April 28, 2004

Mr. Joseph M. Solymossy
Site Vice President
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: LICENSE AMENDMENT REQUEST DATED MARCH 25, 2003, FOR "SAFETY ANALYSES TRANSITION" (TAC NOS. MB8128 AND MB8129)

Dear Mr. Solymossy:

The U.S. Nuclear Regulatory Commission (NRC, Commission) has issued the enclosed Amendment No. 162 to Facility Operating License No. DPR-42 and Amendment No. 153 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 25, 2003, as supplemented by your letters dated June 16, 2003, January 14, February 11, February 23, and April 7, 2004.

The amendments revise the TSs to include implementation of relaxed axial offset control of the reactor core through changes in TS 3.2.1 and TS 3.2.3; relocation of selected operating parameters from TS 2.0, TS 3.1.8 and TS 3.3.1 to the Core Operating Limit Report (COLR) and the revised pressurizer pressure-low allowable value in TS Table 3.3.1-1. The TS changes also include, in TS 5.6.5, the topical reports documenting the NRC-approved methodologies that are used to support COLR implementation.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Mahesh Chawla, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 162 to DPR-42
2. Amendment No. 153 to DPR-60
3. Safety Evaluation

cc w/encls: See next page

Mr. Joseph M. Solymossy April 28, 2004
Site Vice President
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cc w/encls: See next page

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 162
License No. DPR-42

1. The Nuclear Regulatory Commission (the Commission) has found that:

    A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated March 25, 2003, as supplemented by your letters dated June 16, 2003, January 14, February 11, February 23, and April 7, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

    B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

    C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

    D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

    E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-42 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 162, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 28, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 162

FACILITY OPERATING LICENSE NO. DPR-42

DOCKET NO. 50-282

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
2.0-1	2.0-1
3.1.8-1	3.1.8-1
3.2.1-2	3.2.1-2
3.2.1-3	3.2.1-3
3.2.3-1	3.2.3-1
3.2.3-2	----------
3.2.3-3	----------
3.2.3-4	----------
3.3.1-18	3.3.1-18
3.3.1-23	3.3.1-23
3.3.1-24	3.3.1-24
5.0-34	5.0-34
5.0-35	5.0-35
5.0-36	5.0-36
5.0-37	5.0-37
5.0-38	5.0-38
5.0-39	5.0-39
5.0-40	5.0-40
--------	5.0-40a

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 153
License No. DPR-60

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated March 25, 2003, as supplemented by your letters dated June 16, 2003, January 14, February 11, February 23, and April 7, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-60 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 153, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 28, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 153

FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NO. 50-306

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
2.0-1	2.0-1
3.1.8-1	3.1.8-1
3.2.1-2	3.2.1-2
3.2.1-3	3.2.1-3
3.2.3-1	3.2.3-1
3.2.3-2	----------
3.2.3-3	----------
3.2.3-4	----------
3.3.1-18	3.3.1-18
3.3.1-23	3.3.1-23
3.3.1-24	3.3.1-24
5.0-34	5.0-34
5.0-35	5.0-35
5.0-36	5.0-36
5.0-37	5.0-37
5.0-38	5.0-38
5.0-39	5.0-39
5.0-40	5.0-40
--------	5.0-40a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 162 TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 153 TO FACILITY OPERATION LICENSE NO. DPR-60

NUCLEAR MANAGEMENT COMPANY, LLC

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By application dated March 25, 2003, as supplemented by your letters dated June 16, 2003, January 14, February 11, February 23, and April 7, 2004, the Nuclear Management Company (NMC), LLC (the licensee), requested changes to the Technical Specifications (TSs) for the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2. The proposed changes would revise the TS to include implementation of relaxed axial offset control of the reactor core through changes in TS 3.2.1 and TS 3.2.3; relocation of selected operating parameters from TS 2.0, TS 3.1.8 and TS 3.3.1 to the Core Operating Limit Report (COLR) and the revised pressurizer pressure-low allowable value in TS Table 3.3.1-1. In TS 5.6.5, the changes also include the topical reports documenting the NRC-approved methodologies that are used to support COLR implementation.

2.0 REGULATORY EVALUATION

Prairie Island Updated Safety Analysis Report (USAR), Section 3.1.2.1, “Reactor Core Design,” requires “the reactor core with its related controls and protective systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits...” Also, “the core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations...”

Prairie Island USAR, Section 4.3.1.2.1, “Reactor Coolant Pressure Boundary,” requires that “the reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime.” Also, “the Reactor Coolant System in conjunction with its control and protective provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, and maintain the stresses within applicable code stress limits.”

As for TS changes to implement the COLR, Generic Letter (GL) 88-16 (Ref. 7) requires that the parameters to be relocated to the COLR be cycle-specific and be calculated using approved methods with these methods listed in Administrative Control Section of the TSs.

