NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50–280 and 50–281]

Virginia Electric and Power Company Surry Nuclear Power Station; Exemption

I

The Virginia Electric and Power Company (VEPCO, the licensee) is the holder of Facility Operating License No. DPR–32 and Facility Operating License No. DPR–37, which authorize operation of the Surry Nuclear Power Station, Units 1 and 2. The licenses provide that the licensee is subject to all rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (NRC or the Commission) now or hereafter in effect.

The facility consists of two pressurized-water reactors at the licensee's site located in Surry County, Virginia.

II

The Code of Federal Regulations at 10 CFR 70.24, ''Criticality Accident Requirements,'' requires that each licensee authorized to possess special nuclear material shall maintain a criticality accident monitoring system in each area in which such material is handled, used, or stored. Sections 70.24 (a)(1) and (a)(2) specify detection and sensitivity requirements that these monitors must meet. Section 70.24(a)(1) also specifies that all areas subject to criticality accident monitoring must be covered by two detectors. Section 70.24(a)(3) requires licensees to maintain emergency procedures for each area in which this licensed special nuclear material is handled, used, or stored, and provides (1) that the procedures ensure that all personnel withdraw to an area of safety upon the sounding of a criticality accident monitor alarm, (2) that the procedures must include drills to familiarize personnel with the evacuation plan, and (3) that the procedures designate responsible individuals for determining the cause of the alarm and placement of radiation survey instruments in accessible locations for use in such an emergency. Section 70.24(b)(1) requires licensees to have a means by which to quickly identify personnel who have received a dose of 10 rads or more. Section 70.24(b)(2) requires licensees to maintain personnel decontamination facilities, to maintain arrangements for a physician and other medical personnel qualified to handle radiation emergencies, and to maintain arrangements for the transportation of contaminated individuals to treatment facilities outside the site boundary. Section 70.24(c) exempts Part 50 licensees from the requirements of 10 CFR 70.24(c) for special nuclear material used or to be used in the reactor. Subsection 70.24(d) states that any licensee who believes that there is good cause why he should be granted an exemption from all or part of 10 CFR 70.24 may apply to the Commission for such an exemption and shall specify the reasons for the relief requested.

III

By letter dated January 27, 1997, as supplemented March 24, 1997, VEPCO requested an exemption from 10 CFR 70.24(a). The Commission has reviewed the licensee's submittal and has determined that inadvertent criticality is not likely to occur in special nuclear materials handling or storage areas at Surry Nuclear Station, Units 1 and 2. The quantity of special nuclear material other than fuel that is stored on site is small enough to preclude achieving a critical mass.

The purpose of the criticality monitors required by 10 CFR 70.24 is to ensure that if a criticality were to occur during the handling of special nuclear material, personnel would be alerted to that fact and would take appropriate action. Although the staff has determined that such an accident is not likely to occur, the licensee has radiation monitors, as required by General Design Criteria 63, in fuel storage and handling areas. These monitors will alert personnel to excessive radiation levels and allow them to initiate appropriate safety actions. The low probability of an inadvertent criticality together with the licensee's adherence to General Design Criterion 63 constitute good cause for granting an exemption to the requirements of 10 CFR 70.24(a).

IV

The Commission has determined that, pursuant to 10 CFR 70.14, this exemption is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest; therefore, the Commission hereby grants the following exemption:

The Virginia Electric and Power Company is exempt from the requirements of 10 CFR 70.24(a) for the Surry Nuclear Power Station, Unit 1 and Unit 2.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will have no significant impact on the quality of the human environment (62 FR 44495).

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 21st day of August 1997.

For the Nuclear Regulatory Commission.

Samuel J. Collins,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 97–22779 Filed 8–26–97; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Biweekly Notice

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

I. Background

Pursuant to 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 August 4, 1997, through August 15, 1997. The last biweekly notice was published on August 13, 1997 (62 FR 43365).

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

By September 26, 1997, 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: Docketing and Services Branch, 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.

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 amendments request: June 12, 1997

Description of amendments request: The proposed amendments would revise the Limiting Condition for Operation (LCO) of Technical Specification 3.6.1.6 to limit drywell average air temperature instead of primary containment average air temperature, which is the volume-weighted average of both drywell and wetwell atmospheres. This change in monitored parameter is consistent with the approach taken in the improved standard technical specifications for boiling water reactor (BWR) plants of this type (NUREG-1433, Rev. 1, ''Standard Technical Specifications General Electric Plants, BWR/4,'' April 1995). The proposed amendments would additionally change the temperature limit in this LCO from 135°F (primary containment average air temperature) to 150°F (drywell average air temperature).

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

The NRC has provided standards in 10 CFR 50.92 for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Carolina Power & Light Company has reviewed these proposed license amendment requests and has concluded that their adoption would not involve a significant hazards consideration. The basis for this determination follows.

1. The probability of previously evaluated accidents is not a function of the ambient drywell air temperature. The revised drywell average air temperature limit of 150°F does not affect any instrumentation setpoints or allowable values, so [the] likelihood of plant instrumentation initiating a plant transient or accident has not been increased.

The design basis accidents were re-evaluated using an initial drywell air temperature of 150°F. The evaluation results indicate that no containment design requirements are exceeded nor are any regulatory requirements exceeded. Analyses demonstrate that an initial drywell average air temperature of 150°F will ensure that the safety analysis remains valid by ensuring that the peak loss-of-coolant accident drywell temperature does not result in the drywell structure exceeding the maximum allowable temperature of 300°F. Indeed, these evaluations indicate that both the peak drywell pressure and temperature will be slightly less than the peak drywell pressure and temperature resulting from the current 135°F primary containment air temperature limit. Since the drywell temperature and pressure associated with a postulated design basis accident remain less than the drywell maximum design allowable values, revised drywell average air temperature limit of 150°F does not increase the consequences of an accident previously evaluated.

A temporary, one-time exception footnote for the Brunswick Steam Electric Plant (BSEP), Unit No. 2 is being deleted because the period of the footnote's applicability expired on August 15, 1985. Deletion of this footnote is an administrative change that has no effect on the probability or consequences of an accident previously evaluated.

Thus, based on the above, the proposed license amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated. Revising the primary containment temperature limit basis to use the drywell average air temperature and increasing the average air temperature limit from 135°F to 150°F does not physically modify the facility nor does the proposed revision modify the operation of any existing plant equipment. A temporary, one-time exception footnote for BSEP Unit No. 2 is being deleted because the period of the footnote's applicability expired on August 15, 1985. Deletion of this footnote is an administrative change that does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license amendments do not involve a significant reduction in a margin of safety. The drywell average airspace temperature affects the calculated containment response to postulated Design Basis Accidents. Analyses demonstrate that an initial drywell average air temperature of 150°F will ensure that the safety analysis remains valid by ensuring that the peak loss-of-coolant accident drywell air temperature does not result in the drywell structure exceeding the maximum allowable temperature of 300°F. Analyses performed using an initial drywell average air temperature of 150°F also demonstrate that containment design requirements for peak post-accident suppression pool temperature, design basis accident related discharge loads for safety-relief valve piping, and net positive suction head for residual heat removal system and core spray system pumps are met. In addition, setpoints for reactor water level instrumentation located in the drywell have not been adversely affected, drywell equipment environmental qualification is being maintained, and containment performance during a postulated station blackout is not being adversely affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety. The deletion of a temporary, one-time exception footnote for BSEP Unit No. 2 is an administrative change that also 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: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297

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

NRC Project Director: Gordon E. Edison (Acting)

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 amendments request: July 18, 1997Description of amendments request: The proposed amendments would revise two specifications included in the Design Features section of the Technical Specifications (TS). The value for primary containment suppression chamber design temperature (TS 5.2.2.b) would be increased from 200°F to 220°F. The licensee has determined that the original suppression chamber design temperature was 220°F and confirmed that it is still the correct design value. Secondly, the specification for reactor coolant system volume (TS 5.4.2) would be redefined as the vessel volume, rather than the vessel and recirculation system volume, resulting in a change in the associated value from 18,670 cubic feet to 18,320 cubic feet. Additionally, the proposed amendments would correct a typographical error in Design Features TS 5.3.2 regarding the reactor core control rod 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:

10 CFR 50.92 provides standards for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Carolina Power & Light Company has reviewed these proposed license amendment requests and has concluded that their adoption would not involve a significant hazards consideration. The basis for this determination follows.

1. The proposed license amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed amendments correct an inaccurate suppression chamber design temperature to reflect the actual design temperature used during containment analyses and pressure vessel procurement, correct a typographical error, and update the reactor coolant system volume to reflect a more accurate volume used in current analyses. These changes are administrative in nature and do not affect the probability or consequences of any accident previously analyzed.

2. The proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. These changes are administrative in nature and correct the Technical Specifications to accurately represent information used during existing accident analyses. These changes do not introduce a new initiating event and do not create the possibility of a new or different kind of accident previously evaluated.

3. The proposed license amendments do not involve a significant reduction in a margin of safety. As stated above, these changes are administrative in nature and correct the Technical Specifications to accurately represent information used during existing accident analyses. These changes document values currently used in existing accident analyses and, therefore, do not reduce the margin of safety already established by the analyses.

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 North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297

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

NRC Project Director: Gordon E. Edison (Acting)

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

Date of amendment request: July 1, 1997

Description of amendment request: The proposed amendments would revise Technical Specification Table 3.3.7.1-1, ''Radiation Monitoring Instrumentation,'' to require two channels to be operable per trip system as opposed to two per intake. This change reflects a modification to the design of the instrument logic to satisfy single failure requirements. The amendment would also revise the associated action statement to clarify system logic wording.

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

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

The proposed Technical Specification (TS) change clearly defines the system logic and the specific actions required for system operability. It will not change the probability of occurrence of any accidents, because the affected radiation monitoring instrumentation is not an accident initiator. UFSAR [Updated Final Safety Analysis Report] Section 15.9.3.4 analyzed the effects of the loss of ventilation from the Main Control Room in the event of a Station Black Out (SBO). The scope of work for the design change associated with this TS change does not affect this analysis or any of its assumptions The consequences of an accident will not increase, because the trip system redundancy is being restored to meet design basis requirements. The proposed design change will eliminate the potential of exposing main control room personnel to radiation doses that exceed the limits specified in General Design Criteria (GDC) 19. The design change associated with this TS change will comply with the redundancy due to two trip systems, either of which will actuate the control room emergency makeup train as required and the potential for spurious actuations will be reduced due to the logic change to require two channels of one trip system to cause actuation. The overall control logic for the remaining portions of the CREFS [Control Room Emergency Filtration System] is not changed by the design change.

