design against fuel design limits will be performed in accordance with 10 CFR 50.59, which ensures that the reload will not involve an increase in the probability or consequence of an accident previously evaluated.

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

Implementation of ZIRLO^{tm} fuel rod cladding will have no impact, nor does it contribute in any way to the probability or consequences of an accident.

No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of using ZIRLO^{tm} fuel rod cladding. The institution of ZIRLO^{tm} fuel rod cladding will have no adverse effect on, and does not challenge the performance of, any safety related system.

The determination that the fuel rod design limits are met will be performed using NRC approved methodology. Therefore, the proposed amendment does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety is not affected by the implementation of ZIRLO[tm] fuel rod cladding. Use of ZIRLO[tm] fuel rod cladding has been approved by the NRC and does not constitute a significant reduction in the margin of safety.

The margin of safety provided in the fuel design limits is acceptable and will be maintained and not reduced.

In addition, each future reload will involve a 10 CFR 50.59 review to assure that operation of the units within the cycle specific limits will not involve a reduction in the margin of safety. Therefore, the proposed amendment does not significantly reduce the margin of safety.

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

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: January 22, 1998.

Description of amendment request: The amendment would incorporate the proposed revision into Chapter 9 of the Millstone Unit 3 Final Safety Analysis Report. The proposed revision to the Millstone Unit 3 licensing basis would accept the existing use of epoxy coatings on safety-related components.

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

NNECO has reviewed the proposed revision in accordance with 10CFR50.92 and has concluded that the revision does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not satisfied. The proposed revision does not involve [an] SHC because the revision would not:

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

Past experience indicates that failure of previous ARCOR applications may have degraded the performance of SWS [service water system] heat exchangers within one train, but there is no indication that failure of multiple heat exchangers on both trains is feasible. Furthermore, the likelihood of ARCOR material being released has been reduced by improving the application procedure and performing destructive testing to detect disbondment. In addition, the completion of normal heat exchanger performance surveillance's and periodic visual inspections minimizes the potential for disbonded ARCOR to degrade SWS components.

Therefore, the presence of ARCOR coating material within the SWS does not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The application of ARCOR material may lead to the degradation of SWS heat exchangers. However, multiple ARCOR application failures occurring simultaneously either instantaneously or gradually resulting in failure of all SWS heat exchangers in both trains is not considered feasible. An instantaneous failure is discounted by analysis which concludes that normal system operations are more likely to cause the release of degraded ARCOR than what might be expected following a seismic event. Gradual degradation is not expected since normal SWS heat exchanger performance surveillance's will identify heat exchanger tubesheet fouling and thus, provide early detection of coating failures. Therefore, the use of ARCOR coating material within the SWS does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Although the gradual release of ARCOR material creates the potential to simultaneously degrade the performance of mitigating equipment in both trains of safety systems, it is determined to be unrealistic due to normal heat exchanger performance surveillance's. These surveillance's are expected to identify heat exchanger tubesheet fouling and provide early detection and mitigation of a problem with the pipe coatings. Therefore, the application of ARCOR coating within the SWS does not involve a significant reduction in the margin of safety.

In conclusion, based on the information provided, it is determined that the proposed revision does not involve an SHC.

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

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

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

NRC Deputy Director: Phillip F. McKee.

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

Date of amendment request: July 26, 1996, as supplemented September 5 and December 4, 1997.

Description of amendment request: The proposed amendment would, as part of the licensee's power rerate program, increase the maximum power level to 1775 megawatts thermal (MWt). This change is approximately 6.3 percent above the current maximum power level of 1670 MWt.

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

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

The probability of occurrence and consequences of an [accident] previously evaluated have been evaluated for MNGP [Monticello Nuclear Generating Plant] Power Rerate. This evaluation has concluded that MNGP Power Rerate will not involve a significant increase in the probability of occurrence or consequences of previously evaluated accidents.

1. Evaluation of Accident Consequences

(a) ECCS–LOCA Analysis

The Emergency Core Cooling System Loss of Coolant Accident (ECCS–LOCA) performance analysis has been evaluated for MNGP Power Rerate using methodology which has been approved by the NRC for LOCA 10CFR50.46 analyses [requirements]. The current ECCS performance requirements were used in the power rerate analysis; no further parameter relaxations were included in the analysis. The ECCS–LOCA analysis was performed for MNGP Power Rerate for the existing licensed rated thermal power and at a bounding thermal power level of 1880 MWt that is approximately 6% greater than the proposed power rerate to 1775 MWt [megawatts thermal]. In addition, the bounding thermal power level was increased by an additional 2% in accordance with regulatory guidance. The licensing peak clad temperature for the bounding analyzed thermal power level remains below the 10CFR50.46 required limit of 2,200'F. Therefore the analysis demonstrates that MNGP will continue to comply with 10CFR50.46 and 10CFR50, Appendix K at rerated conditions thus the consequences of a LOCA is not significantly increased for the proposed power rerate.

(b) Abnormal Operating Transient Analysis

An evaluation of the Updated Safety Analysis Report (USAR) and reload transients has been performed for MNGP Power Rerate to demonstrate that the proposed power rerate has no adverse effect on plant safety. This evaluation was performed for a power level of 1775 MWt, with the exception that certain event evaluations were performed at 102% of the rerate power level. The transient analysis performed to demonstrate the acceptability of MNGP Power Rerate used the NRC approved methods identified in the MNGP Technical Specifications.

The limiting transient events at the power rerate conditions have been analyzed. This includes all events that establish the core thermal operating limits and the events that bound other transient acceptance criteria. These limiting transients were benchmarked against the existing rated thermal power level by performance of the event analysis at both the proposed rerate power level and the existing rated power level. In addition, an expanded group of transient events was evaluated to confirm that these events were less severe with the power rerate than the most limiting transients. The events included in the expanded group of transient events were chosen based on those events which have been demonstrated to be sensitive to initial power level. This evaluation confirmed that the existing set of limiting transient events remains valid for MNGP Power Rerate. The evaluation was performed for a representative core and demonstrated the overall capability to meet all transient safety criteria for the power rerate. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria and to establish cycle specific operating limits.

The results of the evaluation of transients demonstrate that the power rerate can be accomplished without a significant increase in the consequences of the transients evaluated. The fuel thermal-mechanical limits at the power rerate conditions are within the specific design criteria for the GE [General Electric] fuels currently loaded in the MNGP core. Also, the power-dependent and flow-dependent MCPR [minimum critical power ratio] and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) methods developed as part of the core performance improvement program remain applicable to rerate conditions. The transient event evaluation confirmed that MNGP Power Rerate has no significant effect on the power-dependent and flow-dependent MCPR and MAPLHGR limits. The peak reactor pressure vessel bottom head pressure remains within the ASME [American Society of Mechanical Engineers] requirement for reactor pressure vessel overpressure protection.