Also, Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, specifies the Commission's regulatory requirements related to the content of TSs. The NRC staff's evaluation of the acceptability of the proposed changes on TS 3.3.1 is based on 10 CFR 50.36, "Technical specifications." Paragraph (A) of 10 CFR 50.36(c)(1)(ii) requires that a limiting safety system setting be specified for a variable on which a safety limit has been placed and that the setting be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. Specifically, 10 CFR 50.36(c)(2)(ii) sets forth four criteria to be used in determining whether a limiting condition for operation (LCO) is required to be included in TSs. These criteria are:

> Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant system (RCS) pressure boundary;
>
> Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design-basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
>
> Criterion 3: A structure, system or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; and
>
> Criterion 4: An SSC which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

An LCO must be included in the TS for an SSC which satisfies any of the criteria specified in 10 CFR 50.36(c)(2)(ii). Since the Standard TSs (STSs) are developed based on the 10 CFR 50.36 requirements, and the PINGP uses the Westinghouse-designed nuclear steam supply system, the NRC utilizes NUREG-1431 (Ref. 8), "Standard Technical Specifications - Westinghouse Plants," in its review of the proposed TSs (Refs. 1, 2, 3, 24, 25, and 26) for the PINGP. The NRC staff evaluates the acceptability of the Westinghouse non-loss-of-coolant-accident (non-LOCA) methodologies and confirms that the use of the methodologies is within the approved range and the results of non-LOCA analyses are in compliance with the requirements of Sections 3.1.2.1 and 4.3.1.2.1 of USAR. Also, the NRC staff evaluates TS changes for COLR implementation in accordance with the GL 88-16 guidance that allows the cycle-specific parameters to be relocated to the COLR provided that the parameters are calculated using the approved methods with these methods listed in Administrative Control Section of the TSs.

Regulatory Guide (RG) 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," describes a method acceptable to the NRC for complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initiated within and remain within the TS limits. RG 1.105, Revision 3, endorses Part 1 of Instrument Society of America (ISA) Standard ISA-S67.04-1994, "Setpoints for Nuclear Safety-Related Instrumentation." The NRC staff utilized the regulatory guidance in RG 1.105 and ISA-S67.04-1994 in performing this review.

3.0 TECHNICAL EVALUATION

Background

In a March 25, 2003, letter (Ref. 1), supplemented by letters of June 16, 2003 (Ref. 2), January 14, 2004 (Ref. 3) , February 11 (Ref.24), February 23, 2004 (Ref. 25), and April 7, 2004 (Ref.26), NMC, the licensee for the PINGP, Units 1 and 2, proposed a license amendment request for approval of conversion to Westinghouse non-LOCA methodologies and related TS changes.

NMC has performed non-LOCA transient analyses that support operation of the PINGP with its own methods for many years. It plans to rely on Westinghouse methodologies to perform non-LOCA transient analyses in the future and, therefore, requested the approval of use of Westinghouse methodologies for non-LOCA analyses in this amendment request. In the process of converting to Westinghouse safety analyses, NMC also proposed TS changes in the following areas: (1) implementation of relaxed axial offset control (RAOC) of the reactor core; (2) implementation of Westinghouse methodologies for determining selected core operating parameter values; (3) relocation of selected operating parameters to the COLR; and (4) a decrease of the pressurizer pressure-low Allowable Value. The TSs involved are: TS 2.1.1, “Reactor Core SLs;” TS 3.1.8, “Physics Tests Exceptions - Mode 2;” TS 3.2.1, “Heat Flux Hot Channel Factor $F_Q(Z)$;” TS 3.2.3, “Axial Flux Difference (AFD);” TS Table 3.3.1-1, “Reactor Trip System Instrumentation;” and TS 5.6.5, “Core Operating Limits Report (COLR).”

The proposed TS changes would reflect conversion to Westinghouse non-LOCA methodologies used in transient analyses, increase plant availability and operating flexibility.

On July 26, 2002, the NRC approved the conversion of the PINGP, Units 1 and 2, TS to the Improved Technical Specifications (ITS). In the safety evaluation report (SER) for the ITS conversion, Section G.2.1, “Instrument Setpoint Methodology and New Allowable Value,” specifically addresses the PINGP instrument setpoint methodology.

The licensee submitted Revision 0 of Section 3.3.4.1 of the Engineering Manual, which is the Engineering Design Standard for Instrument Setpoint/Uncertainty Calculations and is used to calculate instrument allowable values. The NRC staff reviewed and accepted licensee’s setpoint methodology based on regulatory requirements and guidance. The calculations submitted in this license amendment request (LAR) are performed in accordance with this methodology.

Review Scope

In support of the request for approval of conversion to Westinghouse methods for non-LOCA analyses, the licensee provided information in References 1 and 2 for the NRC staff to review. References 1 and 2 provide a description of the proposed TS changes with associated supporting justification, as well as reference WCAP Topical Reports (TRs) documenting Westinghouse methods that will be used in the non-LOCA analysis for the PINGP. The licensee indicated that the Westinghouse TRs were previously approved by the NRC and claimed that the TRs are acceptable for referencing in the specific plant licensing application. However, the staff found that additional information was needed to justify the acceptability of the TRs to the specific plant.