The changes proposed to the actions are intended to clarify system logic wording. The actions assure that automatic trip capability is maintained and if not, then the CREFS is placed in the pressurization mode as in the current TS. This is consistent with the current TS.

Based upon the above, the proposed amendment will not increase the probability or consequences of any accident previously evaluated.

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

The elimination of the electrical connection between the redundant trip systems in a given CREFS subsystem will restore trip system independence and eliminate the potential of a single failure disabling the radiation monitoring instrumentation trip function. Specifically, a single failure, resulting from a blown fuse caused by a fault in the affected existing circuit, could remove the control power to the isolation logic relays in both trip systems. These relays require power in order to actuate and perform their safety function. A loss of control power to both trip systems due to the fault could result in exposing main control room personnel to radiation doses that exceed GDC 19 limits.

In addition, the changes to Action Statement 70 of the specification assure that trip capability is maintained.

Based upon the above, the proposed change will not create the possibility of a new or different kind of accident or transient previous evaluated.

3) Involve a significant reduction in the margin of safety because:

The proposed TS change will not prevent the isolation logic relays from performing their function or cause false trips. The alarm/trip setpoints for the affected monitors (including their measurement ranges) remain unchanged. The changes proposed to the actions are intended to clarify system logic wording. The actions assure that automatic trip capability is maintained and if not, then the CREFS is placed in the pressurization mode as in the current TS. This is consistent with the current TS.

Based on the above, the proposed TS change does not involve a significant reduction in the margin of safety.

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

Local Public Document location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348

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

NRC Project Director: Robert A. Capra

Entergy Gulf States, Inc., Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: August 5, 1997

Description of amendment request: The proposed amendment would revise the Technical Specifications for the Safety Limit Minimum Critical Power Ratio (SLMCPR) for Cycle 8 operation.

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

The plant/cycle specific SLMCPRs have been calculated using methods identical to those used by GE (General Electric) to assess the SLMCPR for other BWRs (boiling water reactors). Similar methods were used to determine the value of the SLMCPR for the previous cycle. These methods are within the existing design and licensing basis and cannot increase the probability or severity of an accident. The basis of the SLMCPR calculation is to ensure that greater that 99.9% of all fuel rods in the core avoid transition boiling and fuel damage in the event of the occurrence of Anticipated Operational Occurrences (AOO) or a postulated accident.

The SLMCPR is used to establish the Operating Limit Minimum Critical Power Ratio (OLMCPR). Neither the SLMCPR nor the OLMCPR are initiators or affect initiators of an accident previously evaluated and therefore changes to the SLMCPR do not increase the probability of any accident previously evaluated. The proposed changes involve the use of an accepted methodology in calculating the SLMCPR and, since there is no change in the definition of the SLMCPR, these changes will not affect the consequences of any accident previously evaluated. In addition, the proposed changes do not involve any change in the way the plant is operated. Existing procedures will ensure that the SLMCPR is not violated. Therefore, these changes have no effect on the consequences of an accident.

On these bases, there will be no increase in the probability or consequences of an accident previously analyzed as a result the proposed changes.

The proposed changes consist of SLMCPR calculated from an accepted method of analysis which has been used by many BWRs. These changes do not involve any alteration of the plant and do not affect the plant operation. Neither the SLMCPR nor the OLMCPR can initiate an event, therefore a change to the SLMCPR does not create the possibility of occurrence of a new or different kind of accident from any accident previously evaluated.

The SLMCPR is a Technical Specification numerical value to ensure that 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated. The proposed SLMCPR change results from SLMCPR analysis using the accepted methods as identified in the Attachment.

The margin of safety resides between the SLMCPR and the point at which fuel fails. Maintaining the MCPR above the proposed SLMCPR will maintain the margin of safety associated with GE's SLMCPR methodology. Existing plant procedures will continue to ensure that the SLMCPR is not violated.

Therefore, this request 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 location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005

NRC Project Director: James W. Clifford, Acting

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

Date of amendment request: July 28, 1997

Description of amendment request: This amendment is to modify the actions associated with Technical Specifications Table 3.3-1 for the Reactor Protective Instrumentation and Table 3.3-3 for the Engineered Safety Feature Actuation System 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:

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

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

The proposed change to the ANO-2 Technical Specifications (TS) modifies the allowed outage time that a channel of the Refueling Water Tank (RWT) Level - Low or Steam Generator differential pressure (delta P) can be in the tripped condition from a maximum of approximately 18 months when one channel is inoperable, and 31 days when two channels are inoperable, to 48 hours for either of these conditions.

If a channel of RWT Level Low is in the tripped condition and a single failure occurs that results in one of the other three channels of RWT Level - Low to actuate, a Recirculation Actuation System (RAS) signal would be generated. This scenario would not be considered severe if the condition occurred as a single event. However, during the injection phase of a Loss of Coolant Accident (LOCA) with a channel of RWT Level - Low in the trip condition with the above single failure, a premature RAS actuation would be the result. The premature RAS actuation would prevent the contents of the RWT from being injected into the reactor coolant system and possibly resulting in failure of both trains of Emergency Core Cooling System (ECCS) and the Containment Spray System.

With one channel of Steam Generator delta P in the tripped condition, as allowed by the TS, the plant is vulnerable to the single failure of a second Steam Generator delta P channel under an unisolable Main Steam Line Break condition. The following scenario will result in the faulted Steam Generator being supplied feedwater by the Emergency Feedwater System during an unisolable Main Steam Line Break. One channel of Steam Generator delta P is in the tripped condition as allowed by the TS and a Main Steam Line Break occurs that is unisolable. During this event one of the remaining channels of Steam Generator delta P fails resulting in incorrectly feeding the faulted Steam Generator. Reducing the time that a channel of RWT Level - Low or Steam Generator delta P can be placed in the tripped condition will reduce the probability of these scenarios from occurring.

The consequences of feeding the faulted Steam Generator during a main steam line break event or a premature RAS actuation during a LOCA are both significant. The proposed change reduces the allowed time a channel of RWT Level - Low or Steam Generator delta P can be in the tripped condition. Reducing the time the channel can be in the tripped condition and thus, the exposure time to this scenario, would not be an accident initiator or involve an increase in the consequences of any accident previously evaluated.

The remaining proposed changes are consistent with NUREG-1432, ''Standard Technical Specifications for Combustion Engineering Plants'' and are intended to correct the actions required by TS Tables 3.3-1 and 3.3-3 to the current NRC approved guidance.

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

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

The proposed change does not modify the design or configuration of the plant. The proposed change provides a more conservative time limit for a channel to be in the tripped condition and provides the required actions when a channel is out of service. There has been no physical change to plant systems, structures or components nor will the proposed change reduce the ability of any of the safety related equipment required to mitigate anticipated operational occurrences or accidents. This change will potentially increase the ability of safety related equipment to perform their functions. The configuration allowed by the proposed specification is permitted by the existing specification.

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

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

The proposed change provides a more restrictive time limit for a channel of RWT Level Low or Steam Generator delta P to be in the tripped condition than is currently allowed by the TS. By reducing the allowed time, the probability is reduced that a single failure of another channel would result in a premature RAS actuation during the injection phase of a LOCA or the feeding of a faulted Steam Generator. By limiting the vulnerability to these events and their consequences, the proposed change will increase the margin of safety.

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

Based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change 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 location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005-3502

NRC Project Director: James W. Clifford, Acting

Florida Power and Light Company, Docket No. 50-335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of amendment request: July 22, 1997

Description of amendment request: The proposed amendment will incorporate a recent evaluation of a postulated inadvertent opening of a Main Steam Safety Valve (MSSV) into the current licensing basis for St. Lucie Unit 1. An assessment of the potential consequences of this specific transient is not presently contained in the Updated Final Safety Analysis Report (UFSAR), and the proposed license amendment is required by 10 CFR 50.59(c).

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

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Unit 1 UFSAR includes analyses for excess load events; however, a stuck open MSSV is not specifically evaluated in the UFSAR. This proposed amendment will add an evaluation of an inadvertent opening of an MSSV to the licensing basis of the plant. The probability of occurrence of an excess load event is not increased by this amendment since the frequency of initiating events has not changed and there is no change to the plant or plant operation as a result of this amendment. Thus, there is no significant increase in the probability of any accident previously analyzed.

The radiological consequences of an excess load event other than steam line ruptures are discussed in UFSAR Section 15.2.11.2.3, and are based on the inadvertent opening of an Atmospheric Steam Dump Valve (ADV). This proposed amendment revises the radiological consequences of the UFSAR excess load event to incorporate the results of a recent evaluation of an inadvertent opening of an MSSV. The consequences of the postulated MSSV scenario are greater than those of an inadvertent opening of an ADV, but the predicted two hour site boundary doses remain a small fraction of 10 CFR 100 limits. In addition, the Unit 1 results are bounded by the St. Lucie Unit 2 analysis results which are reported in Section 15.1.3.1.1.3 of the Unit 2 UFSAR. Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment will add an evaluation of an inadvertent opening of an MSSV to the licensing basis of the plant. The evaluation addresses an anticipated operational occurrence (AOO) and is classified as an Excess Load event under the PSL1 [Plant St. Lucie Unit 1] accident classification criteria. Although an analysis of this specific transient is not currently provided in the UFSAR, analyses of Excess Load events other than steam line ruptures are reported in UFSAR Section 15.2.11. The amendment does not change plant design or operation and does not introduce new failure modes or system interactions. Thus, operation of the facility with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed license amendment adds an engineering evaluation to the licensing basis of the plant to address the consequences of a postulated stuck open MSSV. A change is not being made to plant design or operation. A change is not being made to any Technical Specification Limiting Condition for Operation, Action, or Surveillance Requirement. The evaluation demonstrates that, post-trip, the reactor would remain subcritical throughout the transient, and that the radiological consequences of a stuck open MSSV are a small fraction of 10 CFR 100 limits. Therefore, operation of the facility in accordance with the 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 location: Indian River Community College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34981-5596

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, et al., Docket No. 50-389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of amendment request: August 1, 1997

Description of amendment request: The proposed amendment will extend the semi-annual surveillance interval specified in Table 4.3-2 of the Technical Specifications for testing the Engineered Safety Features Actuation System (ESFAS) subgroup relays to an interval consistent with Combustion Engineering Owners Group Report CEN-403, Revision 1-A, March 1996. The proposed surveillance interval is at least once per 18 months, with testing to be performed on a staggered test basis.

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

(1) Operation of the facility, in accordance with the proposed amendment, would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment revises the testing frequency of ESFAS subgroup relays, and is based on demonstrated relay reliability. These relays actuate the engineered safety features (ESF) equipment which is installed to mitigate design basis accidents. ESF system components are not considered initiators of any design basis accident. Therefore, operation of the facility with the proposed amendment would not involve a significant increase in the probability of an accident previously evaluated.