The effects of plant transients were evaluated by assessing a number of disturbances of process variables and malfunctions or failures of equipment consistent with USAR. The transient events were evaluated against the Safety Limit Minimum Critical Power Ratio, (SLMCPR). The SLMCPR is determined using NRC-approved methods. The limiting transient events are slightly more severe when initiated from the rerate power level. The power rerate transient evaluation results show a slightly more limiting event initial CPR [critical power ratio] (less than or equal to 0.02) than that initiated from the present rated power level for the near limiting transients. However, for the most limiting transient, the evaluation of a representative core showed that no change is required to the Operating Limit MCPR for the power rerate and that the integrity of the SLMCPR is maintained. The margin of safety established by the SLMCPR is not affected and the event consequences are not significantly affected by the proposed power rerate to 1775 MWt. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria and to establish cycle specific operating limits.

The results demonstrate that the MNGP core thermal power output can be safely increased to the power rerate level without significant effect on the consequences of previously evaluated postulated transient events. The results of the rerate transient analysis are summarized as follows.

(1) Events Resulting in a Nuclear System Pressure Increase

(a) Main Generator Load Rejection with No Steam Bypass

At rerated conditions, the fuel transient thermal and mechanical overpower results remain below the NRC accepted design criteria.

(b) Main Turbine Trip with No Steam Bypass

At rerate conditions, the fuel transient thermal and mechanical overpower results remain below the NRC accepted design criteria.

(c) Main Steam Isolation Valve Closure, Flux Scram

The peak reactor pressure vessel bottom head pressure for rerate conditions is slightly higher than the reactor pressure vessel bottom head pressure at current conditions. However, the resultant pressure is still below the ASME overpressure limit of 1,375 psig [pounds per square inch].

(d) Slow Closure of a Single Turbine Control Valve

The results of this transient for the power rerate remain non-limiting as compared with other more severe pressurization events.

(2) Event Resulting in a Reactor Vessel Water Temperature Decrease

(a) Feedwater Controller Failure-Maximum Demand

The delta CPR calculated for this event at rerate conditions is about 0.01 higher than the corresponding value for the current rated power when the impact of the new condensate pumps is factored in. The trend for the Feedwater Controller Failure-Maximum Demand event is consistent with the analysis for the current rated power. The fuel thermal margin results are within the acceptable limits for the fuel types analyzed.

(b) Loss of Feedwater Heating

This event at the rerate conditions remains significantly less than the cycle operating MCPR limit. The results at low core flow conditions are actually slightly higher than for the high core flow condition because of increased inlet coolant subcooling into the reactor core. The calculated thermal and mechanical overpower limits at the power rerate conditions for this event also meet the fuel design criteria.

(c) Inadvertent HPCI [high-pressure coolant injection] Actuation

For the limiting condition analyzed, both the high water level setpoint and the high reactor pressure vessel steam dome pressure scram setpoints are not reached. Based on the peak average fuel surface heat flux results, the HPCI actuation event will be bounded by the limiting pressurization event with respect to delta Critical Power Ratio ([delta] CPR) considerations. In addition, the fuel transient thermal and mechanical overpower limits remain within the NRC accepted design values.

(3) Event Resulting in a Positive Reactivity Insertion

(a) Rod Withdrawal Error (RWE)

The current Rod Block Monitor (RBM) system for MNGP with power dependent setpoints was analyzed for the rod withdrawal error event at the power rerate conditions using a statistical approach consistent with NRC approved methods. The analysis concluded that the transient is slightly more severe with a greater delta Critical Power Ratio ([delta] CPR) from the initial most limiting CPR. However, the fuel and mechanical overpower results remain within the NRC accepted design criteria.

(4) Event Resulting in a Reactor Vessel Coolant Inventory Decrease

(a) Pressure Regulator Failure to Full Open

The results of this transient for the power rerate remain non-limiting as compared with other more severe pressurization events.

(b) Loss of Feedwater Flow

This transient event does not pose any direct threat to the fuel in terms of a power increase from the initial conditions. Water level declines rapidly and a low level causes a reactor scram. The closure of the main steam isolation valves and the actuation of High Pressure Coolant Injection and Reactor Core Isolation Cooling terminate the event. This event was included in the power rerate evaluation to provide assurance that sufficient water makeup capability is available to keep the core covered when all normal feedwater is lost. The generic analysis performed in support of the extended power uprate program shows that at the power rerate conditions a large amount of water remains above the top of the active fuel. These sequences of events do not require any new operator actions or shorter operator response times. Therefore, the operator actions for the event do not significantly change for the power rerate.

(5) Event Resulting in a Core Coolant Flow Decrease

(a) Recirculation Pump Seizure

The recirculation pump seizure assumes instantaneous stoppage of the pump motor shaft of one recirculation pump. As a result, the core flow decreases rapidly. The heat flux decline lags core power and flow and could result in a degradation of core heat transfer. At the power rerate conditions, the transient results confirmed that the consequences of the pump seizure event remain non-limiting.

(6) Event Resulting in a Core Coolant Flow Increase

(a) Recirculation Flow Controller Failure Increasing Flow

The results of this transient for the power rerate remain non-limiting as compared with other more severe pressurization events.

(c) Design Basis Accident Challenges to the Containment

The primary containment response to the limiting design basis accident was evaluated for a bounding reactor power level approximately 6% greater than the proposed power rerate to 1775 MWt. In addition, the bounding reactor power level was increased by an additional 2% in accordance with regulatory guidance. The effect of the power rerate on the short term containment response (peak values) as well as the long term containment response for containment pressure and temperature confirms the suitability of the plant for operation at the bounding power level, thus the proposed power rerate to 1775 MWt is acceptable. Factors of safety provided in the ASME Code are maintained and safety margin is not affected for the power rerate to 1775 MWt.

Short-term containment response analyses were performed for the limiting design basis LOCA consisting of a double-ended guillotine break of a recirculation suction line, to demonstrate that operation at a bounding reactor power will not result in exceeding the containment design limits. This limiting design basis LOCA event results in the highest short-term containment pressures and dynamic loads. The analysis determined that for a bounding reactor power the maximum drywell pressure values are bounded by the current USAR analysis value and by the containment design pressure. The power rerate to 1775 MWt has no adverse effect on the containment structural design pressure.

Because there will be more residual heat with increased thermal power, the containment long term response will have slightly higher temperatures. Long term suppression chamber temperatures remain within the design temperature of the structure, thus factors of safety provided in the ASME code are maintained and safety margin is not affected. Analysis confirmed that ECCS pump NPSH is adequate for this temperature response. It was confirmed that the long term response does not adversely affect the containment structure or the environmental qualification (EQ) of equipment located in the drywell or suppression chamber room. The drywell long term temperature response is not adversely affected for a bounding reactor power. An analytical power level of 1880 MWt bounds the decay heat associated with the 1775 MWt power level with a one sided confidence interval of 95%. The containment long term response is therefore acceptable for the power rerate to 1775 MWt.

The impact of a reactor power increase on the containment dynamic loads have been determined, evaluated and found to have no adverse effects for conditions which well bound the proposed power rerate. Thus the containment dynamic loads were found to be acceptable for the power rerate to 1775 MWt.