Methodologies or computer codes used to support licensing applications involving LOCA or non-LOCA analysis are, in general, documented in TRs and are reviewed by the NRC staff on generic bases. The NRC defines the bases for acceptance of the TRs with restrictions on the applications, as appropriate, in the NRC staff's safety evaluations (SEs) approving the TRs. A generic TR describing a methodology or computer code does not provide full justification for each plant-specific application. In the situation where a plant-specific license amendment request references a TR that has not been previously applied to the specific-plant, the NRC staff will request that the licensee submit a plant-specific analysis to demonstrate applicability of the TR. During the review, the NRC staff requested the licensee to provide additional information to justify the TR(s) applicability for the PINGP. Specifically, the NRC staff requested the licensee to perform an analysis for the updated final safety analysis report (UFSAR) limiting events with the Westinghouse methods, and to demonstrate that the use of the methods is within the approved ranges and the results of the analysis meet the applicable acceptance criteria. The NRC staff also asked the licensee to address its compliance with each of the SE limitations imposed on use of the applicable TRs. Further, the NRC staff requested that in order to satisfy the GL 88-16 guidance for COLR implementation, the approved TRs should be included in Administrative Control Section of the TSs if they were used to support relocation of cycle-specific parameters to COLR.

In response, the licensee provided information in Reference 3 with several attachments: Attachments 2 through 4 (Ref. 4) provide the licensee's responses to the NRC staff's request for additional information (RAI); Attachment 5 (Ref. 5) is a licensing report documenting the non-LOCA analysis performed with Westinghouse methods; Attachment 6 (Ref. 6) addresses the licensee's compliance with the SE(s) restrictions on applicability of TRs; Attachment 7 documents low pressurizer pressure trip setpoint calculations; and Attachments 8 and 9 consist of marked-up and retyped copies of the additional TS changes as a result of an evaluation addressing the NRC staff's RAI. After the review of the licensee's responses (Refs. 3 through 6), the NRC staff issued a follow-up RAI requesting the licensee to address the issues of steam generator (SG) water level measurement uncertainties and effects of delay time of reactor coolant pumps (RCPs) coastdown on the applicable transients behavior. The licensee provided its response in Reference 24. The NRC staff has reviewed the LAR with the associated supporting information (Refs. 1 & 2), and the responses to the RAI (Refs. 3 through 6, and 24). Based on its review, the NRC staff has prepared the following evaluation: Section 3.1 for Westinghouse methods and computer codes used for the transient analyses; Section 3.2 for transient analyses; and Section 3.3 for the TS changes.

3.1 Westinghouse Methodologies and Computer Codes used for Transient Analyses

The licensee proposed (Refs. 1 through 6) to use the following Westinghouse methodologies for non-LOCA analyses:

3.1.1 Determination of the Thermal Overpower ΔT and Overtemperature ΔT Trip Functions

The overpower ΔT and overtemperature ΔT trip functions are designed to provide protection against fuel centerline melt (overpower ΔT) and departure from nucleate boiling (DNB) (overtemperature ΔT) during transients. The licensee proposed to use the methodologies documented in WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Overtemperature ΔT Trip Functions," for determination of the ΔT trip functions. As described in WCAP-8745-P-A, the trip functions are represented by four terms: the leading term is an adjustable constant; the second term is dependent on the average coolant temperature, and is lead/lag compensated for instrumentation and piping delays; the third term accounts for the effects of coolant density and heat capacity on the relationship between ΔT and core power for the overpower ΔT trip function, and the effects of pressure on the design limit for the overtemperature ΔT trip function; the fourth term, f(ΔI), accounts for the effects of adverse power distributions and is dependent on the axial flux difference. The parameter values of the ΔT trip functions are determined based on the following design criteria: (1) the uranium dioxide melting temperature is not exceeded for 95 percent of the fuel rods at the 95 percent confidence level; and (2) at least a 95 percent probability that DNB does not occur at the limiting fuel rod at a 95 percent confidence level. To satisfy these design acceptance criteria, the licensee proposed to limit the calculated fuel centerline temperature to no greater than 4700 °F, and limit the minimum calculated DNBR no less than 1.17 for the WRB-1 DNBR correlation with the VIPRE-01 code applicable to the 14x14 OFA fuel in the PINGP.

The staff found that the proposed fuel centerline temperature limit is consistent with that specified in the approved Westinghouse topical report, WCAP-8745-P-A (Ref. 11), and the design DNB limit was previously approved by NRC for the 14x14 OFA fuel used in the PINGP core (Ref. 22). The licensee used the same ΔT trip functions (except for the values of the parameters) as those documented in WCAP-8745-P-A, and determined the values of parameters in the ΔT trip functions using NRC-approved design acceptance criteria. Therefore, the NRC concluded that use of the methodologies discussed in WCAP-8745-P-A is acceptable.

3.1.2 Relaxation of Constant Axial Offset Control

The current procedure, constant axial offset control (CAOC), for axial power distribution control requires that axial offset be kept within a narrow band (±5 percent specified in the PINGP COLR) of target value during normal plant operation in order to ensure that unallowed power shapes do not occur. The licensee proposed to use the relaxation of CAOC (RAOC) procedure documented in WCAP-10216-P-A (Ref. 10), "Relaxation of Constant Axial Offset Control/F_Q Surveillance Technical Specification," to increase operating flexibility. The result of the RAOC procedure is a curve of allowed ΔI, the axial flux difference (difference between the upper and lower excore detector readings), as a function of power. The RAOC procedure requires that the allowed axial flux difference (ΔI) be bounded by those used in the analysis for LOCA events and non-LOCA Condition II events. With the RAOC implemented, a TS must be established to require that the ΔI be maintained with the acceptable band as a function of power.

The NRC found in WCAP-10216-P-A that the RAOC procedure was previously approved for licensing applications for Westinghouse plants. Since the PINGP is a Westinghouse plant and the RAOC control limits will be determined by the approved Westinghouse LOCA and non-LOCA methods, the NRC staff concludes the licensee's use of the methods documented in WCAP-10216-P-A is acceptable.