The proposed amendment does not alter the design or operation of ESF systems. The mean time between failures demonstrated by the ESFAS subgroup relays is significantly greater than the proposed surveillance interval, and testing will be performed on a staggered test basis. This, in addition to ESF redundancy, provides assurance that these systems will continue to function as evaluated to mitigate design basis accidents. Therefore, operation of the facility, in accordance with the proposed amendment, would not involve a significant increase in the consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment will not change the physical plant or the modes of operation defined in the facility license. The changes do not involve the addition of new equipment or the modification of existing equipment, nor do they alter the design of St. Lucie plant systems. Therefore, operation of the facility, in accordance with the proposed amendment, would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed amendment revises the surveillance interval for testing the ESFAS subgroup relays consistent with the Combustion Engineering Owners Group topical report CEN-403, Revision 1-A, and conforms to criteria specified in the associated safety evaluation issued by the NRC staff. The St. Lucie Unit 2 subgroup relay mean time between failures is significantly greater than the proposed surveillance interval, and testing will be performed on a staggered test basis. ESFAS setpoints, system operation, and plant configuration will not be changed, and the subgroup relays are not subject to time-related instrument drift. Accident analyses assumptions, initial conditions, and conclusions reported in the Updated Final Safety Analysis Report are not changed by the revised surveillance interval. Therefore, operation of the facility in accordance with the 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 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

GPU Nuclear (GPUN) Corporation, et al., Docket No. 50-289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of amendment request: July 30, 1997

Description of amendment request: The purpose of this Technical Specification change request (TSCR) is to incorporate additional system leakage limits and leak test requirements for systems outside containment which were not previously contained in Technical Specification 4.5.4 nor considered in the TMI-1 Updated Final Safety Analysis Report (UFSAR) design basis accident (DBA) analysis dose calculations for 2568 MWt. This TSCR also revises the Technical Specification 3.15.3 Bases for the Auxiliary and Fuel Handling Building Ventilation System (AFHBVS). The revisions to Technical Specification 3.15.3 Bases for the AFHBVS serve to clarify system design requirements and accident analysis considerations. The revision states that the AFHBVS is not credited in reducing off-site dose for the Maximum Hypothetical Accident (MHA) or the Waste Gas Tank Rupture (WGTR) accident analysis dose calculations.

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 (SHC), which is presented below:

GPUN has determined that this TSCR poses no significant hazards consideration as defined by 10 CFR 50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. No physical modifications which would change structures, systems, or components are being made or proposed by this TSCR. This change has no [effect] on the LOCA [loss-of-coolant accident] safety analysis for ECCS [emergency core cooling system] performance. The results of revised MHA dose calculation are less than that previously evaluated in the UFSAR for the exclusion area boundary (EAB). In addition the doses are below the 10 CFR 100 guideline limits for both the EAB and low population zone (LPZ) ..., and below the 10 CFR 50 Appendix A, GDC [General Design Criteria]-19 limits for the control room. The LPZ increases in dose consequence are the result of using more conservative assumptions in the revised analyses and the new values remain a small fraction of the 10 CFR 100 limits. The WGTR dose calculation is not affected by this TSCR. The proposed Technical Specification changes ensure that the MHA and WGTR accident analysis parameters remain bounded during plant operation.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated. This TSCR does not involve any physical modifications which would affect structures, systems, or components, nor does it involve any changes in plant operation. The only changes resulting from this TSCR are revisions to leakage limits and testing requirements necessary to reflect the revised MHA analysis and to correct discrepancies identified by the NRC .... Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. This TSCR does not involve changes to Technical Specification defined Safety Limits, Limiting Conditions for Operation, and does not involve any change to safety system setpoints for operation. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

Local Public Document location: Law/Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105

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

NRC Project Director: Ronald B. Eaton (Acting)

GPU Nuclear Corporation, et al., Docket No. 50-289, Three Mile Island Nuclear Station, Unit No. 1, (TMI-1) Dauphin County, Pennsylvania

Date of amendments request: August 12, 1997

Description of amendments request: The amendment requests changes to the Surveillance Specification of the Technical Specification (TS) for the once through steam generator (OTSG) inservice inspection for TMI-1 Cycle 12 Refueling (12R) examinations applicable to TMI-1 Cycle 12 operation. These proposed changes impose axial and circumferential extent sizing limitations in addition to TS requirements for inside diameter (ID) initiated degradation where bobbin coil eddy current test (ECT) signal amplitudes do not permit reliable through wall sizing. Editorial changes are being made to improve consistency of format, to the Bases which relate to the requested changes in Section 4.19 of the TS, and to the reporting requirements in Section 4.19.5 of 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:

GPU Nuclear has determined that this TSCR [Technical Specification Change Request] poses no significant hazards consideration as defined by 10 CFR 50.92.

A. These proposed changes do not represent a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The only accidents previously evaluated that could be significantly affected by changes to the OTSG tube inservice inspection requirements are the steam generator tube rupture (STGR) and the main steam line break (MSLB) accidents.

The proposed flaw disposition strategy based on measurable eddy current parameters of axial and circumferential extent for Inside Diameter (ID) Initiated Inter-Granular Attack (IGA) will provide high confidence that unacceptable flaws that do not have the required structural integrity to withstand the MSLB are removed from service. The proposed axial and circumferential length limits for eddy current inside diameter degradation indications meet the RG [Regulatory Guide] 1.121 acceptance criteria for margin to failure for MSLB applied differential pressure and axial tube loads. The capability for detection of flaws is unaffected and the identification of tubes which should be repaired or removed from service is maintained or improved. The operation of the OTSG or related structures, systems, or components is otherwise unaffected. Therefore, neither the probability nor consequences of a SGTR is significantly increased either during normal operation or due to the limiting loads of [an] MSLB accident.

Neither the editorial changes in format, punctuation, or grammar nor the administrative changes or changes in reporting requirements, as described above, could significantly affect the probability of occurrence or consequences of any accident previously evaluated.

B. These proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because there are no hardware changes involved nor changes to any operating practices. These changes involve only the OTSG tube inservice inspection surveillance requirements, which could only affect the potential for OTSG primary-to-secondary leakage. The proposed changes impose additional flaw length limits for ID IGA that go beyond existing requirements to assure tube structural and leakage integrity.

In addition, neither the editorial changes in format, punctuation, or grammar nor the administrative changes, as described above, could possibly create the possibility of an accident of a new or different type from any previously evaluated. These changes are included only to improve the clarity and readability of the Technical Specifications and comply with the NRC's desire to obtain the results of the inspections as soon as practical.

Therefore, these changes do not create the potential for single or multiple tube ruptures or any other kind of accident different from those that have been evaluated.

C. Those proposed changes do not involve a significant reduction in a margin of safety because the changes are more restrictive than the current technical specification and the margins of safety defined in R.G. 1.121 are retained. The probability of detecting degradation is unchanged since the bobbin coil eddy current methods will continue to be the primary means of initial detection and the probability of leakage from any indications left in service remains acceptable small. The strategy for dispositioning ID initiated IGA will continue to provide a high level of confidence that tubes exceeding the allowable limits for tube integrity are repaired or removed from service.

In addition, neither the editorial changes in format, punctuation, or grammar nor the administrative changes or changes in reporting requirements, as described above, could significantly affect a margin of safety and are included only to improve the clarity and readability of the Technical Specifications and comply with the NRC's desire to obtain the results from tube inspections as soon as practical.

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 amendments request involves no significant hazards consideration.

Local Public Document location: Law/Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105

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

NRC Project Director: Ronald B. Eaton, Acting

GPU Nuclear Corporation, et al., Docket No. 50-289, Three Mile Island Nuclear Station, Unit No. 1, (TMI-1) Dauphin County, Pennsylvania

Date of amendment request: August 14, 1997

Description of amendment request: The proposed license amendment, if approved, would revise the TMI-1 Updated Final Safety Analysis Report (UFSAR) Section 14.1.2.9-Steam Line Break analysis to include the environmental dose consequences associated with postulated accident-induced steam generator tube leakage not previously analyzed. The revised environmental dose consequences for the TMI-1 Steam Line Break analysis would be increased above the values previously reviewed by the NRC.

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

GPU Nuclear has determined that this License Amendment Request poses no significant hazards as defined by 10 CFR 50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. This change has no effect on structures, systems or components prior to the postulated steam line break accident or any other accident. OTSG [once through steam generator] tube loads resulting from other postulated accidents are bounded by the calculated steam line break accident tube loads. Other TMI-1 design basis accidents, which could result in OTSG tube loads and environmental dose consequences, involve releases within the reactor building. These events generally result in rapid depressurization of the primary system which minimizes the differential pressure needed to establish a significant primary-to-secondary leak rate and the OTSG is isolated. Accordingly, leakage to the environment as a result of induced tube loads from postulated accidents other than steam line break is insignificant and therefore need not be considered. The existing steam line break criteria is maintained in that OTSG structural integrity is assured and postulated doses remain within 10 CFR 100 limits. The new radiological consequences of the revised steam line break dose calculation are below 10 CFR 100 limits for the exclusion area boundary (EAB) and low population zone (LPZ). The 10 CFR 50, Appendix A, GDC [General Design Criterion]-19 limits for the control room are not affected by this change since the source term assumed for the TMI-1 control room habitability analysis remains bounding.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated. This change has no impact on any plant structures, systems or components. OTSG tube structural integrity is maintained. The only impact is the revised radiological consequences of the steam line break analysis to account for hypothetical accident induced primary-to-secondary leakage.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. This change to the steam line break

dose consequences does not involve a significant reduction in a margin of safety. The new radiological consequences of the revised steam line break dose calculation are below 10 CFR 100 limits for the EAB and LPZ, and do not affect the TMI-1 control room habitability analysis results. This change has no impact on any structures, systems or components.

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 location: Law/ Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105

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

NRC Project Director: Ronald B. Eaton, Acting

Niagara Mohawk Power Corporation, Docket No. 50-410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of amendment request: July 31, 1997

Description of amendment request: The proposed amendment would change Action Statement 36 of Technical Specification (TS) Table 3.3.3-1, ''Emergency Core Cooling System Actuation Instrumentation,'' so as to specify actions to be taken if one or more channels per trip function should be inoperable in the high-pressure core spray (HPCS) drywell pressure and reactor water level instrumentation. Presently, Action 36 only addresses actions for the plant condition of having one channel per trip function inoperable. Specifically, Action 36 would be changed to require that, with the number of operable channels less than required by the minimum operable channels per trip function requirement, then (1) with one channel inoperable, the inoperable channel is to be placed in the tripped condition within 24 hours or the HPCS system is to be declared inoperable, and (2) with more than one channel inoperable, the HPCS system is to be declared inoperable.