The MNGP Power Rerate evaluation of the primary containment response to the design basis accident confirmed that the power rerate does not result in a significant increase in consequences for a bounding reactor power approximately 6% greater than the proposed power rerate to 1775 MWt.

(d) Radiological Consequences of Design Basis Accidents

For MNGP Power Rerate, the radiological consequences of the limiting design basis accidents were re-evaluated. These evaluations included the effect of the power rerate on the radiological consequences of accidents presented in USAR Section 14.7.

This evaluation was performed using inputs and evaluation techniques consistent with the current regulatory guidance, the current GE analysis methods, and the appropriate plant design basis. The inputs and analysis methods used for MNGP Power Rerate differ from those utilized in the current licensing basis evaluation presented in the USAR and the AEC [Atomic Energy Commission] safety evaluation supporting plant initial licensing. The MNGP Power Rerate evaluations used the more contemporary staff approved methods. The inputs used in the MNGP Power Rerate evaluation provide a conservative assessment of the potential radiological consequences. The conclusions of these evaluations are consistent with the original licensing basis evaluations. The radiological consequences of the limiting design basis accidents remain well within 10CFR100 guidelines for a bounding thermal power approximately 6% greater than the proposed power rerate of 1775 MWt. In addition the bounding thermal power level was increased by an additional 2% in accordance with regulatory guidance.

To conservatively analyze the change in consequences, the evaluation of radiological consequences using the analysis inputs and methods was performed for the existing licensed rated thermal power and a thermal power bounding the proposed power rerate. This provides a conservative bounding change in consequences for the requested power rerate to 1775 MWt.

The MNGP Power Rerate evaluation of the radiological consequences of design basis accidents confirmed that the power rerate does not result in a significant increase in consequences for a bounding power level approximately 6% greater than the proposed power rerate. The results remain below the 10CFR100 guideline values as well as the licensing basis established in the March 18, 1970 AEC safety evaluation. Therefore, the postulated radiological consequences do not represent a significant change in accident consequences and are clearly within the regulatory guidelines for the proposed power rerate to 1775 MWt.

(e) Other Evaluations

(1) Performance Improvements

The MNGP Power Rerate safety analysis has been performed taking into account the implementation of the following previously approved special operational features.

(a) Maximum Extended Load Line Limit/Increase Core Flow (MELLL/ICF)

The safety analysis for rerate conditions shows that the extended operating domain as analyzed by MELLL/ICF remains valid for the power rerate conditions.

(b) Average Power Range Monitor/Rod Block Monitor Technical Specification (ARTS) Improvements

The safety analysis for rerate conditions shows that the ARTS improvements remain valid for the power rerate conditions.

(c) Single Loop Operation (SLO)

The safety analysis for rerate conditions shows that the single loop operating mode remains valid for the power rerate conditions. The MELLLA trip setpoints determined for two-loop operation were confirmed to be acceptable for single loop operation with a correction applied to account for the actual effective drive flow applied when operating in single loop. The single loop settings have been conservatively established to be consistent with the two loop settings while ensuring the appropriate corrections are applied to the MAPLHGR and the operating limit MCPR to account for single loop operation.

(2) Effect of Power Rerate on Support Systems

An evaluation was performed to address the effect of MNGP Power Rerate on accident mitigation features, structures, systems, and components within the balance of plant. The results are as follows:

Auxiliary systems such as, building heating, Ventilation and Air Conditioning (HVAC) systems, reactor building closed cooling water, service water and emergency service water, spent fuel pool cooling, process auxiliaries such as instrument air and makeup water and the post-accident sampling system were confirmed to operate acceptably under normal and accident conditions at rerate conditions.

The secondary containment and standby gas treatment system were confirmed to be able to adequately contain, process, and control the release of normal and post-accident levels of radioactivity at rerate conditions.

Instrumentation was reviewed and confirmed to be capable of performing its control and monitoring functions under rerate conditions. As required, analyses were performed to determine the need for setpoint changes for various functions (*e.g.*, APRM [average power range monitor] neutron flux scram setpoints). In general, setpoints are to be changed only to maintain adequate difference between plant operating parameters and trip setpoints, while ensuring safety performance is demonstrated. The revised setpoints have been established using the NRC reviewed methodology as guidance.

Electric power systems including the turbine generator and switchgear components were verified as being capable of providing the electrical load as a result of the rerate power levels. An evaluation of the auxiliary power system for the power rerate conditions confirmed that the system has sufficient capacity with the changes identified in Exhibit I [of the 12/4/97 submittal] to support all required loads for safe shutdown, to maintain a safe shutdown condition, and to operate the required engineered safeguards equipment following postulated accidents. No safety-related electrical loads were affected which would adversely impact the emergency diesel generators.

Piping systems were evaluated for the effect of operation at higher power levels, including transient loading. The evaluation confirmed that, with few exceptions, piping and supports are adequate to accommodate the increased loading resulting from operation at rerate power conditions. In a few cases, piping supports will be modified to accept higher forces due to rerate conditions.

The effect of rerate conditions on high energy line break (HELB) was evaluated. The evaluation confirmed structures, systems, and components important to safety are capable of accommodating the effects of jet impingement and blowdown forces and the environmental effects resulting from HELB events at rerate conditions.

Control room habitability was evaluated. With the implementation of minor hardware and non-hardware changes to the control room ventilation system, Post-accident Control Room and Technical Support Center doses at rerate conditions were confirmed to be within the guidelines of General Design Criterion 19 of 10CFR50, Appendix A.

The environmental qualification of equipment important to safety was evaluated for the effect on normal and accident operating conditions at rerate power levels. The equipment remains qualified for the new conditions. Minor adjustments will reflect some changes to maintenance frequencies. The preventative maintenance program will continue to provide for equipment maintenance or replacement to ensure equipment environmental qualification at rerate power conditions.

(3) Effect on Special Events

The consequences of special events (*i.e.*, ATWS [anticipated transient without scram], 10CFR50, Appendix R, and Station Blackout) remain within NRC accepted criteria for rerate conditions. Concurrent malfunctions assumed to occur during accidents have been accounted for in the safety analyses for rerate conditions. The consequences of these equipment malfunctions does not change with implementation of the MNGP Power Rerate program. The generic ATWS analysis for operation at rerate conditions is being revised. The revision is not expected to affect MNGP compliance with NRC acceptance criteria.

(f) Conclusion

The evaluation of the Emergency Core Cooling System performance has demonstrated the criteria of 10CFR50.46 are satisfied, thus the margin of safety established by the criteria is maintained. The analysis demonstrated that the ECCS will function with the most limiting single failure to mitigate the consequences of the accidents and maintain fuel integrity. The system will continue to perform as required under rerate conditions to mitigate the consequences of accidents and thus the power rerate does not adversely affect ECCS performance in a manner to increase the severity of consequences. Challenges to the containment have been evaluated and the integrity of the fission product barrier has been confirmed. The radiological consequences of design basis accidents have been evaluated and it was found that the effect of the proposed power rerate on postulated radiological consequences does not result in a significant increase in accident consequences. These evaluations have been performed for a bounding reactor power approximately 6% greater than the proposed power rerate. In addition the bounding reactor power level was increased by an additional 2% in accordance with regulatory guidance. Thus the evaluations provide conservative bounding results for the proposed power rerate to 1775 MWt and demonstrate that the proposed power rerate does not result in significant increase in accident consequences.