3.1.3 Revised Thermal Design Procedure for Thermal-Hydraulic Analyses

The licensee proposed to use the revised thermal design procedure (RTDP) to perform statistical core thermal-hydraulic analyses. Unlike the deterministic method, in which the uncertainties of various plant and operating parameters are assumed simultaneously at their worst uncertainty limits in the safety analyses, the RTDP methodology statistically accounts for the system uncertainties in plant operating parameters, fabrication parameters, nuclear and thermal parameters, as well as the DNB correlation and computer codes uncertainties. The RTDP methodology establishes a design DNBR limit that statistically accounts for the effects of the key parameters on DNB. Therefore, when the RTDP methodology is used to perform thermal-hydraulic analyses, initial condition uncertainties can be neglected in the plant parameters that are sensitive to the DNBR calculations as they are already included in the RTDP DNBR limit. The RTDP methodology is documented in WCAP-11397-P-A, "Revised Thermal Design Procedure."

The design DNBR limit is calculated based on the system uncertainties in plant operating parameters and the uncertainties of the DNB correlation and computer codes used for the specific plant. During the review, the NRC staff requested the licensee to discuss its calculation of the design DNBR limit for the PINGP. In Reference 4, the licensee indicated that the RTDP in WCAP-11394-P-A and PINGP plant-specific uncertainties were used to determine the design DNBR limit. Specifically, power, temperature, pressure and flow uncertainties were chosen to bound the uncertainties for the PINGP. The uncertainty values for the WRB-1 correlation with VIPRE-01 were taken from WCAP-14565-P-A. Uncertainties for the thermal-hydraulic core and system codes (VIPRE-01 and RETRAN-02) were assumed to be the same as those assumed in WCAP-11397-P-A. The NRC staff found that the licensee's calculation of the design DNBR limit adequately follows the approved RTDP method described in WCAP-11397-P-A. Therefore, the NRC staff concluded that the use of the RTDP as documented in WCAP-11397-P-A to perform statistical core thermal-hydraulic analyses is acceptable.

Use of the RTDP methodology requires that the uncertainty of the plant-specific input parameters to the RTDP be verified. Four operating parameter uncertainties need to be addressed in the uncertainty analysis of the RTDP are:

(1) Reactor power
(2) Reactor coolant system flow
(3) Pressurizer pressure
(4) Reactor coolant system T_{avg}

Reactor power is continuously monitored, and the nuclear instrumentation system (NIS) power range reactor power indication is verified against the plant's computer-based thermal power monitor (TPM; also known as calorimetric reactor power indication) every 24 hours.

The licensee has submitted two calculation documents:

"SPCNI017: Unit 1 Calorimetric Uncertainty," and "SPCNI018: Unit 2 Calorimetric Uncertainty." The purpose of these calculations is to determine the error of the calorimetric calculation performed by the TPM. The error determination is based on the error of the four inputs to the TPM along with the error that the monitor introduces during the analog-to-digital conversion of the input signals. These documents indicate that the uncertainty of the calorimetric calculation is within ±1.62 percent. The licensee also calculated the uncertainty associated with the plant's NIS power range reactor power indication at the NIS racks for channel 1N41.

The licensee has submitted a calculation document titled "SPCRE005: NIS Power Range Indication Uncertainty for Control." Because of the similarities between various NIS power range channels, this calculation is applicable to any Unit 1 or Unit 2 NIS power range channel.

The calculation document indicates that the NIS power range indication uncertainty is within ±1.44 percent of rated thermal power (RTP).

RCS flow is monitored by performing a calorimetric RCS flow calculation at the start of each fuel cycle. The calculation determines the uncertainty of the calculated calorimetric RCS flow at PINGP. The licensee has submitted a calculation document titled “SPCRE006: PINGP Calorimetric RCS Flow Uncertainty.” The calculation shows that the uncertainty associated with calculation of the calorimetric RCS flow is within ±2.5 percent nominal RCS flow.

Pressurizer pressure is a controlled plant parameter. The uncertainties associated with the pressure control system are documented in a calculation report titled “SPCRE003: Pressurizer Pressure Control Uncertainty.” The calculation shows that the uncertainty associated with the ability of the plant’s automatic control systems to maintain desired pressurizer pressure during normal, steady state full-power plant operation conditions is within ±37.2 psig.

RCS T_{avg} is a plant parameter controlled through the plant’s rod control system. The uncertainties associated with the plant’s RCS T_{avg} control system is documented in a calculation report titled “SPCRE004: RCS T_{avg} Control Uncertainty.” The calculation shows that the uncertainty associated with the ability of the plant’s automatic control systems to maintain desired RCS T_{avg} during normal, steady state full-power plant operation conditions is within ±3.2 °F with additional bias uncertainty of -0.5 °F.

Based on the above calculations, the instrument uncertainties for the four operating parameters are as follows:

(1) Reactor power calorimetric uncertainty: ±1.62 percent
NIS indication uncertainty: ±1.44 percent RTP (random)
(2) Reactor coolant system flow uncertainty: ±2.5 percent nominal RCS flow
(3) Pressurizer pressure uncertainty: ±37.2 psig
(4) RCS T_{avg} uncertainty: ±3.2 °F with additional bias uncertainty of -0.5 °F

The NRC staff verified that the above determined uncertainties are within the bounds specified for RTDP uncertainties shown in the submittal dated January 14, 2004, in response to NRC staff Question No. 6.