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

1. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The changes to Table 3.3.3-1, Action 36, will allow Action 36 to be in effect for the plant condition where more than one channel is inoperable per trip function in the HPCS drywell pressure and reactor water level instrumentation and will clarify the actions required if more than one channel is inoperable. Specifically, this action statement will allow the HPCS to be declared inoperable rather than to initiate plant shutdown per TS 3.0.3. None of the precursors of previously evaluated accidents are affected and therefore, the probability of an accident previously evaluated is not increased.

The HPCS system will continue to perform its safety function to automatically initiate and inject water into the vessel. The out of service time for the initiating instruments remains bounded by the out of service time for HPCS. Therefore, these changes will not involve a significant increase in the consequences of an accident previously evaluated.

2. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any previously evaluated.

The changes to Table 3.3.3-1, Action 36, will allow Action 36 to be in effect for plant conditions where more than one channel is inoperable per trip function in the HPCS drywell pressure and reactor water level instrumentation and will clarify the actions required if more than one channel is inoperable. No physical modification of the plant is involved and no changes to the methods in which plant systems are operated are required. The changes do not introduce any new failure modes or conditions that may create a new or different accident. Therefore, the changes do not by themselves create the possibility of a new or different kind of accident [from any accident] previously evaluated.

3. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The change to Table 3.3.3-1, Action 36, will allow Action 36 to be in effect for plant conditions where more than one channel is inoperable per trip function in the HPCS drywell pressure and reactor water level instrumentation and will clarify the actions required if more than one channel is inoperable. The changes do not adversely affect any physical barrier to the release of radiation to plant personnel or to the public. The proposed change provides consistency between the ECCS [emergency core cooling system] instrumentation and system TS. The TS also continues to require the operability of other injection systems coincidental with HPCS inoperability. The change has the benefit of avoiding unnecessary challenges to plant systems during an unnecessary plant shutdown. Therefore, the changes do not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005-3502

NRC Project Director: Alexander W. Dromerick, Acting Director

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

Date of amendment request: April 14, 1997

Description of amendment request: The proposed amendment would allow the Safety Review Committee (SRC) to perform a review, rather than an audit, of plant staff performance. The proposed amendment also involves a title change.

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

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

Response:

This amendment application does not involve a significant increase in the probability or consequences of an accident previously analyzed. The proposed changes allow the SRC to perform a review, rather than an audit, of plant staff performance. This change does not diminish the SRC—s effectiveness. A review of the 1995 QA [quality assurance] audit of plant staff performance shows that no findings were issued. This indicates that the other review mechanisms currently in place are sufficient to ensure that plant staff performance is monitored.

The position title change is an administrative change as all previously performed functions are being maintained and the responsibilities and reporting chain for this position remain the same. Therefore, the proposed changes do not affect the probability or consequences of any previously analyzed accident.

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

Response:

This amendment application does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes affect an SRC audit requirement and a position title. These changes do not affect plant equipment or the way the plant operates. Therefore, they cannot create a new or different kind of accident.

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

Response:

This amendment application does not involve a significant reduction in a margin of safety. The requested Technical Specification revisions require the SRC to review rather than audit facility staff performance and will not diminish the effectiveness of the SRC. A review of the 1995 audit confirms that performance of the annual audit is redundant as no findings or recommendations concerning plant staff performance were made. The QA/ORG [Operations Review Group] quarterly trend reports and SRC review of plant staff performance are adequate to ensure that plant staff performance is properly monitored.

The position title change is an administrative change as all previously performed functions are being maintained and the responsibilities and reporting chain for this position remain the same. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mr. Charles M. Pratt, 10 Columbus Circle, New York, New York 10019

NRC Project Director: Alexander W. Dromerick, Acting

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

Date of amendment request: May 29, 1997

Description of amendment request: The amendment would revise the definition of Containment Integrity in Section 1.10, and revise Section 3.6 and Table 3.6-1 for consistency. Several valves would be added to Table 3.6-1 to be consistent with the revised definition in Section 1.10. The amendment would also add a footnote stating that valves SP-SOV-506 and SP-SOV-507 in Table 4.4-1, ''Containment Isolation Valves'' are sealed from weld channel and containment penetration pressurization system (WCCPPS).

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

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

The revision of the definition of containment integrity in Section 1.10, Section 3.6.A.1, the Basis, and the addition of existing containment isolation valves into the Table of Containment Isolation Valves in the Technical Specifications does not change the design, operation or testing of the plant. Section 1.10 is being revised to clearly cover all non-automatic containment isolation valves, and the valves are being added to be consistent with the revised definition. The valves being added are currently identified as containment isolation valves and tested as specified in the Final Safety Analysis Report. Additionally, valves CB-3, 4, 7 & 8 are controlled in accordance with Section 1.10.5 (revised numbering) for the airlock doors. Because the design and operation are not being changed, the addition of the valves has no effect on the probability or consequences of an accident.

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

Changing the definition in Section 1.10 and the list of containment isolation valves for consistency does not change the design, operation or testing of the plant. Section 1.10 is being revised to clearly cover all non-automatic containment isolation valves, and the valves are being added to be consistent with the revised definition. The valves being added are currently identified as containment isolation valves and tested as specified in the Final Safety Analysis Report. Therefore, without changing design, operation or testing of the plant this does not create a new or different type of accident.

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

The proposed changes in the definition for containment integrity and the listings of Containment Isolation Valves in the Technical Specifications does not involve a significant reduction in the margin of safety because the change reflects current design, operation and testing of the plant, and will not alter plant operation.

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 location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601

Attorney for licensee: Mr. Charles M. Pratt, 10 Columbus Circle, New York, New York 10019

NRC Project Director: Alexander W. Dromerick, Acting

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

Date of amendment request: June 25, 1997

Description of amendment request: The proposed amendment would allow for up to +17/-12 steps of control rod misalignment for core power greater than 85% rated thermal power.

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

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

Response:

No. Based on the Westinghouse evaluation in WCAP-14668, the Authority has determined that all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the TS [technical specification] Bases is not reduced in any of the IP3 licensing basis accident analysis (even for misalignments to [plus or minus] 24 steps for core power [less than or equal to] 85% of RTP). Increasing the magnitude of allowed control rod indicated misalignment is not a contributor to the mechanistic cause of an accident evaluated in the FSAR [final safety analysis report]. Neither the rod control system nor the rod position indicator function is being altered. Therefore, the probability of an accident previously evaluated has not significantly increased. Because design limitations continue to be met, and the integrity of the reactor coolant system pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid. Therefore, the consequences of an accident previously evaluated will not be significantly increased.

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

Response:

No. Based on the Westinghouse evaluation in WCAP-14668, the Authority has determined that all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the TS is not reduced in any of the IP3 licensing basis accident analysis. Increasing the magnitude of allowed control rod indicated misalignment is not a contributor to the mechanistic cause of any accident. Neither the rod control system nor the rod position indicator function is being altered. Therefore, an accident which is new or different than any previously evaluated will not be created.

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

Response:

No. Based on the Westinghouse evaluation in WCAP-14668, the Authority has determined that all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the TS Bases is not reduced in any of the IP3 [Indian Point Unit 3] licensing basis accident analysis based on the changes to safety analyses input parameter values as discussed in WCAP-14668. Since the evaluations in Section 3.0 of WCAP-14668 demonstrate that all applicable acceptance criteria continue to be met, the proposed change will not involve a significant reduction in 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 location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601

Attorney for licensee: Mr. Charles M. Pratt, 10 Columbus Circle, New York, New York 10019

NRC Project Director: Alexander W. Dromerick, Acting

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

Date of amendment request: June 19, 1997, as supplemented by letters dated July 30 and 31, 1997

Description of amendment request: The proposed amendment would provide changes to Technical Specification (TS) 4.1.3.1.2, ''Control Rod Operability,'' TS 3.1.3.6, ''Control Rod Drive Coupling,'' TS 3.1.3.7, ''Control Rod Position Indication'', TS 3.1.4.1, ''Rod Worth Minimizer,'' TS 3/4.1.4.2, ''Rod Sequence Control System,'' TS 3/4.10.2, ''Special Test Exceptions - Rod Sequence Control System,'' the Bases for TS 2.2.1.2, ''Average Power Range Monitor,'' the Bases for TS 3/4.1.4, ''Control Rod Program Controls,'' and the Bases for TS 3/4.10.2, ''Rod Sequence Control System.'' The changes are proposed in order to eliminate the Rod Sequence Control System (RSCS) Limiting Condition for Operation and Surveillance Requirements from the TSs and reduce the Rod Worth Minimizer (RWM) low power setpoint from 20% to 10%. Changes are also proposed as necessary to delete reference to the RSCS from the TSs and to incorporate additional requirements necessary to support the elimination of the RSCS.

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

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

A. RSCS Deletion

The RSCS system restricts the pattern of control rods prior to a postulated control rod drop accident (RDA) so as to minimize the reactivity worth of the dropped rod. The RSCS provides no mitigation following the postulated RDA. The ability to restrict the pattern of control rods also allows the RSCS to be able to reduce the probability of a Continuous Rod Withdrawal During Reactor Startup, as described in the Hope Creek UFSAR [Updated Final Safety Analysis Report] Section 15.4.1.2 and Appendix 15B. However, to determine the consequence of such a rod withdrawal event, the RSCS is not credited, and the rod is assumed to be fully withdrawn from the core at its maximum rate. The RDA is therefore the only analyzed accident impacted by the proposed deletion of the RSCS system. Since the RSCS system plays no role in preventing a[n] RDA, it therefore does not affect the probability of occurrence of this postulated accident.

As stated in an NRC Safety Evaluation Report dated December 27, 1987, the RSCS system is the result of requirements promulgated by the NRC staff in the early 1970's in response to unknowns and perceived problems relating to the RDA. The GE [General Electric] calculational methodology being used at that time produced results showing that, even without pattern errors, calculated enthalpies for the RDA approached limiting values. In addition, the Rod Worth Minimizer (RWM) Technical Specifications were not effective in ensuring RWM availability and use, and the system was poorly maintained and frequently bypassed thus providing no significant protection. Second operator substitution for the RWM was used routinely and was providing minimal protection. Finally, no reliable study existed to address the probability of exceeding enthalpy limits as a result of an RDA.

Information associated with the above concerns has been significantly expanded or modified. Studies using improved methodologies have proven significantly lower peak fuel enthalpy values compared with methodologies in use when the RSCS was originally developed. In addition, a reliable probability study has been completed showing that the probability of an RDA exceeding NRC limits is very low. As a result, NRC review of the RSCS requirements has concluded that the RSCS system is not needed and operation without it is acceptable provided: 1) TSs are modified to minimize the use of the second operator option, 2) procedures and quality control associated with the second operator option are reviewed to ensure that this option provides an effective and truly independent monitoring process; and 3) rod patterns used are at least equivalent to Banked Pattern Withdrawal System (BPWS) patterns. Each of these items has been addressed for the Hope Creek Generating Station.