The abnormal transients have been analyzed under the power rerate conditions, and the analysis has confirmed that the power rerate to 1775 MWt has only a minor effect on the minimum critical power ratio and that no change to the safety limit critical power ratio results, thus the margin of safety as assured by the safety limit critical power ratio is maintained. The effect of the power rerate on the consequences of abnormal transients which result from potential component malfunctions has been shown to be acceptable, thus the power rerate does not result in a significant increase in transient event consequences.

The spectrum of analyzed postulated accidents and transients has been investigated, and has been determined to meet the current regulatory criteria for the MNGP at rerate conditions. In the area of core design, the fuel operating limits will still be met at the rerate power level, and fuel reload analyses will show plant transients meet the criteria accepted by the NRC as specified in the plant Technical Specifications. The evaluation of transient and accident consequences was performed consistent with the proposed changes to the plant Technical Specifications. Therefore, the proposed Operating License and Technical Specification changes will not cause a significant increase in the consequences of an accident previously evaluated for the Monticello plant.

2. Evaluation of the Probability of Previously Evaluated Accidents

The proposed power rerate imposes only minor increases in the plant operating conditions. No changes are required to the rated core flow, rated reactor pressure, or turbine throttle pressure. The power rerate will result in moderate flow increases in those system[s] associated with the turbine cycle (*i.e.*, condensate, feedwater, main steam, etc.). For MNGP Power Rerate, the small increase in operating temperatures for balance of plant support systems has no significant effect on LOCA or other accident probabilities.

The increase in flow rates in balance of plant systems is addressed by compliance with NRC Generic Letter 89–08, ''Erosion/Corrosion in Piping.'' The MNGP Power Rerate evaluations have confirmed that the power rerate has no significant effect on flow induced erosion/corrosion. The worst case limiting feedwater and main steam piping flow increases were evaluated to be approximately proportional to the power increase. The affected systems are currently monitored by the MNGP Erosion/Corrosion program. Continued monitoring of the systems provides a high level of confidence in the integrity of potentially susceptible high energy piping systems.

The occurrence frequency of accident precursors and transients [has] been addressed when required by applying the guidance of NRC reviewed setpoint methodology to insure that acceptable trip avoidance is provided during operational transients subsequent to implementation of rerate. The setpoint evaluation has confirmed that MNGP Power Rerate does not result in any increase in challenges to the plant protective instrumentation.

Plant systems, components, and structures have been verified to be capable of performing their intended functions under rerate conditions with a few minor exceptions. Where necessary, some components will be modified prior to implementation of the MNGP Power Rerate Program to accommodate the revised operating conditions (*e.g.*, a limited number of pipe supports changes, instrumentation setpoint changes, control room habitability improvements). MNGP Power Rerate does not significantly affect the reliability of plant equipment. Where reliability effects have been identified, modifications and administrative controls will be implemented prior to the power rerate to adequately compensate. No new components or system interactions that could lead to an increase in accident probability are created due to the power rerate.

The probability (*i.e.*, frequency of occurrence) of design basis accidents occurring is not affected by the increased power level, as the applicable criteria established for plant equipment (*e.g.*, ANSI Standard B31.1, ASME Code,) will still be followed as the plant is operated at the rerate power level. The MNGP Power Rerate analysis basis assures that the power dependent margin prescribed by the Code of Federal Regulations (CFR) will be maintained by meeting the appropriate regulatory criteria. Similarly, factors of safety specified by application of the Code design rules have been demonstrated to be maintained, as have other margin-assuring acceptance criteria used to judge the acceptability of the plant. Reactor scram setpoints as established are such that there is no significant increase in scram frequency due to rerate conditions. No new challenges to safety-related equipment will result from the power rerate. Therefore, the proposed Operating License and Technical Specifications changes do not involve a significant increase in the probability of an accident previously evaluated.

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

The basic Boiling Water Reactor configuration, operation and event response is unchanged by the power rerate. Analysis of transient events has confirmed that the same transients remain limiting and that no transient events result in a new sequence of events which could lead to a new accident scenario. The MNGP Power Rerate analyses confirmed that the accident progression is basically unchanged by the power rerate.

An increase in power level will not create a new fission product release path, or result in a new fission product barrier failure mode. The same fission product barriers such as the fuel cladding, the reactor coolant pressure boundary and the reactor containment, remain in place. Fuel rod cladding integrity is ensured by operating within thermal, mechanical, and exposure design limits and is demonstrated by the MNGP Power Rerate transient analysis and accident analysis. Similarly, analysis of the reactor coolant pressure boundary and primary containment have demonstrated that the power rerate has no adverse effect on these fission product barriers. The proposed changes to the plant Technical Specifications to support the power rerate implementation are consistent with the MNGP Power Rerate analyses and assure transient and accident mitigation capability in compliance with regulatory requirements.

The effect of MNGP Power Rerate on plant equipment has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode resulting from the power rerate was identified. The full spectrum of accident considerations defined in the USAR have been evaluated and no new or different kind of accident resulting from the power rerate has been identified. MNGP Power Rerate uses already developed technology and applies it within the capabilities of already existing plant equipment in accordance with presently existing regulatory criteria which includes accepted codes, standards, and methods. GE has designed BWRs of higher power levels than the rerate power of any of the currently operating BWR fleet and no new power dependent accidents have been identified. In addition, MNGP Power Rerate does not create any new sequence of events or failure modes that lead to a new type of accident.

All actions to ensure that safety-related structures, systems, and components will remain within their design allowable values and ensure they can perform their intended functions under rerate conditions will be taken prior to implementation of the power rerate. MNGP Power Rerate does not increase challenges to or create any new challenge to safety-related equipment or other equipment whose failure could cause an accident. Plant modifications required to support implementation of MNGP Power Rerate will be made to existing systems (*e.g.*, a limited number of pipe supports, instrumentation setpoints, control room habitability improvements), rather than by adding new systems of a different design which might introduce new failure modes or accident sequences. The Technical Specification changes required to implement the power rerate require little change to the plant's configuration, and all changes have been evaluated and are acceptable.

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

C. The proposed Operating License changes do not involve a significant reduction in a margin of safety.

The accident analysis, as well as a majority of the plant specific evaluations performed in support of MNGP Power Rerate have been performed assuming a bounding steady state power level 112.6% of the existing licensed limit of 1670 MWt, and approximately 6% above the licensed maximum thermal power level of 1775 MWt proposed by MNGP Power Rerate. In addition, the bounding reactor power level was increased by an additional 2% in accordance with regulatory guidance when applicable for the evaluation of accidents and transients. For plant conditions associated with a bounding analysis power level, the analyses demonstrated operating margin to criteria establishing margins of safety, thus additional operating margin is demonstrated and assured for the proposed power rerate to 1775 MWt and added confidence is established in the integrity of criteria establishing margin to safety.