3.1.4 WCAP 14483-A, “Generic Methodology for Expanded Core Operating Limits Report.”

The licensee referenced Westinghouse TR, WCAP-14483-A, “Generic Methodology for Expanded Core Operating Limits Report,” for licensing applications. The TR provides justification to support the TS changes required to expand current COLRs associated with the Westinghouse plants. The NRC staff found that WCAP-14483-A (Ref. 13) was previously approved for the Westinghouse plants. Therefore, the NRC staff determined that referencing WCAP-14483-A for licensing application is acceptable for the PINGP, a Westinghouse plant. The expanded COLR allows relocation of the following cycle-specific parameters to COLRs: (1) the DNB parameters of RCS average temperature (T_{avg}), RCS flow rate and pressurizer pressure; and (2) the overtemperature ΔT (OT ΔT) and overpower ΔT (OPΔT) trip setpoint parameter values. The current reactor core safety limit figure can also be relocated to the COLR and replaced with the safety DNBR limit and fuel centerline melting temperature.

3.1.5 Westinghouse Reload Evaluation Methodology

The Westinghouse reload evaluation methodology is documented in WCAP- 9272-P-A (Ref. 9). This method is based on the concept of a bounding analysis. The method assumes that validity of the reference analysis is established for the reload core under consideration, the key safety parameters for the reload core use values that are conservatively bounded by those in the reference analysis. For each reload core, the key safety parameter values are examined to determine whether a transient analysis is required or not. If all key safety parameters are conservatively bounded, the reference safety analysis remains valid for the reload core. If a reload parameter is not bounded, further analysis or evaluation is required for the reload core. For each design-basis event, the method identifies the cycle-dependent safety parameters and their limiting directions. The NRC staff found that the methodology was previously approved (Ref. 9) by the NRC for Westinghouse plants in performing reload analysis. Since the reload evaluation methodology is sensitive to the set of computer codes and methods of analysis being applied, the NRC staff's SE approving the methodology requires that any significant change in methods or codes by Westinghouse must be evaluated for its impact on the reload SE methodology of WCAP-9272. In addressing the SE restriction, the licensee proposed to use the NRC-approved Westinghouse methods and codes to perform its reload analysis. The NRC staff's evaluation (discussed in Section 3.1.6 below) determines that the Westinghouse methods and codes are acceptable for the licensee to use for licensing applications. Therefore, the NRC staff concludes that use of the Westinghouse reload methodology satisfies the SE restriction and is acceptable.

3.1.6 Computer Codes and Models Used for non-LOCA Safety Analyses

As indicated in Reference 6, the licensee proposed to perform reload analyses with the following computer codes and models:

<u>VIPRE-01</u>: This code is used to perform core thermal-hydraulic analyses, determining coolant density, mass velocity, enthalpy, vapor void, static pressure and the DNBR distribution along parallel flow channels within the reactor core under normal operational and transient conditions. As documented in WCAP-14565-P-A (Ref. 22), the code has been approved by the NRC for use to support operation of Westinghouse plants. Since the PINGP has a Westinghouse-designed core and its core analysis (discussed in Section 3.2 below) shows that the core thermal-hydraulic conditions during transients are within the NRC-approved range of the code, the NRC staff concludes that the application of VIPRE-01 for the PINGP core thermal-hydraulic calculations is acceptable.

<u>WRB-1 and W-3 Critical Heat Flux (CHF) Correlations</u>: The safety DNBR limits have been imposed to assure that there is at least a 95 percent probability at a 95 percent confidence level that the hot rods in the core do not experience a DNB during a transient. For the 14x14 optimized fuel assembly (OFA) fuel in the PINGP reactor core, the licensee used the VIPRE-01 code and the WRB-1 correlation with a safety limit of 1.17 for the DNBR analysis. When any of conditions were outside the range of the WRB-1 correlation, the licensee used the W-3 correlation. Specifically, the steamline break analysis used the W-3 correlation with the safety DNBR limit of 1.45 for the RCS pressures within the range of 500 to 1000 psia. Also, the rod withdrawal event from subcritical conditions was analyzed using the W-3 correlation for locations below the bottom non-mixing vane grid. The safety DNBR limit for the W-3 correlation is 1.3 for the RCS pressures above 1000 psia. Since the safety limits for both WRB-1 and W-3

CHF correlations with use of VIPRE-01 were previously approved (Ref. 22) by the NRC for Westinghouse plants, and the PINGP specific application of the CHF correlations (discussed in Section 3.2 below) are within the applicable ranges of the approved correlations, the NRC staff concludes that the use of the correlations with the associated safety DNBR limits is acceptable.