As a result of the resolution of the original concerns associated with the RDA, the RWM system and limited use of the second operator option, when properly instituted, are now deemed to provide adequate protection to maintain the consequences of the RDA at an acceptable level. The remaining concerns regarding operation without the RSCS system and proper use of the second operator substitution option have been addressed for the Hope Creek Generating Station. We therefore conclude that the redundant RSCS system is no longer necessary and its deletion from the Technical Specifications will not significantly increase the probability or consequences of an RDA.

B. RWM Setpoint Reduction

The RWM system restricts the pattern of control rods prior to a postulated control rod drop accident (RDA) so as to minimize the reactivity worth of the dropped rod. The RWM provides no mitigation following the postulated RDA. The ability to restrict the pattern of control rods also allows the RWM to be able to reduce the probability of a Continuous Rod Withdrawal During Reactor Startup, as described in the Hope Creek UFSAR Section 15.4.1.2 and Appendix 15B. However, to determine the consequence of such a rod withdrawal event, the RWM is not credited, and the rod is assumed to be fully withdrawn from the core at its maximum rate. The RDA is therefore the only analyzed accident impacted by the proposed reduction in the RWM setpoint. Since the RWM system plays no role in preventing a[n] RDA, it therefore does not affect the probability of occurrence of this postulated accident.

Existing calculations have demonstrated that no significant RDA can occur above 10% power. Calculations by both General Electric and the Brookhaven National Laboratory indicate that, even with significant error patterns, peak fuel enthalpy is reduced well below required limits at 10% power. The 20% limit was originally required as an extreme bound because of the then existing uncertainties in the analyses. Based on the current analyses, the 10% level is now acceptable and deemed to provide adequate protection to maintain the consequences of an RDA at an acceptable level. Changing the RWM setpoint from 20% to 10% will therefore not significantly increase the consequences of any previously analyzed accident.

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

A. RSCS Deletion

Operation of the RSCS cannot cause or prevent an accident; this system functions to minimize the consequences of an RDA. The Bank Position Withdrawal Sequence (BPWS) will still be used to ensure that rod pull pattern[s] are constrained to those assumed in the RDA. The RSCS has no impact on the operation of any other system, and therefore its deletion will not contribute to a malfunction in any other equipment nor create the possibility of a new or different accident from any accident previously evaluated.

B. RWM Setpoint Reduction

Operation of the RWM cannot cause or prevent an accident; this system functions to minimize the consequences of an RDA. The RWM has no impact on the operation of any other system, and therefore changing its setpoint from 20% to 10% will not contribute to a malfunction in any other equipment nor create the possibility of a new or different accident from any accident previously evaluated.

3. Do not involve a significant reduction in a margin of safety.

A. RSCS Deletion

When the original decisions were made regarding the need for the RSCS system, numerous perceived problems in the RDA analysis existed. As noted in the discussion of the consequences of previously analyzed accidents in Item 1 above: 1) the perceived RDA problems have been resolved; 2) reviews of the RDA have concluded that the RSCS is not needed to mitigate the consequences of an RDA; and 3) operation without the RSCS is acceptable. The RWM and limited use of second operator substitution, when properly instituted, are now deemed adequate to ensure that peak fuel enthalpies remain below NRC limits. Therefore, the deletion of the redundant RSCS system will not significantly decrease any margin of safety.

B. RWM Setpoint Reduction

The Bases for the HCGS TSs state that when thermal power is greater than 20%, there is no possible rod worth that, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 calories per gram. Existing calculations demonstrate that the RDA is not a significant concern above 10% power, and therefore, a mitigation system is not needed for higher power level operation. Calculations by both General Electric and the Brookhaven National Laboratory indicate that, even with significant error patterns, peak fuel enthalpy is reduced well below required limits (280 calories per gram) at 10% power. The 20% limit was originally required as an extreme bound because of the then existing uncertainties in the analyses. Based on the current analyses, the 10% level is now acceptable and deemed to provide adequate assurance that the peak fuel enthalpy will remain below the NRC limits during a postulated RDA. Changing the RWM setpoint from 20% to 10% will therefore not significantly reduce any margin of safety.

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

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

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

NRC Project Director: John F. Stolz

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 30, 1997

Description of amendments request: The proposed amendments would change the Farley Technical Specifications to: revise and clarify the requirements for the Control Room Emergency Filtration System (CREFS), the Penetration Room Filtration System (PRFS) and the related Storage Pool Ventilation System (SPVS); revise the required number of radiation monitoring instrumentation channels; and delete the Containment Purge Exhaust Filter (CPEF) specification.

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:

Pursuant to 10 CFR 50.92, SNC [Southern Nuclear Operating Company, Inc.] has evaluated the proposed amendments and has determined that operation of the facility in accordance with the proposed amendments would not involve a significant hazards consideration. The basis for this determination is as follows:

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

The proposed changes to convert from ANSI N510-1980 to ASME N510-1989 for specific FNP [Joseph M. Farley Nuclear Plant] filtration surveillance testing requirements and related changes do not affect the probability of any accident occurring. The consequences of any accident will not be affected since the proposed changes will continue to ensure that appropriate and required surveillance testing for FNP filtration systems will be performed consistent with the revised accident analyses. The results of the fuel handling accident remain well within the guidelines of I0 CFR Part 100 and the doses due to a LOCA [loss-of-coolant accident], including ECCS [emergency core cooling system] recirculation loop leakage, remain within the guidelines of I0 CFR Part 100 and General Design Criterion 19 of Appendix A to I0 CFR Part 50. Relocating specific testing requirements to the FNP FSAR [Final Safety Analysis Report] has no effect on the probability or consequences of any accident previously evaluated since required testing will continue to be performed.

Therefore, the proposed TS [Technical Specification] changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Testing differences between ANSI N510-1980 and ASME N510-1989 have been evaluated by SNC and none of the proposed changes have the potential to create an accident at FNP. ASME N510-1989 has been endorsed and approved by the NRC for licensee use in NUREG 1431 [Standard Technical Specifications Westinghouse Plants]. Testing the additional channels of radiation monitoring and verification of penetration room boundary integrity do not require the affected systems to be placed in configurations different from design. Thus, no new system design or testing configuration is required for the changes being proposed that could create the possibility of any new or different kind of accident from any accident previously evaluated. Relocating specific testing requirements to the FSAR has no effect on the possibility of creating a new or different kind of accident from any accident previously evaluated since it is an administrative change in nature.

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

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

Conversion from the testing requirements of ANSI N510-1980 sections 10, 12, and 13 to ASME N510-1989 sections 10, 11, and 15 has been previously approved by the NRC at other nuclear facilities. ASME N510-1989 has been approved and endorsed by the NRC in NUREG 1431. The safety factor associated with the conservative charcoal adsorber laboratory test methods and dose calculations ensures that doses will continue to meet the guidelines of 10 CFR Part 100 and GDC [General Design Criterion] 19 of Appendix A to 10 CFR Part 50. The enhanced testing of radiation monitoring instrumentation and the penetration room boundary integrity provide additional assurance that the acceptance criteria of the safety analyses and the resultant margins of safety are not reduced. Relocating specific testing requirements to the FSAR has no effect on the margin of plant safety since required testing will continue to be performed. Clarifying the 10 hour run with heaters on is consistent with the Improved TS language and accomplishes the purpose for the surveillance. Therefore, SNC concludes based on the above, that the proposed changes do not result in a significant reduction of margin with respect to plant safety as defined in the Final Safety Analysis Report or the bases of the FNP technical specifications.

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

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

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

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201

NRC Project Director: Herbert N. Berkow

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 30, 1997

Description of amendments request: The proposed amendments would change the Farley Technical Specifications to incorporate the requirements necessary to change the basis for prevention of criticality in the fuel storage pool. This change eliminates the need for Boraflex as a neutron absorbing material in the fuel pool criticality analysis for both Unit 1 and Unit 2.

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

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

There is no significant increase in the probability of a fuel assembly drop accident in the spent fuel pool when considering the presence of soluble boron in the spent fuel pool water for criticality control. The handling of the fuel assemblies in the spent fuel pool has always been performed in borated water.

The consequences of a fuel assembly drop accident in the spent fuel pool are not affected when considering the presence of soluble boron.

Although the probability of misloading an assembly in the spent fuel racks may increase due to new assembly placement constraints, there is no significant increase in the probability of an accidental misloading of spent fuel assemblies into the spent fuel pool racks that will cause a criticality accident when considering the presence of soluble boron in the pool water for criticality control. Sufficient soluble boron will be maintained in the spent fuel pool to maintain k_{eff} below 0.95 following a postulated single misload. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the Technical Specification spent fuel rack storage configuration limitations. The addition of the spent fuel pool storage configuration surveillance in proposed new Technical Specifications 3.7.14 for Unit 1 and 3.7.15 for Unit 2 will provide increased assurance that a spent fuel pool inventory verification will be completed in a timely manner (7 days) after the relocation or addition of fuel assemblies in the spent fuel storage pool.

There is no significant increase in the consequences of the accidental misloading of spent fuel assemblies into the spent fuel pool racks because criticality analyses demonstrate that the pool will remain subcritical following an accidental misloading if the pool contains an adequate boron concentration. The proposed new Technical Specifications limitations will ensure that an adequate spent fuel pool boron concentration will be maintained.

In the event of failure of a spent fuel pool cooling pump, or loss of cooling to a spent fuel pool heat exchanger, the second spent fuel pool cooling train provides 100 percent backup capability, thus ensuring continued cooling of the spent fuel pool. However, even if a loss of spent fuel pool cooling were to occur, there is sufficient soluble boron to prevent K_{eff} from exceeding 0.95.

There is no significant increase in the probability of the loss of normal cooling to the spent fuel pool water when considering the presence of soluble boron in the pool water for subcriticality control since a high concentration of soluble boron has always been maintained in the spent fuel pool water.

A loss of normal cooling to the spent fuel pool water causes an increase in the temperature of the water passing through the stored fuel assemblies. This causes a decrease in water density which would result in a decrease in reactivity when Boraflex neutron absorber panels are present in the racks.

However, since Boraflex is not considered to be present, and the spent fuel pool water has a high concentration of boron, a density decrease causes a positive reactivity addition. However, the additional negative reactivity provided by the proposed 2000 ppm boron concentration limit, above that provided by the concentration required to maintain K_{eff} less than or equal to 0.95 (400 ppm), will compensate for the increased reactivity which could result from a loss of spent fuel pool cooling event. Because adequate soluble boron will be maintained in the spent fuel pool water, there is no significant increase in the consequences of a loss of normal cooling to the spent fuel pool.