The cycle specific transient analysis, as well as the analysis to establish plant instrumentation set points have been performed assuming a plant steady state power level of 1775 MWt. This analysis approach was taken in order to demonstrate safety and equipment margins while ensuring appropriate cycle specific operating limits. The evaluation of transient events and instrument setpoints demonstrated operating margin to criteria establishing margins of safety for the proposed power rerate conditions.

The MNGP Power Rerate analysis basis assures that the power dependent safety margin assuring criteria prescribed by the Code of Federal Regulations (CFR) will be maintained by meeting the appropriate regulatory criteria. Similarly, factors of safety specified by application of the code design rules have been maintained, as have other margin-assuring acceptance criteria used to judge the acceptability of the plant.

1. Fuel Thermal Limits

No change is required in the basic fuel design to achieve the rerate power levels or to maintain the margins as discussed above. No increase in the allowable peak bundle power is requested for the power rerate. The abnormal transients have been evaluated under the power rerate conditions for a representative core configuration. The analysis has confirmed that the power rerate has no adverse effect on the operating limit Minimum Critical Power Ratio (MCPR) and that no change to the safety limit MCPR results, thus the margin of safety as assured by the safety limit MCPR is maintained. The fuel operating limits such as Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and the operating limit MCPR will still be met at the rerate power level. The MNGP Power Rerate analyses have confirmed the acceptability of these operating limits for the power rerate without an adverse effect on margins to safety. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria and to establish cycle specific operating limits.

2. Design Basis Accidents Challenges to Fuel

The evaluation of the Emergency Core Cooling System performance has demonstrated the criteria of 10CFR50.46 are satisfied, thus the margin of safety established by the criteria is maintained. This evaluation was performed for a bounding reactor power level approximately 6% greater than the proposed power rerate. In addition the bounding reactor power level was increased by an additional 2% in accordance with regulatory guidance. The analysis demonstrates that MNGP will continue to comply [with] the 10 CFR 50.46 at the rerate conditions and that the margin of safety established by the regulation is maintained for the proposed power rerate.

3. Design Basis Accident Challenges to Containment

The primary containment response to the limiting design basis accident was evaluated for a bounding reactor power level approximately 6% greater than the proposed power rerate to 1775 MWt. In addition, the bounding reactor power level was increased by an additional 2% in accordance with regulatory guidance. The effect of the power rerate on the short term containment response (peak values) as well as the long term containment response for containment pressure and temperature confirms the suitability of the plant for operation at the bounding power level, thus the proposed power rerate to 1775 MWt is acceptable. Factors of safety provided in the ASME Code are maintained and safety margin is not affected for the power rerate to 1775 MWt.

Short-term containment response analyses were performed for the limiting design basis LOCA consisting of a double-ended guillotine break of a recirculation suction line, to demonstrate that operation at a bounding reactor power will not result in exceeding the containment design limits. The analysis determined that for a bounding reactor power the maximum drywell pressure values are bounded by the current USAR analysis value and by the containment design pressure. The power rerate to 1775 MWt has no adverse effect on the containment structural design pressure.

Long term suppression chamber temperatures remain within the design temperature of the structure, thus factors of safety provided in the ASME code are maintained and safety margin is not affected. An analytical power level of 1880 MWt bounds the decay heat associated with the 1775 MWt power level with a one sided confidence interval of 95%. Analysis confirmed that ECCS pump NPSH is not adversely affected with this temperature response. It was confirmed that the long term response does not significantly affect the containment structure or the environmental qualification (EQ) of equipment located in the drywell or suppression chamber room.

The impact of a reactor power increase on the containment dynamic loads [has] been determined, evaluated and found to have no adverse effects for conditions which well bound the proposed power rerate. Thus the containment dynamic loads were found to be acceptable for the power rerate to 1775 MWt.

The MNGP Power Rerate evaluation of the primary containment response to the design basis accident confirmed that the power rerate does not result in a reduction in margins of safety for a bounding reactor power approximately 6% greater than the proposed power rerate to 1775 MWt.

4. Design Basis Accident Radiological Consequences

The Updated Safety Analysis Report (USAR) provides the radiological consequences for each of the design basis accidents. The magnitude of the potential consequences is dependent upon the quantity of fission products released to the environment, the atmospheric dispersion factors and the dose exposure pathways. For power rerate, the atmospheric dispersion factors and the dose exposure pathways do not change. Therefore, the only factor which will influence the magnitude of the consequences is the quantity of activity released to the environment. This quantity is a product of the activity released from the core and the transport mechanisms between the core and the effluent release point.

The radiological consequences of design basis accidents have been evaluated, and it was found that the consequences did not result in a significant increase in consequences for a bounding reactor power level approximately 6% greater than the proposed power rerate. In addition, the bounding reactor power level was increased by an additional 2% in accordance with regulatory guidance. The results remain below the 10CFR100 guideline values as well as the licensing basis established in the March 18, 1970 AEC safety evaluation. Therefore, the postulated radiological consequences are clearly within the regulatory guidelines and all radiological safety margins are maintained for the power rerate to 1775 MWt.

5. Transient Evaluations

The effects of plant transients were evaluated by assessing a number of disturbances of process variables and malfunctions or failures of equipment consistent with USAR. The transient events were evaluated against the Safety Limit Minimum Critical Power Ratio, (SLMCPR). The SLMCPR is determined using NRC-approved methods. The Power Rerate transient analyses were performed using the approved methodology specified in the plant Technical Specifications. The limiting transient events are slightly more severe when initiated from the rerate power level. The power rerate transient evaluation results show a slightly more limiting transient initial CPR (less than or equal to 0.02) than that initiated from the present rated power level for the near limiting transients. However, for the most limiting transient, the evaluation of a representative core showed that no change is required to the Operating Limit MCPR for the power rerate and that the integrity of the SLMCPR is maintained. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria and to establish cycle specific operating limits.

The fuel thermal-mechanical limits at the power rerate conditions are within the specific design criteria for the GE fuels currently loaded in the MNGP core. Also, the power-dependent and flow-dependent MCPR and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) methods developed as part of the core performance improvement program remain applicable to rerate conditions. The transient event evaluation confirmed that MNGP Power Rerate has no significant effect on the power-dependent and flow-dependent MCPR and MAPLHGR limits. The peak reactor pressure vessel bottom head pressure remains within the ASME requirement for reactor pressure vessel over pressure protection.

The margin of safety established by the SLMCPR is not affected by the proposed power rerate to 1775 MWt.

6. Technical Specification Changes

The Technical Specifications ensure that the plant and system performance parameters are maintained at the values assumed in the safety analysis. The Technical Specification (setpoints, trip settings, etc.) are selected such that the actual equipment is maintained equal to or conservative with respect to the inputs used in the safety analysis. Proper account is taken of inaccuracies introduced by instrument drift, instrument accuracy, and calibration accuracy. The Technical Specifications address equipment availability and limit equipment out-of-service to assure that the plant can be expected to have at least the complement of equipment available to deal with plant transients as that assumed in the safety analysis. The evaluations and analyses performed to demonstrate the acceptability of MNGP Power Rerate were performed using inputs consistent with the proposed changes to the plant Technical Specifications.