RETRAN-02: This code simulates a multi-loop system using a model containing a reactor vessel, hot and cold-leg piping, SGs and pressurizer. The code also includes point kinetics and reactivity effects of the moderator, fuel, boron, and control rods. The secondary side of the SG uses a homogeneous, saturated mixture for analyses of thermal transients and water level responses for indication and control. As documented in WCAP-14882-P-A, the code has been generically approved (Ref. 23) by the NRC for Westinghouse to analyze system responses to non-LOCA transients for Westinghouse pressurize-water reactors. The licensee proposed (Ref. 6) to use RETRAN-02 to perform analyses of the following events: (1) uncontrolled rod cluster control assembly (RCCA) withdrawal at power; (2) excessive heat removal due to feedwater system malfunctions; (3) loss of reactor coolant flow; (4) locked pump rotor; (5) loss of external electrical load; (6) loss of normal feedwater; (7) loss of all AC power to the station auxiliaries; and (8) rupture of a steam pipe. The NRC staff found that: (1) each event proposed to be analyzed for the PINGP using RETRAN-02 is listed in Table 1 of the NRC staff's SE approving RETRAN-02; (2) the code is reliable in calculating the system responses during transients; and (3) the results of the transient analysis (discussed in Section 3.2 below) meet the acceptance criteria for the analysis of record. The NRC staff, therefore, concludes that the licensee's use of RETRAN-02 in performing non-LOCA analysis for the PINGP is acceptable.

LOFTRAN: The LOFTRAN code provides a simulation of the RCS response and calculates system parameters such as core power, RCS flow, RCA primary and secondary pressures and temperatures. The code was previously approved (Ref. 15) for Westinghouse plants for use in performing the design-basis non-LOCA transients. The licensee proposed to use LOFTRAN to analyze the rod drop event. The NRC staff found that the event proposed to be analyzed using LOFTRAN is listed in the NRC's SE approving LOFTRAN, and the results of the analysis (discussed in Section 3.2 below) show that the acceptance criteria for the analysis of record are met. The NRC staff, therefore, concludes that the licensee's use of LOFTRAN in performing the rod drop event analysis for the PINGP is acceptable.

TWINKLE: This multi-dimensional spatial neutronics code uses an implicit finite-difference method to solve the two group transient neutronics equations in one, two, and three dimensions. This code is documented in WCAP-7979-P-A (Ref. 17). The licensee proposed to apply TWINKLE to the PINGP for analysis of the uncontrolled RCCA withdrawal from a subcritical condition and the RCCA ejection event. The NRC staff found that the TWINKLE code was previously approved by NRC for Westinghouse plants in calculating the kinetics response of a reactor for transients such as the uncontrolled RCCA withdrawal from a subcritical condition and the RCCA ejection event. Since the TWINKLE code is a generic neutron kinetics code approved by the NRC, and the licensee applied TWINKLE using the nuclear characteristics specific to the PINGP fuel for two events (Ref. 5), the application of the TWINKLE code for the proposed use is acceptable.

FACTRAN: As documented in WCAP-7908-P-A (Ref. 16), the NRC has approved the code for Westinghouse plants in calculating the transient heat flux at the surface of a rod. Since FACTRAN is an NRC-approved code for calculating thermal transients in a fuel rod and the licensee complied (Ref. 6) with the SE restrictions (related to initial fuel temperatures, the gap heat transfer coefficient and number of concentric rings used for a fuel rod, for example) imposed on use of FACTRAN, the licensee's application of FACTRAN is acceptable.

PHONIX-P and ANC: Both codes address three-dimensional features of the nuclear characteristics of the fuel. PHONIX-P is used to generate the cycle-specific nuclear cross sections. ANC with the input from PHONIX-P is used to calculate the nuclear characteristics such as power distributions, control rod worth, and reactivity feedback coefficients. As documented in WCAP-11596-P-A (Ref. 20) and WCAP-10965-P-A (Ref. 18), the NRC has approved both PHONIX-P and ANC for Westinghouse plants in calculating the nuclear characteristics. Since the codes were approved by the NRC and the licensee applied the codes using fuel design characteristics specific to the PINGP, the licensee's application of the codes is acceptable.

Method for the Rod Ejection Analysis: As documented in WCAP-7588 Rev. 1-A (Ref. 14), the NRC has approved the method which relies on spatial kinetics models for the Westinghouse plants to perform the rod ejection analysis. Since the methods were approved by the NRC and the licensee applied the method using plant conditions specific to the PINGP, the licensee's application of the method is acceptable.

The rod ejection analysis method limits the calculated radial average fuel enthalpy to below 200 cal/gm, which is less than the acceptance criterion of 280 cal/gm specified in RG 1.77. Experiment data from an European country indicated that failure of high burnup fuels at lower values of enthalpy than the limits specified in RG 1.77. However, generic analyses performed by the industry that assumed low enthalpy for fuel failure showed that the radiological consequences of the rod ejection event meet the acceptance criteria of Standard Review Plan (SRP) 15.4.8 (Appendix A). The generic analyses are predicated on conservative treatment of experimental fuel data applied to existing and planned cores within approved burnup limits for pressurized-water reactors (PWRs). In addition, there is a broad agreement among the NRC, the industry, and the international community, that burnup degradation in the margin for low-enthalpy fuel failure is likely to be regained by application of more detailed 3-dimensional analysis methods of the fuel response to rod ejection events. Therefore, the NRC staff concludes that although the RG 1.77 fuel failure enthalpy limit may not be conservative, the generic analyses provide reasonable assurance that radiological consequences of the rod ejection event will not violate the acceptance criterion in SRP Section 15.4.8 for the PWR cores operating within the current NRC-approved burnup limits. The NRC staff will not approve further extension of burnup limits until additional experimental information on fuel behavior is available to demonstrate that the fuel cladding will satisfy the regulatory acceptance criteria used in the rod ejection analysis for licensing applications.