Therefore, based on the conclusions of the above analysis, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Spent fuel handling accidents are not new or different types of accidents, they have been analyzed in Section 15.4.5 of the Final Safety Analysis Report.

Criticality accidents in the spent fuel pool are not new or different types of accidents, they have been analyzed in the Final Safety Analysis Report and in Criticality Analysis reports associated with specific licensing amendments for fuel enrichments up to 5.0 weight percent U-235.

Proposed new Technical Specifications 3.7.13 for Unit 1 and 3.7.14 for Unit 2 on the spent fuel pool boron concentration do not represent new concepts. The boron concentration in the spent fuel pool has always been maintained near at the limit of the RWST [refueling water storage tank] boron concentration for refueling purposes. These new proposed Technical Specifications establish new boron concentration requirements for the spent fuel pool water consistent with the results of the revised criticality analysis [ ].

Since soluble boron has always been maintained in the spent fuel pool water, the implementation of this new requirement will have little effect on normal pool operations and maintenance. The implementation of the proposed new limitations on the spent fuel pool boron concentration will only result in increased sampling to verify boron concentration. This increased sampling will not create the possibility of a new or different kind of accident.

Because soluble boron has always been present in the spent fuel pool, a dilution of the spent fuel pool soluble boron has always been a possibility. However, it was shown in the spent fuel pool dilution evaluation [ ] that a dilution of the Farley spent fuel pool which could reduce the spent fuel storage rack K_{eff} to less than 0.95 is not a credible event. Therefore, the implementation of new limitations on the spent fuel pool boron concentration will not result in the possibility of a new kind of accident.

Proposed new Technical Specifications 3.7.14 for Unit 1 and 3.7.15 for Unit 2, and 5.6.1.1.e., 5.6.1.1.f, and 5.6.1.1.g. (for Unit 1) specify the requirements for the spent fuel rack storage configurations, and do not represent new concepts. These proposed new spent fuel pool storage configuration limitations are consistent with the assumptions made in the spent fuel rack criticality analysis, and will not have any significant effect on normal spent fuel pool operations and maintenance and will not create any possibility of a new or different kind of accident. Verifications will continue to be performed to ensure that the spent fuel pool loading configuration meets specified requirements.

As discussed above, the proposed changes will not create the possibility of a new or different kind of accident. There is no significant change in plant configuration, equipment design or equipment. The accident analysis in the Final Safety Analysis Report remains bounding.

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

The proposed Technical Specification changes and the resulting spent fuel storage operating limits will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality analysis [ ] performed in accordance the Westinghouse spent fuel rack criticality analysis methodology described in [WCAP-14416-NP-A, ''Westinghouse Spent Fuel Rack Criticality Analysis Methodology,'' Revision 1, November 1996].

The criticality analysis utilized credit for soluble boron to ensure K_{eff} will be less than or equal to 0.95 under normal circumstances, and storage configurations have been defined using a 95/95 K_{eff} calculation to ensure that the spent fuel rack K_{eff} will be less than 1.0 with no soluble boron.

Soluble boron credit is used to provide safety margin by maintaining K_{eff} less than or equal to 0.95, including uncertainties, tolerances, and accident conditions in the presence of spent fuel pool soluble boron.

The loss of substantial amounts of soluble boron from the spent fuel pool which could lead to exceeding a K_{eff} of 0.95 has been evaluated [ ] and shown to be not credible.

The evaluations which...show that the dilution of the spent fuel pool boron concentration from 2000 ppm to 400 ppm is not credible, combined with the 95/95 calculation, which shows that the spent fuel rack K_{eff} remain less than 1.0 when flooded with unborated water, provide a level of safety comparable to the conservative criticality analysis methodology required by [USNRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, June 1987, USNRC Spent Fuel Storage Facility Design Bases (for comment) Proposed Revision 2, 1981, Regulatory Guide 1.13, and ANS, Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations, ANSI/ANS-57.2-1983].

Therefore, the proposed changes in this license amendment will not result in a significant reduction in the plant's 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 location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: July 11, 1997

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) to implement 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program,'' with four exceptions as detailed in the licensee's application. Specifically, changes are requested for TSs 3.7/4.7, STATION CONTAINMENT SYSTEMS, their associated BASES, and changes to TS Table 4.7.2. Included in the above changes is a revision to the conservative wording of Surveillance Requirement (SR) 4.7.A.3 that is being replaced by wording from the Standard Technical Specifications, and the relocation of this SR to the Limiting Condition for Operation. The change to TS Table 4.7.2 updates the information in the Table to the current operational practices, as approved by an NRC letter dated May 3, 1982. In addition, a description of Vermont Yankee's Primary Containment Leakage Rate Testing Program (PCLRTP) will be added to the Administrative Controls Section (6.0) of the TSs. The testing intervals for the containment system and for the components that penetrate the primary containment, under Option B of Appendix J will be performance-based.

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:

Option B

1. Does the change involve a significant increase in the probability or consequences of an 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 contribute to initiation of any accidents previously evaluated. Thus, the proposed change cannot increase the probability of any accident previously evaluated.

The proposed change potentially affects the leak-tight integrity of the containment structure designed to mitigate the consequences of a loss-of-coolant accident (LOCA). The function of the containment is to maintain functional integrity during and following the peak transient pressures and temperatures which result from any LOCA. The containment is designed to limit fission product leakage following the design basis LOCA. Because the proposed change does not alter the plant design or test method, only the frequency of measuring Type A, B and C leakage, the proposed change does not directly result in an increase in containment leakage. However, decreasing the test frequency can increase the probability that an increase in containment leakage could go undetected for an extended period of time. Based upon the results of the periodic containment Type A or Integrated Leak Rate Tests (ILRTs) and Type B and C or Local Leak Rate Tests (LLRTs) surveillance tests, this is not expected during the remaining life of the plant. The risk resulting from the proposed changes is as follows:

Type A Testing

NUREG/CR-4330 (NRC86) found that the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of the containment. It is also determined that on an expected individual dose basis, the effect of containment leakage is small.

Industry wide, ILRTs have only found a small fraction of the leaks that exceed current acceptance criteria. Only three percent of all leaks are detected by ILRTs, and therefore, by extending Type A testing intervals, only three percent of all leaks have a potential for remaining undetected for longer periods of time. In addition, when leakage has been detected by ILRTs, the leakage rate has been only about two times the allowable leakage rate.

NUREG-1493, ''Performance-Based Containment Leakage Test Program'', found that these observations, together with the insensitivity of reactor accident risk to the containment leakage rate, show that reducing the Type A leakage test frequency would have a minimal impact on public risk.

Type B and C Testing

NUREG-1493 found that while Type B and C tests can identify the vast majority (greater than 95 percent) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. The risk model used in NUREG-1493 suggests that the number of components tested would be reduced by about 60 percent with less than a three-fold increase in the incremental risk due to containment leakage. Since, under existing requirements, leakage contributes less than 0.1 percent of overall accident risk, the overall impact is very small. NUREG-1493 found that while the extended testing intervals for Type B and C tests led to minor increases in potential offsite dose consequences the actual decrease of on-site (worker) doses would be reduced in proportion to the number of Type B or C tests not performed.

EPRI Research Project Report TR-104285, ''Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals,'' also concluded that a relaxation of the test intervals for Type B and C penetrations results in a negligible increase in total plant risk.

Based on the above VYNPC [Vermont Yankee Nuclear Power Corporation] has concluded that the proposed change will not result in a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed 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 initiation of any accidents. This change involves the reduction in Type A, B, and C test frequency. The methods of performing the tests are not changed. No new accident modes are created by extending the testing intervals. No safety-related equipment or safety functions are altered as a result of this change. Extending the test frequency has no influence over nor does it contribute to, the possibility of a new or different kind of accident or malfunction from those previously analyzed.

Based upon the above, VYNPC has concluded that the proposed change will not create the possibility of a new or different kind of accident from those previously evaluated.

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

As stated in the Technical Support Document (TSD) for the NRC's Option B to

Appendix J rule change, NUREG-1493 concludes a reduction in the frequency of Type A testing from the current three per ten years to one per ten years leads to an imperceptible increase in risk. It also concludes that a reduction in the frequency of Type B testing of electrical penetrations should be possible with no adverse impact on risk. A vast majority of leakage paths are identified by Type C testing of containment isolation valves and, based on the model of component failure with time, performance-based alternatives to the current Type C testing intervals are feasible without significant risk impacts.

4.7.A.3

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

The proposed change does not result in any hardware or operating procedure changes. Closed and de-activated automatic valves, closed manual valves or blind flanges that serve as primary containment isolation valves are not assumed to be initiators of any analyzed event. The role of these devices is to isolate containment during analyzed events, thereby limiting consequences. The change establishes compensatory measures using closed and de-activated automatic valves, closed manual valves or blind flanges as an isolation barrier which is equivalent to those already included in the current Technical Specifications. The proposed change does not introduce any new failure modes, such that a single active failure could allow a primary containment release through an un-isolated path. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This change does not result in any changes to equipment design or capabilities or the operation of the plant. The change still ensures the primary containment boundary is maintained. Thus, this change does 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?

Closed and de-activated automatic valves, closed manual valves or blind flanges which are used to satisfy the compensatory measures of 4.7.A.3 are primary containment isolation devices will be leak tested per the PCLRTP. In addition, the Technical Specification establishes these devices as an isolation barrier that cannot be adversely affected by a single active failure. As a result, any reduction in a margin of safety will be insignificant and offset by the benefit gained with equivalent compensatory measures to ensure the primary containment boundary is maintained, which reduces unnecessary plant shutdown transients.

Table 4.7.2 Editorial Change

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

This change updates the information presented in this Table to reflect current practice. The methods of maintaining an inerted containment and differential pressure between the drywell and suppression pool have been previously docketed. The valves to now be shown normally closed on the Table are large (6'' and 18'') purge valves and the valves to be shown as normally open to provide makeup nitrogen are both 1'' in size. The probability of an accident is not significantly increased, since the subject valves are not considered to be initiators of any accident previously evaluated. The consequences of an accident are not significantly increased, since each of the subject valves receives a close signal from PCIS [primary containment isolation system]. In addition, PCIS closure of the two one inch valves will terminate the associated release pathway more rapidly than the existing valve lineup reflected on the Table. Thus it is concluded that this change will not involve any significant increase in the probability or consequences of an accident previously evaluated.