The events that form the Technical Specification Bases were evaluated for the power rerate conditions using inputs and initial conditions consistent with the proposed Technical Specification changes. Although some changes to the Technical Specifications are required for the power rerate, no NRC acceptance limit will be exceeded. Therefore, the margins of safety assured by safety limits and other Technical Specification limits will be maintained. The changes to the Technical Specification Bases proposed by this submittal are consistent with the evaluations which demonstrated acceptability of the power rerate.

7. Conclusion

The spectrum of postulated accidents, transients, and special events has been investigated and [has] been determined to meet the current regulatory criteria for the MNGP at the power rerate conditions. In the area of core design, the fuel operating limits will still be met at the rerate power level, and fuel reload analyses will show plant transients meet the criteria accepted by the NRC as specified in the plant Technical Specifications. Challenges to fuel or ECCS performance were evaluated and shown to meet the criteria of 10 CFR 50.46 and 10 CFR 50, Appendix K. Challenges to the containment have been evaluated and the integrity of the fission product barrier has been confirmed. Radiological release events have been evaluated and shown to meet the guidelines of 10 CFR 100. The proposed Operating License and Technical Specification changes are consistent with the MNGP Power Rerate evaluation performed. The evaluations demonstrated compliance with the margin assuring acceptance criteria contained in applicable codes and regulations. Therefore, the proposed Operating License and Technical Specifications changes will not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Cynthia A. Carpenter

Philadelphia Electric Company, Docket No. 50–352, Limerick Generating Station (LGS), Unit 1, Montgomery County, Pennsylvania

Date of amendment request: February 9, 1998.

Description of amendment request: The amendment request proposes to revise the LGS, Unit 1 Technical Specifications (TS) Section 2.1 and its associated TS Basis to reflect the change in the minimum critical power ratio (MCPR) safety limit due to the plant-specific evaluation performed by General Electric Company (GE) for LGS, Unit 1, Cycle 8.

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

The revised MCPR Safety Limits for LGS Unit 1 Technical Specifications, and their use to determine cycle-specific thermal limits, have been calculated using NRC-approved methods (*i.e.*, GESTAR–II, Rev. 13) and are based on LGS Unit 1 Cycle 8 specific inputs. The use of these methods assures that the [safety limit for minimum critical power ratio] SLMCPR value is within the existing design and licensing basis, and cannot increase the probability or severity of an accident.

The basis of the MCPR Safety Limit calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The MCPR Safety limit preserves the existing margin to transition boiling and fuel damage in the event of a postulated accident. The probability of fuel damage is not increased.

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

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

The MCPR Safety Limit is a Technical Specification numerical value designed to ensure that fuel damage from transition boiling does not occur as a result of the limiting postulated accident. The MCPR Safety Limit is not an accident initiator; therefore, it cannot create the possibility of any new type of accident. The new MCPR Safety Limits are calculated using NRC-approved methods (*i.e.*, GESTAR–II, Rev. 13) and are based on LGS Unit 1, Cycle 8 specific inputs.

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

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

The margin of safety as defined in the TS Bases will remain the same. The new MCPR Safety Limits are calculated using NRC-approved methods (*i.e.*, GESTAR–II, Rev. 13), which are in accordance with the current fuel design and licensing criteria, and are based on LGS Unit 1 Cycle 8 specific inputs. The MCPR Safety Limit remains high enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity.

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

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

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

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, PA 19101.

NRC Project Director: John F. Stolz.

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

Date of amendment request: October 14, 1997.

Description of amendment request: The proposed changes would correct the maximum exposure dependent, infinite lattice multiplication factor for fuel bundles and provide for installation of additional storage racks to increase spent fuel capacity.

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

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

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

A change in the infinite lattice neutron multiplication factor for a fuel bundle in the reactor core geometry which ensures the criticality limit for fuel in the spent fuel pool [SFP] geometry is met does not affect initiation of any accident.

Operation in accordance with the revised limit ensures the consequences of previously analyzed accidents are not changed. Storage of additional fuel assemblies in the pool does not affect the probability or consequences of dropping a fuel assembly, since this accident is localized to a small area of the storage array. Likewise, addition of specifications containing details presently in plant design documents and editorial changes do not change the probability or consequences of a previously analyzed accident.

2. Create the possibility of a new or different kind of accident for any accident previously evaluated because:

A change in the infinite lattice neutron multiplication factor for a fuel bundle in the reactor core geometry which ensures the criticality limit for fuel in the spent fuel pool geometry is met does not affect the types of reactivity accidents which may occur. Therefore changing the limit will not [create the possibility of] a new or different type of accident. Maintenance of available decay heat removal systems ensures that no new type of loss of cooling accident associated with the SFP will occur as a result of storing additional irradiated fuel assemblies. Likewise, addition of specifications containing details presently in plant design documents and editorial changes do not create the possibility of a new or different type of accident.

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

The revised limit on infinite lattice neutron multiplication factor for a fuel bundle in the reactor core geometry ensures maintenance of the same margin of safety with respect to criticality as presently exists for storage of fuel in the SFP. Storing additional irradiated fuel assemblies in the pool does not affect the margin of safety with regard to pool cooling since sufficient heat removal systems will be maintained available to ensure maintenance of acceptable pool temperatures. Addition of specifications containing details presently in other design documents and editorial changes have no effect on the margin of safety.

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

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

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

NRC Project Director: S. Singh Bajwa.

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

Date of amendment request: February 9, 1998.

Description of amendment request: The proposed amendment would revise the Virgil C. Summer Nuclear Station Technical Specifications (TS) to remove emergency diesel generator (1) accelerated testing requirements (TS 3/4.8.1, Table 4.8–1), and (2) special reporting requirements (TS Surveillance Requirement 4.8.1.1.3) in accordance with NRC Generic Letter (GL) 94–01, ''Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators.''

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.

This change will provide flexibility to structure the emergency diesel generator maintenance program based on the risk significance of the structures, systems, and components that are within the scope of the maintenance rule. The removal of the diesel generator accelerated testing is acceptable as the maintenance rule applies system and train specific performance criteria to monitor diesel generator performance. These criteria include a running availability and reliability measure. The performance criteria for the diesel generator reliability and unavailability established by the maintenance rule, and the causal determinations and corrective actions required for functional failures and/or exceeding performance criteria, is considered to be an acceptable method for monitoring diesel generator performance.

As the diesel generator performance will [continue] to be assured by the maintenance rule, the proposed changes do not affect any of the initiators for an accident previously evaluated. The changes do not impact the diesel's design sources, operating characteristics, system functions, or system interrelationships. The failure mechanisms for the accidents previously analyzed are not affected, and no additional failure modes are created that could cause an accident previously evaluated. Since the changes are administrative in nature, and the diesel generator performance and reliability will continue to be assured by the maintenance rule, the proposed changes cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to the initiation of any accidents. The proposed changes only affect the methods used to monitor and assure diesel generator performance. The performance criteria for both the diesel generator reliability and unavailability established by the maintenance rule, and the causal determinations and corrective actions required for functional failures and/or exceeding performance criteria, is considered by GL 94–01 to be an acceptable method for monitoring diesel generator performance.