Models in two Westinghouse TRs: WCAP-11394-P-A (Ref. 19) documents the methodology for analysis of the dropped rod event, and WCAP-12910 Rev. 1-A (Ref. 21) documents the methodology used in assessing and applying the pressurizer safety valve response to the transient analysis. The NRC staff found that the NRC had previously approved both TRs for Westinghouse plants in performing plant transient analysis. Since the TRs were approved by the NRC, and the PINGP is a Westinghouse-designed plant, the licensee's proposed use of the TRs is consistent with current licensing practice and is acceptable.

In response to the NRC staff's request, the licensee evaluated its compliance with the conditions specified in the SEs approving Westinghouse TRs (discussed in Section 3.1 above) that were referenced in the licensee's non-LOCA analysis (Refs. 1 through 6), and determined that the SEs conditions imposed on use of the methodologies have been met (Ref. 6). Accordingly, the NRC staff concludes that the licensee adequately addressed the staff's concern related to conformance to the SEs conditions.

Since the current Westinghouse non-LOCA transient analysis methodologies had not been previously applied to the PINGP, the NRC staff requested the licensee to perform an analysis for the USFAR limiting events with the Westinghouse methods (discussed in Section 3.1 above), and demonstrate that the use of the Westinghouse methods is acceptable. In response, the licensee performed (Ref. 5) the requested analysis. The transient analysis (discussed in Section 3.2 below) shows that the results are reliable and meet the applicable acceptance criteria for the analysis of record. Therefore, the NRC staff concludes that the application of the generically-approved Westinghouse methodologies (discussed in Section 3.1 above) to the PINGP for non-LOCA transient analysis is acceptable.

3.2 Non-LOCA Transient Analyses

Events Analyzed

As part of its review, the NRC staff requested the licensee to provide additional information to justify the acceptability of the generically-approved Westinghouse methods for the PINGP application. Specifically, the NRC staff requested the licensee to perform analyses for the UFSAR limiting events with the Westinghouse methods, and to demonstrate that the use of the methods is within the approved ranges and the results of the analysis meet the applicable acceptance criteria. In response, the licensee performed the requested transient analysis for UFSAR Chapter 14 events and presented the results in Reference 5 for the NRC staff to review and approve. The events analyzed are:

- Uncontrolled RCCA Withdrawal from a Subcritical Condition
- Uncontrolled RCCA Withdrawal at Power
- Rod Cluster Control Assembly Misalignment
- Chemical and Volume Control System Malfunction (Boron Dilution)
- Excessive Heat Removal due to Feedwater System Malfunctions
- Excessive Load Increase Incident
- Loss of Flow Event and Locked-Rotor Accident
- Loss of External Electrical Load
- Loss of Normal Feedwater
- Loss of AC to the Station Auxiliaries
- Rupture of a Steam Pipe Core Response
- RCCA Ejection

Initial Plant Conditions

Key safety parameters considered in the non-LOCA transient analysis are as follows:

- A nominal full-power (with RCP heat) of 1657 MWt.
- A nominal full-power reactor coolant vessel average temperature of 560 °F.
- A RCS thermal design flow (TDF) of 178,000 gpm.

- A maximum SG Tube plugging of 25 percent for Westinghouse original SGs (OSGs), and 10 percent for Framatome ANC replacement SGs (SGS) with a maximum loop-to-loop tube plugging asymmetry of 10 percent.
- A nominal pressurizer pressure of 2,250 psia.
- A radial peaking factor of 1.70 for analyses using the revised thermal design procedure (RTDP) and of 1.77 for non-RTDP analyses.
- A total peaking factor of 2.50.

For most transients that were analyzed for DNB concerns, the RTDP methodology (Ref. 12) was utilized. With the RTDP method, nominal values were assumed for the initial conditions of power, temperature, pressure, and flow. The corresponding uncertainties were statistically accounted in determining the DNBR design safety limit. The RTDP transient analyses assumed the minimum measurement flow of 182,500 gpm.

For transient analyses that were not DNB-limited, or for which RTDP was not used, the initial conditions were obtained by applying the maximum uncertainties to the nominal values in the most conservative direction. In these analyses, the RCS flow was assumed to be equal to the TDF, and the following uncertainties were used:

- The power level uncertainty of ±2 percent
- The RCS temperature measurement uncertainty of ±4 °F
- The RCS pressure measurement uncertainty of ±40 psi

Consistent with the Westinghouse reload evaluation methodology described in Reference 9, the values of kinetics parameters (control rod worth, reactivity feedback coefficient, shutdown margin, for example) used in the analysis were selected to conservatively bound those expected in subsequent operating cycles. A ±3 percent setpoint tolerance was included in the modeling of the main steam safety valves (MSSVs) and pressurizer safety valves (PSVs). The valves setpoints were assumed in the most conservative direction for each case. For example, for the loss of external load event, valves setpoints with inclusion of positive tolerances increase effective MSSV and PSV opening pressures, which would result in a higher increase in the RCS pressure. Therefore, for the overpressurization consideration, the analysis assumed that the valves opened at the setpoints with inclusion of positive tolerances. Use of valves setpoints with inclusion of negative tolerances lowers effective valves opening pressures, which would cause an earlier opening of the MSSVs and PSVs, and a lower increase in the RCS pressure. A lower RCS pressure could result in a lower DNBR. Therefore, for the DNBR calculations, the analysis considered negative tolerances for opening of the MSSVs and PSVs.