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

All four valves whose listed normal positions are proposed to be changed are PCIS valves and receive the same closing signal. All are tested in accordance with our Appendix J and IST [inservice testing] programs. No changes in equipment design or operation are proposed, only the listed normal positions of the subject valves. Thus, this change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The valves to be listed as normally open are significantly smaller and faster closing than the purge valves currently listed as open. Thus the change in the listed normal position of these four valves provides a more conservative initial condition than is currently depicted in Table 4.7.2. No changes in equipment design or operation are proposed. Thus, it is concluded that there is no 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 location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301

Attorney for licensee: R. K. Gad, III, Ropes and Gray, One International Place, Boston, MA 02110-2624

NRC Project Director: Ronald B. Eaton, Acting

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

Date of amendment request: August 14, 1997 (TSCR 199)

Description of amendment request: These amendments would revise: TS 15.4.2.B. ''In-Service Inspection and Testing of Safety Class Components Other than Steam Generator Tubes,'' to modify item 2 to change the reference from TS 15.4.4 to the Containment Leakage Rate Testing Program; TS 15.6.12.A.1, ''Containment Leakage Rate Testing Program,'' to eliminate the one-time requirement for Unit 2 Type A testing since the testing has been completed; and TS Bases 15.4.4 to delete the specific bases for containment purge valve testing and to delete a reference that is no longer used.

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

1. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed administrative changes correct discrepancies in the Technical Specifications introduced as a result of Amendment 169 to Operating License DPR-24 for Point Beach Nuclear Plant Unit 1 and Amendment 173 to Operating License DPR-27 for Point Beach Nuclear Plant Unit 2. These changes correct references to containment isolation valve testing in the Specifications and Bases. These amendments were evaluated as acceptable in a safety evaluation dated October 9, 1996. Therefore, these changes do not result in an increase in the probability or consequences of any accident previously evaluated.

The Point Beach Nuclear Plant Unit 2 containment was tested and found acceptable within the maximum interval defined by a one-time Technical Specifications requirement. Subsequent testing will be performed in accordance with the approved testing program defined by Technical Specifications 15.6.12. Therefore, the Technical Specification requirements are met. These requirements are established to ensure the containment performs and is maintained as designed and assumed in the safety analyses. The removal of the one-time specific periodicity requirements for the Unit 2, Type A containment integrated leak rate test does not result in a significant increase in the probability or consequence of any accident previously evaluated.

2. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the Technical Specifications do not change the requirements for the Point Beach Nuclear Plant containments to perform as designed and evaluated in the safety analyses. Test requirements in the Technical Specifications continue to meet the standards evaluated and approved by the NRC to ensure the containments continue to perform as designed and analyzed. Administrative discrepancies in the Specifications and bases are also corrected. Therefore, no new or different kind of accident from any accident previously evaluated is created.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not involve a significant reduction in a margin of safety.

The proposed changes to the Technical Specifications ensure consistency with Amendment 169 to Point Beach Nuclear Plant Unit 1 Operating License DPR-24 and Amendment 173 to Point Beach Nuclear Plant Unit 2 Operating License DPR-27. Testing of the Unit 2 containment has been performed within the maximum time limit allowed by the one-time test requirement of Technical Specification 15.6.12. Testing requirements continue to meet NRC requirements and ensure the containment continues to operate as designed and analyzed. Administrative corrections to the Specifications and bases ensure consistency with previously approved amendments. Therefore, a margin of safety is not reduced.

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

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

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

NRC Project Director: John N. Hannon

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

Date of amendment request: July 29, 1997

Description of amendment request: This license amendment request revises the wording of Action Statement 5.a to Technical Specification Table 3.3-1. ''Reactor Trip System Instrumentation.'' This action statement prescribes a set of actions to be accomplished when a source range neutron detector is inoperable with the plant shut down. The proposed wording change will clarify the times and order in which these actions are to be performed.

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.

In MODE 3, 4, or 5 with the rod control system capable of rod withdrawal or rods not fully inserted, the source range neutron detectors provide a reactor trip signal on high neutron flux to provide core protection against an uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition. This trip function is actuated when either of two independent source range channels indicates a neutron flux level above a preselected manually adjustable setpoint. If the rod control system is not capable of rod withdrawal with rods fully inserted, the source range detectors are not required to trip the reactor.

NUREG-1431, Revision 1, ''Standard Technical Specifications Westinghouse Plants,'' allows one source range neutron detector to be out of service for up to 48 hours. One additional hour is allowed to open the reactor trip breakers and suspend operations involving the addition of positive reactivity. This was the same action sequence prescribed for the source range neutron detectors prior to the implementation of Amendment No. 96 to the Wolf Creek Technical Specifications, which inadvertently resulted in an ambiguous rewording of the action. The proposed rewording of the action statement clarifies the proper timing of the required actions, and is consistent with NUREG-1431, Revision 1.

The proposed change does not introduce any new potential accident initiating conditions and does not alter any plant operating procedures or method of operation of any plant components or systems. Allowing positive reactivity changes during the 48 hour period in which one source range neutron detector is inoperable is acceptable since the remaining detector will still provide the reactor trip function and control room indication when the reactor trip breakers are closed, and control room indication when the reactor trip breakers are open. This is consistent with the provisions in NUREG-1431, Revision 1. Thus, the proposed change does not affect any system's ability to mitigate the consequences of an accident and will not increase the probability of occurrence of any 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.

The proposed change does not affect the method of operation of any plant component or system, and does not create any new, or alter any existing, accident initiators. The proposed change clarifies that positive reactivity changes may be allowed during the 48 hour period in which a source range neutron detector is inoperable, as provided for in NUREG-1431, Revision 1. This action does not affect the capability of the remaining source range neutron detector to provide a reactor trip signal on high neutron flux during this period when the reactor trip breakers are closed, nor does it affect the ability of the remaining detector of providing control room indication. This function of the source range neutron detectors is discussed in Chapter 15 of the Wolf Creek Updated Safety Analysis Report. This proposed change does not modify any existing plant equipment, add any new plant equipment, or alter any component or system operating parameters or procedures. Therefore, this proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The source range neutron detectors provide a reactor trip function during shutdown conditions when the reactor trip breakers are closed. When the reactor trip breakers are open they provide control room alarm/indication, only. The proposed change clarifies that positive reactivity changes may be allowed during the 48 hour period in which a source range neutron detector is inoperable. This is consistent with the provisions in NUREG-1431, Revision 1 and with Wolf Creek Technical Specification Table 3.3-1, Action 5.a, prior to the implementation of Amendment No. 96. In Amendment No. 96 the wording of this action was changed such that this allowance was no longer clear. With one source range neutron detector inoperable with the reactor trip breakers closed, the reactor trip on high neutron flux function is still provided by the remaining source range neutron detector. With one source range neutron detector inoperable with the reactor trip breakers open, control room indication of high neutron flux is still provided. As stated above, this is consistent with NUREG-1431, Revision 1, as well as with the action requirements prior to the implementation of Amendment No. 96. This proposed change, then, does not affect the margin of safety provided by the source range neutron detectors.

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.

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 amendments request: July 25, 1997

Brief description of amendments: The proposed amendments would modify Technical Specification (TS) 4.0.5.f in a manner that would allow exceptions to the NRC staff's positions on intergranular stress corrosion cracking in boiling water reactor austenitic stainless steel piping, where specific written relief has been granted by the NRC. TS 4.0.5.f now requires that the Brunswick Steam Electric Plant, Units 1 and 2, Inservice Inspection program be performed in accordance with the positions identified in NRC Generic Letter 88-01. Date of publication of individual notice in **Federal Register:** August 12, 1997 (62 FR 43187)

Expiration date of individual notice: September 11, 1997

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

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

Date of application for amendment: August 4, 1997

Brief description of amendment: The proposed amendment would revise the Technical Specifications to extend the frequency for certain surveillances related to the emergency diesel generators. Date of publication of individual notice in the FEDERAL REGISTER:August 12, 1997 (62 FR 43189)

Expiration date of individual notice: September 11, 1997

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

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: August 6, 1997

Description of amendment request: The proposed amendment would revise Technical Specification Table 2.2-1 and 3/4.2.5 to allow the reactor coolant system total flow to be determined using cold leg elbow tap differential pressure measurements. Date of individual notice in the **Federal Register:** August 14, 1997 (62 FR 43556)

Expiration date of individual notice: September 15, 1997

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

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

Commonwealth Edison Company, Docket No. 50-455, Byron Station, Unit No. 2, Ogle County, Illinois, Docket No. STN 50-457, Braidwood Station, Unit No. 2, Will County, Illinois

Date of application for amendments: May 24, 1997, as supplemented by letters dated May 31, June 20 and June 24, 1997

Brief description of amendments: The amendments revise Technical Specification 4.5.2.b.1 to include the use of Ultrasonic Testing (UT) to verify that the emergency core cooling system (ECCS) is completely filled with water. For the ECCS subsystem with high point vent valves in direct communication with the operation system, UT is acceptable in lieu of physically opening the vents.

Date of issuance: August 13, 1997

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

Amendment Nos.: 91 and 84

Facility Operating License Nos. NPF-66 and NPF-77: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 10, 1997 (62 FR 31633) The May 31, June 20, June 24, and July 18, 1997, submittals provided additional clarifying information that did not change the proposed initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 13, 1997. No significant hazards consideration comments received: No

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

Commonwealth Edison Company, Docket Nos. 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: June 9, 1997

Brief description of amendments: The amendments authorize a change to the realistic dose values for the process gas system rupture in Section 15.0 of the

Byron/Braidwood (B/B) Updated Final Safety Analysis Report (UFSAR). During preparation of a UFSAR change package, ComEd discovered that the Final Safety Analysis Report (FSAR) had not been updated to correct an error from the previous revision of the dose calculation. Since the correct dose value is greater than that previously reported, the consequences of the accident had increased, and an unreviewed safety question resulted.

Date of issuance: August 13, 1997

Effective date: August 13, 1997

Amendment Nos.: 92, 92, 85, 85

Facility Operating License Nos. NPF-37, NPF-66, NPF-72 and NPF-77: The amendments authorize a change to the Byron/Braidwood UFSAR.

Date of initial notice in **Federal Register:** July 10, 1997 (62 FR 37079). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 13, 1997. No significant hazards consideration comments received: No

Local Public Document 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

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

Date of application for amendment: April 30, 1997

Brief description of amendment: The amendment revises the Big Rock Point Plant license and technical specifications to reflect the licensee's name change from ''Consumers Power Company'' to ''Consumers Energy Company.''

Date of issuance: August 14, 1997

Effective date: August 14, 1997

Amendment No.: 119

Facility Operating License No. DPR-6: Amendment revised the license and the Technical Specifications.

Date of initial notice in **Federal Register:** June 4, 1997 (62 FR 30630) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 14, 1997. No significant hazards consideration comments received: No.