No SSC [structure, system, or component], method of operating, or system interface is altered by this change. The changes do not impact the diesel's design sources, operating characteristics, system functions, or system interrelationships. The failure mechanisms for the accidents are not affected, and no additional failure modes are created. Because the proposed changes are administrative in nature, and the diesel generator performance and reliability will continue to be assured by the maintenance rule, the proposed changes cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes only affect the methods used to monitor and assure diesel generator performance. The performance criteria for both the diesel generator reliability and unavailability established by the maintenance rule, and the causal determinations and corrective actions required for functional failures and/or exceeding performance criteria, is considered by GL 94–01 to be an acceptable method for monitoring diesel generator performance. No margin [of] safety as defined in the basis for any technical specification is impacted by these changes. This change does not impact any uncertainty in the design, construction, or operation of any SSC.

Diesel generator response to accident initiators is unchanged. No SSC, method of operating, or system interface is altered by this change. The changes do not impact the diesel's design sources, operating characteristics, system functions, or system interrelationships. Because the proposed changes are administrative in nature, and the diesel generator performance and reliability will continue to be assured by the maintenance rule, the proposed changes cannot involve a significant reduction in the margin [of] safety.

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

Local Public Document Room location: 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 Project Director: William M. Dean.

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

Date of amendment request: January 28, 1998.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Sections 6.3 and 6.12 to reflect the merger of the positions of Superintendent Radiation Protection and Superintendent Chemistry into one new position, Manager Chemistry/Radiation Protection.

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

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

The proposed change does not involve a significant increase in the probability of consequences of an accident previously evaluated. These changes involve administrative changes to the WCNOC organization.

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 does not create the possibility of a new or different kind of accident from any accident previously evaluated. This change is administrative in nature and does not involve a change to the installed plant systems or the overall operating philosophy of Wolf Creek Generating Station.

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

The proposed change does not involve a significant reduction in a margin of safety. This change does not involve any changes in overall organizational commitments and will not affect qualification requirements of any unit staff personnel. A position and title change alone does not reduce the margin of safety.

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

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

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

NRC Project Director: William H. Bateman.

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

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

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

Northern States Power Company, Docket No. 50–282, Prairie Island Nuclear Generating Plant, Unit 1, Goodhue County, Minnesota

Date of amendment request: January 15, 1998.

Description of amendment request: The proposed amendment would initiate a one-time only change for Prairie Island Unit 1 Cycle 19 that would allow the use of the moveable incore detector system for measurement of the core peaking factors with less than 75% and greater than or equal to 50% of the detector thimbles available.

Date of individual notice in the **Federal Register**: January 30, 1998 (63 FR 4676).

Expiration date of individual notice: March 2, 1998.

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

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

NRC Project Director: Cynthia A. Carpenter.

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.

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

Date of application for amendment: May 16, 1997, as supplemented November 14, 1997.

Brief description of amendment: The amendment involves replacing the service water (SRW) heater exchangers with new plate and frame heat exchangers (PHEs), having increased thermal performance capability. The Saltwater (SW) and SRW piping configuration will be modified as necessary to allow proper fit-up to the new components. A flow control scheme to throttle saltwater flow to the heat exchangers and the associated bypass lines will be added. Saltwater strainers with an automatic flushing arrangement will be added upstream of each heat exchanger. The majority of the physical work associated with this modification is restricted to the SRW pump room. The amendment is partially denied to the extent that the licensee is not authorized to operate with one PHE secured, and removing one containment air cooler from service to enable the affected subsystem to remain operable while the one PHE is secured.

Date of issuance: February 10, 1998.

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

Amendment No.: 225.

Facility Operating License No. DPR–53: Amendment revised the Updated Final Safety Analysis Report.

Date of initial notice in **Federal Register**: June 18, 1997 (62 FR 33118).

The November 14, 1997, letter provided clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 10, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

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

Date of application for amendments: November 6, 1997, as supplemented by letter dated January 28, 1998.

Brief Description of amendments: The amendments to Technical Specification (TS) Limiting Conditions for Operation (LCO) 3.3.5.5, Instrumentation for Control Room Emergency Ventilation System (CREVS) and 3.7.2, Control Room Emergency Ventilation System, and associated Bases for the Brunswick Steam Electric Plant (BSEP) Units 1 and 2 will be limited in duration (approximately 3 months) and will allow operation of both BSEP units to continue while upgrades to the control building ventilation system, including new air conditioning (AC) units and improved ductwork supports, are being installed. Part of the planned work requires opening the ductwork at the evaporative (i.e. cooling) coils. Temporary barriers will be constructed to preserve the leakage integrity of the control room pressure boundary; however, the temporary barriers will not be seismically qualified. While the permanent AC units are out of service, temporary AC units will be utilized. During the upgrade installation, the AC for the control room will not be protected from certain external events (e.g., seismic events, environmental hazards such as tornadoes and hurricanes, radiological sabotage, and missile hazards), as required by the system design and licensing basis, and will not fully meet single failure criteria.

Date of issuance: February 6, 1998.

Effective date: February 6, 1998.

Amendment Nos.: 191 and 222.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments authorize changes to the facility's Technical Specifications.

Date of initial notice in **Federal Register**: December 3, 1997 (62 FR 63973).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 6, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 28, 1997. Information related to the proposed restoration of the primary coolant dose equivalent iodine-131 (DEI) to their original licensing basis had been previously submitted in Commonwealth Edison Company's (ComEd) letter dated November 13, 1996, which was supplemented in subsequent letters dated March 20, June 24, August 19 and November 3, 1997.

Brief description of amendments: The amendments revise the technical specifications (TS) to reflect the forthcoming replacement of the original steam generators (OSG) in Byron, Unit 1, and Braidwood, Unit 1, which are Westinghouse Model D4 steam generators (SG), with the replacement steam generators (RSG) which are Babcock and Wilcox, International (BWI) SG. The present revisions to the TS remove the interim plugging criteria (IPC) related to outer diameter stress corrosion cracking (ODSCC) in the OSG as well as the F* alternative repair criteria and two separate SG tube sleeving methodologies which are not needed for the RSG.

Date of issuance: February 3, 1998

Effective date: This license amendment is effective as of the date of its issuance and shall be implemented in the first operating cycle after installation of the BWI replacement steam generators

Amendment Nos.: 101, 101, 92 and 92.

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**: December 17, 1997 (62 FR 66134). The November 13, 1996, and March 20, June 24, August 19 and November 3, 1997, submittals provided 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 February 3, 1998.

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.

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: November 6, 1995, and March 11, 1996, as supplemented June 5, 1997. The June 5, 1997, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the December 20, 1995, and April 10, 1996, **Federal Register** notices.