Local Public Document location: North Central Michigan College, 1515 Howard Street, Petoskey, Michigan 49770

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

Date of application for amendment: May 8, 1997, as supplemented June 10, and July 25, 1997

Brief description of amendment: The amendment incorporates additional NRC-approved topical reports into the Technical Specifications (TS).

Date of issuance: August 12, 1997

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

Amendment No.: 202

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

Date of initial notice in **Federal Register:** June 4, 1997 (62 FR 30633) The June 10 and July 25, 1997, letters provided clarifying information that did not change the scope of the May 8, 1997, application or the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 12, 1997. No significant hazards consideration comments received: No

Local Public Document location: Law/Government Publications Section, State Library of Pennsylvania (REGIONAL DEPOSITORY), Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105

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

Date of application for amendments: February 29, 1996 (AEP:NRC:1232), and supplemented November 15, 1996 (AEP:NRC:1232A), and February 4, 1997 (AEP:NRC:1232B)

Brief description of amendments: The amendments revise the Technical Specifications and associated bases to increase the minimum borated water volume in the boric acid storage system and decrease the required boron concentration.

Date of issuance: August 7, 1997

Effective date: August 7, 1997, with full implementation when the required plant modifications are completed, but not later than August 31, 1998.

Amendment Nos.: 216 and 200

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

Date of initial notice in **Federal Register:** April 24, 1996 (61 FR 18172) The November 15, 1996, and February 4, 1997, supplements only provided the schedule for the plant modifications and procedure changes associated with this amendment and did not change the staff's proposed determination of no significant hazards consideration. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 7, 1997.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 20, 1996

Brief description of amendments: The amendments reduce the frequency and scope of reactor coolant pump flywheel inspections.

Date of issuance: August 8, 1997

Effective date: August 8, 1997, with full implementation within 45 days.

Amendment Nos.: 217 and 201

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

Date of initial notice in **Federal Register:** June 18, 1997 (62 FR 33126) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 8, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: September 13, 1996, as supplemented by letter dated September 25, 1996

Brief description of amendment: The amendment revised Technical Specification 5.5.B to designate the President, Maine Yankee as the responsible official for matters related to the Nuclear Safety Audit and Review (NSAR) Committee. The amendment includes some minor editorial changes to the same technical specification.

Date of issuance: August 8, 1997

Effective date: August 8, 1997, to be implemented within 30 days of the date of issuance.

Amendment No.: 159

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

Date of initial notice in **Federal Register:** November 6, 1996 (61 FR

57487) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 8, 1997. No significant hazards consideration comments received: No.

Local Public Document location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578

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

Date of application for amendment: June 13, 1997

Brief description of amendment: The amendment modifies Technical Specification (TS) Surveillance Requirement 4.4.1.3.3 to be consistent with the requirements of TS 3.4.1.3. Specifically, the change brings TS 4.4.1.3.3 into agreement with TS 3.4.1.3 by requiring that the specified reactor coolant and/or residual heat removal system loops be verified in operation and circulating reactor coolant at least once per 12 hours during Mode 4.

Date of issuance: August 5, 1997

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

Amendment No.: 145

Facility Operating License No. NPF-49: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 2, 1997 (62 FR 35850) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 5, 1997. No significant hazards consideration comments received: No.

Local Public Document location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut 06385

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

Date of application for amendment: January 27, 1997, as supplemented May 16, 1997

Brief description of amendment: The amendment changes the Technical Specifications to permit control rod misalignment of up to plus or minus 18 steps when the core thermal power is less than 85% of rated power.

Date of issuance: August 11, 1997

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

Amendment No.: 176

Facility Operating License No. DPR-64: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 19, 1997 (62 FR 33445) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 11, 1997. No significant hazards consideration comments received: No

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

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

Date of application for amendment: March 26, 1997

Brief description of amendment: The amendment revises TS 4.5.2.a for the two charging/high head safety injection (HHSI) pump cross connect valves (XVG-8133A and XVG-8133B) and charging pump mini-flow header isolation valve (XVG-8106) in the emergency core cooling system (ECCS). The proposed amendment adds these valves to the list of valves in TS Surveillance Requirement 4.5.2.a on page 3/4 5-4, consequently these valves will be verified once every 12 hours to indicate that they are in the required position with power to the valve operators removed.

Date of issuance: August 8, 1997

Effective date: August 8, 1997

Amendment No.: 136

Facility Operating License No. NPF-12: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** May 21, 1997 (62 FR 27801) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 8, 1997. No significant hazards consideration comments received: No

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

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

Date of application for amendment: November 14, 1995, as supplemented July 11, 1996 and July 24, 1997

Brief description of amendment: The amendment revises Technical Specification 3/4.8.4.2 for motor-operated valves thermal overload protection and bypass devices at Virgil C. Summer Nuclear Station.

Date of issuance: August 13, 1997

Effective date: August 13, 1997

Amendment No.: 137

Facility Operating License No. NPF-12: Amendment adds a new License Condition and revises the Technical Specifications.

Date of initial notice in **Federal Register:** December 20, 1995 (60 FR 65684) The July 11, 1996, and July 24, 1997 submittals contained clarifying information only and did not change the proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 13, 1997. No significant hazards consideration comments received: No

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

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee Date of application for amendments: September 26, 1996, as supplemented on August 12, 1997 (TS 96-04)

Brief description of amendments: The amendments change the Technical Specifications (TS) by relocating the fire protection program details to the Updated Final Safety Analysis Report and Fire Protection Plan in accordance with Generic Letters 86-10 and 88-12.

Date of issuance: August 12, 1996

Effective date: August 12, 1996

Amendment Nos.: 227 and 218

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

Date of initial notice in **Federal Register:** July 2, 1997 (62 FR 35843) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 12, 1997. No significant hazards consideration comments received: No

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

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

Date of application for amendments: August 22, 1996, as revised July 14, 1997

Brief description of amendments: These amendments revise Section 3.A of Facility Operating Licenses DPR-24 and

DPR-27 from a licensed power level of 1518 megawatts thermal to 1518.5 megawatts thermal. A similar revision is made in the bases of Technical Specification 15.3.1.B, ''Pressure/ Temperature Limits.''

Date of issuance: August 6, 1997

Effective date: August 6, 1997

Amendment Nos.: 175 and 179

Facility Operating License Nos. DPR-24 and DPR-27: Amendments revised the licenses.

Date of initial notice in **Federal Register:** October 9, 1996 (61 FR 52972) The July 14, 1997, supplement provided a corrected bases page and did not affect the staff's no significant hazards considerations determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 6, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 12, 1997, as supplemented on March 11, 1997 (TSCR 196)

Brief description of amendments: These amendments revise Point Beach Nuclear Plant's (PBNP) Technical Specifications (TSs) to relocate turbine overspeed protection specifications, limiting conditions for operation, surveillance requirements, and associated bases from TS Section 15.3.4, ''Steam and Power Conversion System,'' and Section 15.4.1, ''Operational Safety Review,'' to the Final Safety Analysis Report (FSAR) in accordance with Generic Letter 95-10.

Date of issuance: August 6, 1997

Effective date: These license amendments are effective as of the date of issuance and shall be implemented by incorporating the turbine overspeed protection specifications, limiting conditions for operation, surveillance requirements, and associated bases into the FSAR by June 30, 1998.

Amendment Nos.: 176 and 180

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

Date of initial notice in **Federal Register:** April 23, 1997 (62 FR 19838) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 6, 1997. No significant hazards consideration comments received: No.

Local Public Document location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241

Dated at Rockville, Maryland this 20th day of August 1997.

For the Nuclear Regulatory Commission

John A. Zwolinski,

Acting Director, Division of Reactor Projects - I/II, Office of Nuclear Reactor Regulation.

[Doc. 97–22635 Filed 8–26–97; 8:45 am]

BILLING CODE 7590–01–F

POSTAL SERVICE

Specifications for Information Based Indicia Program (IBIP) Postal Security Devices and Indicia (Postmarks)

AGENCY: Postal Service.

ACTION: Notice of USPS response to public comments and availability of Specifications.

SUMMARY: The Postal Service received hundreds of comments in response to our **Federal Register** notices on the draft specifications for Information Based Indicia Program Postal Security Device (PSD) and Indicium. The Postal Service has reviewed all those comments and developed a response. Some of the comments were within the scope of the draft proposed specifications and some of the comments were not. Those within the scope of the draft proposed specifications have responses included herein. Those outside the scope of the draft proposed specifications will be included in subsequent responses. Some of the topics not dealt with herein include key management, host system specifications, cash management, certificate authority, product life-cycle management, mail classes, customer usage requirements, market research, procurement policy, product submission requirements, product/ service provider infrastructure, and program development activities.

ADDRESSES: Copies of the draft PSD and Indicium specifications dated July 23, 1997, may be obtained from Ed Zelickman, United States Postal Service, 475 L'Enfant Plaza SW Room 1P801, Washington, DC 20260–6807. Comments should be submitted to the same address. These documents supersede all previously issued Indicium and PSD Specifications. Copies of all written comments may be inspected between 9 a.m. and 4 p.m., Monday through Friday, at the above address.

DATES: All written comments must be received on or before October 27, 1997.

FOR FURTHER INFORMATION CONTACT: Ed Zelickman at (202) 268–3940.

SUPPLEMENTARY INFORMATION: The Postal Service received hundreds of comments on the proposed draft Information Based Indicia Program (IBIP) Indicia and Postal Security Device specifications (62 FR 37631, July 14, 1997). Those outside the scope of the draft proposed specifications will be dealt with in subsequent specifications and documents and will not be addressed herein.

Indicium Specification

Many comments were received regarding Indicium data contents. Generally, these comments fall into six categories:

1. Reserve Field Usage

The specific use of the reserved field has not been defined. Product Service Providers are welcome to suggest how the customer or service provider could best use this field. This field was installed in the indicia data set as a customer defined field.

2. The PSD Certificate in the Indicium

The USPS has included in the initial draft the PSD certificate in the indicia. The removal of the certificate in subsequent releases of these specifications is dependent upon the key management infrastructure.

3. Size and Format of the Indicium Fields

The USPS feels that all fields (except the reserve field) in the indicia contribute to either the security/ verification of the indicia or the audit control of IBIP products. We will continue to explore replacement methods in an effort to reduce indicia size.

4. Rate Category Definition

The Rate category is defined in the draft DMM and CFR policies and is not defined in these documents.

5. Ascending Register as a Data Element

The ascending register along with the device ID provides absolute uniqueness to each indicium. The inclusion of the ascending register also provides one audit control data element.

6. Special Purpose Field

The special purpose field is included as an audit control field. This data element within the barcode should match the human readable value on the mailpiece. If these two do not match, this could be a fraud indicator.

Many comments were received regarding the use of digital signatures and associated technology. Specifically, a question arose on use of varying hash