Brief description of amendments: These amendments revise the alarm setpoints for the effluent radiation and in-containment area radiation monitors listed in Technical Specification (TS) Table 3.3–6. These revisions make these alarm setpoints consistent with criteria for the Emergency Action Levels (EALs) approved by the Nuclear Regulatory Commission in August 1994. The EALs use these monitors as an indication of fission product barrier challenges or failures. These amendments also revise Action Statement 36 of TS Table 3.3–6 to reflect a previously approved change (License Amendment Nos. 188 and 70) in reporting frequency (change from semi-annual to annual) for effluent releases. The revision to Action Statement 36 makes it consistent with the previously approved change. These amendments include several editorial changes to the TSs which do not change the intent of the TSs.

Date of issuance: February 9, 1998.

Effective date: Both units, as of date of issuance, to be implemented within 60 days.

Amendment Nos.: 211 and 89.

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

Dates of initial notice in **Federal Register**: December 20, 1995 (60 FR 65677) and April 10, 1996 (61 FR 15988). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 9, 1998.

No significant hazards consideration comments received: No.

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

Duquesne Light Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of application for amendment: November 4, 1997.

Brief description of amendment: The amendment revises Item 6.a.2, ''4.16kV Emergency Bus (Start Diesel),'' of Table 3.3–4 of Technical Specification 3.3.2.1. The change reduces the trip setpoint for starting the emergency diesel generators on emergency bus undervoltage from a trip setpoint of greater than or equal to 83 percent with a 12-cycle delay time to a setpoint of greater than or equal to 75 percent of nominal bus voltage with a time delay of less than 0.9 seconds including auxiliary relay times. The amendment also revises the allowable value from greater than or equal to 81 percent of nominal bus voltage to greater than or equal to 74 percent of nominal bus voltage with a time delay of less than 0.9 seconds including auxiliary relay times.

Date of issuance: February 11, 1998.

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

Amendment No: 212.

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

Date of initial notice in **Federal Register**: December 3, 1997 (62 FR 63976).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 11, 1998.

No significant hazards consideration comments received: No.

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

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: August 6, 1997.

Brief description of amendment: The amendment eliminated the provisions in Technical Specification 3.8.1, ''AC Sources—Operating,'' for accelerated testing of the emergency diesel generators (DG). The changes are the following: (1) the frequency of verifying DG starts and operation in Surveillance Requirements (SRs) 3.8.1.2 and 3.8.1.3, respectively, are changed to 31 days, from the present reference to Table 3.8.1–1, and (2) Table 3.8.1–1, ''Diesel Generator Test Schedule,'' is deleted. The emergency diesel generators provide emergency AC power to the site with the loss of offsite AC power.

Date of issuance: February 9, 1998.

Effective date: February 9, 1998.

Amendment No: 134.

Facility Operating License No. NPF–29: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: September 24, 1997 (62 FR 50003).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 9, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120.

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

Date of application for amendment: September 3, 1997.

Brief description of amendment: The amendment authorizes Northeast Nuclear Energy Company, through a license condition, to incorporate changes to the description of the facility in the Updated Final Safety Analysis Report (UFSAR). This change revises the UFSAR by modifying the operation of the onsite emergency diesel generators and their associated fuel oil supplies.

Date of issuance: January 23, 1998.

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

Amendment No.: 212.

Facility Operating License No. DPR–65: Amendment revised the Updated Final Safety Analysis Report.

Date of initial notice in **Federal Register**: September 24, 1997 (62 FR 50009).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 23, 1998.

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.

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

Date of application for amendments: October 24, 1997.

Brief description of amendments: The amendments revise the containment hydrogen analyzer Technical Specifications (TSs) surveillance requirements of TS 4.6.4.1 to increase the calibration frequency from once per refueling outage to quarterly.

Date of issuance: January 29, 1998.

Effective date: As of the date of issuance.

Amendment Nos. 204 and 186.

Facility Operating License Nos. DPR–70 and DPR–75. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 17, 1997 (62 FR 66140).

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, NJ 08079.

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

Date of amendments request: June 13, 1997, as supplemented by letter dated January 7, 1998.

Brief Description of amendments: The amendments change Technical Specification (TS) 3.9.13 by adding a footnote to clarify the required electrical power sources for the penetration room filtration system when it is aligned to the spent fuel pool room during refueling operations. In addition, the associated Bases section of the TS will be modified to provide additional details concerning the proposed TS change.

Date of issuance: February 5, 1998.

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

Amendment Nos.: Unit 1–134; Unit 2–126.

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

Date of initial notice in **Federal Register**: July 16, 1997 (62 FR 38138).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 5, 1998.

No significant hazards consideration comments received: No.

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

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

Date of amendments request: October 16, 1997.

Brief Description of amendments: The amendments change the Farley Units 1 and 2 TS by revising the number of allowable charging pumps capable of injecting into the reactor coolant system (RCS) when the temperature of one or more of the RCS cold legs is equal to or less than 180° F.

Date of issuance: February 5, 1998.

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

Amendment Nos.: Unit 1–135; Unit 2–127.

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

Date of initial notice in **Federal Register**: December 3, 1997 (62 FR 63983).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 5, 1998.

No significant hazards consideration comments received: No.

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

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of application for amendment: October 28, 1996, as supplemented by letters dated August 19, 1997, and October 16, 1997.

Brief description of amendment: This amendment revises TS Section 3/4.8.1, ''A.C. Sources,'' TS Section 3/4.8.2, ''Onsite Power Distribution Systems,'' TS Table 4.8.1, ''Battery Surveillance Requirements,'' and the associated bases. Surveillance requirements have been modified to account for the increase in the fuel cycle. Administrative changes were also made.

Date of issuance: February 3, 1998.

Effective date: February 3, 1998.

Amendment No.: 219.

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

Date of initial notice in **Federal Register**: January 2, 1997 (62 FR 132).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 3, 1998.

No significant hazards consideration comments received: No. The supplemental information provided by the Licensees did not affect the proposed no significant hazards consideration determination.

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

Dated at Rockville, Maryland, this 18th day of February 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–4620 Filed 2–24–98; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Issuer Delisting; Notice of Application To Withdraw From Listing and Registration; (Complete Management, Inc., Common Shares, $.001 Par Value; 8% Convertible Subordinated Debentures Due 2003; 8% Convertible Subordinated Debentures Due December 15, 2003) File No. 1–12848

February 17, 1998.

Complete Management, Inc. (''Company'') has filed an application with the Securities and Exchange Commission (''Commission''), pursuant to Section 12(d) of the Securities Exchange Act of 1934 (''Act'') and Rule 12d2–2(d) promulgated thereunder, to withdraw the above specified securities (''Securities'') from listing and registration on the American Stock Exchange (''Amex'' or ''Exchange'').

The reasons cited in the application for withdrawing the Securities from listing and registration include the following:

The Securities also are listed for trading on the New York Stock Exchange, Inc. (''NYSE'') pursuant to a Registration Statement on Form 8–A that became effective on September 5, 1997. Trading in the Securities on the NYSE commenced at the opening of business on September 8, 1997.

The Company has complied with Amex Rule 18 by filing with the Exchange a certified copy of the regulations adopted by the Company's Board of Directors authorizing the withdrawal of the Securities from listing and registration on the Amex, and by