<u>SG Water Level Measurement Errors</u>

The licensee indicated in Reference 5 that SG water level instrumentation was used to determine the initial SG water inventory assumed in the transient analysis. The low-low SG water level trip was credited in the analysis of the loss of normal feedwater (LONF) and loss of all AC power (LOAC) events to trip the reactor and to actuate auxiliary feedwater (AFW). Also, the low SG wide range level signal was credited in the anticipated transient without scram (ATWS) analysis to trip the reactor and the turbine, and actuate the AFW. The NRC staff found that insufficient information was provided by the licensee regarding the consideration of SG water level measurement errors in the safety analysis to justify this credit and requested additional information as discussed below.

For the Westinghouse-manufactured SGs, Westinghouse issued a number of Nuclear Service Advisory Letters (NSALs) addressing the SG water level measurement errors. NSAL-02-3 and its revision deal with the uncertainty in the SG water level measurement caused by the placement of the mid-deck plate between the upper and lower pressure taps. This results in a delayed actuation of the SG water level low-low trip signal that trips the reactor and actuates the auxiliary feedwater. NSAL-02-4 deals with uncertainties in the measurement created because the void content of the two-phase mixture above the mid-deck plate is not reflected in the calculation. This results in a premature actuation of the SG water level high-high trip signal that isolates the feedwater system. NSAL-02-5 deals with potential inaccuracies in the initial conditions assumed in safety analyses affected by SG water level. The safety analyses may not be bounding because the velocity head under some conditions may increase the uncertainties in the SG water level control system. NSAL-03-09 indicates that Westinghouse has developed a program for the Westinghouse Owners Group that evaluates the effects on the SG water level control system uncertainties from various items including the mid-deck plate, feedwater ring and feedwater ring supports, lower-deck plate supports, non-recoverable losses due to carryunder, decrease in subcooling due to carryunder, and transient conditions due to events such as the single-loop LONF. Under the program, Westinghouse evaluated the design features of Westinghouse-designed SGs and other phenomena associated with Westinghouse SGs as they affect uncertainties with respect to the SG water level control system, and the SG water level low, low-low and high-high reactor trip functions. Westinghouse has documented the results of its program in WCAP-16115, “Steam Generator Level Uncertainties Program.” Westinghouse recommends that all licensees with plants using Westinghouse SGs review the WCAP-16115 results to determine the impact on plant-specific SG level uncertainty, in addition to the effects identified in NSALs 02-3, -4 and -5.

During the review, the NRC staff requested the licensee to discuss: (1) how the specific-plant accounts for the applicable uncertainties documented in the Westinghouse NSALs and the guidance specified in WCAP-16115 in determining the initial SG water level and SG water level setpoints; and (2) the effects of SG water level uncertainties on the analyses of the non-LOCA transients and ATWS.

In response, the licensee indicated (Ref. 24) that its setpoint calculations consider instrumentation uncertainties in accordance with the NRC-approved PINGP engineering manual Section 3.3.4.1, “Engineering Design Standard for Instrument Setpoint/Uncertainty Calculations.” The licensee indicated that it has an operating experience program which requires its engineering staff to address vendor information such as those provided by Westinghouse in NSALs. Specifically, NSAL-02-3, 02-4 and 02-5 were fully evaluated and the effects were accounted for in PINGP’s SG water level setpoint calculations. The licensee indicated that it also evaluated NSAL-03-9 and associated WCAP-16115-P issues. Based on its evaluation, the licensee stated that the information in WCAP-16115-P had no significant impact on PINGP SG level setpoint calculations, and that no plant setpoints or TS Allowable Values would be affected.

Instrumentation uncertainties of +11 percent narrow range span (NRS) to -15 percent NRS, which bound the uncertainties calculated by the licensee for the PINGP, were included in modeling of the initial SG water level. The SG water level was assumed in the most conservative direction for each case. For example, in the LONF analysis, it is conservative to assume a high initial SG water level because it delays the time of reactor trip caused by a low-low SG level. The analysis assumed an initial SG level of 55 percent NRS (the programed full-power values of 44 percent NRS plus 11 percent NRS uncertainty). In relation to the low-low and high-high SG level setpoints, the bottom and top of the NRS (0 percent NRS and 100 percent NRS) were assumed, respectively, in the transient analyses. Applicable uncertainties which are consistent with the assumptions used in the transient analyses are accounted for in the establishment of the TS Allowable Values for the low-low SG level setpoint. As for the ATWS analyses, the low SG level setpoint was assumed to be 35 percent wide range level (WRL), which included allowance for instrumentation uncertainties. The actual plant setpoint is 42.5 percent WRL. Since the licensee used the NRC-approved methods for setpoint calculations and the effects of instrumentation uncertainties discussed in the Westinghouse NSALs and the associated WCAP-16115 were accounted for in the transient analyses, the NRC staff concludes that the issue regarding the SG level measurement errors is resolved.

<u>RCP Coastdown Delay Times</u>

The licensee indicated in Reference 5 that in the analysis of the LOAC power event, the RCPs were assumed to lose power and begin coasting down 2 seconds following the turbine trip, resulting from the reactor trip. For the steam line break (SLB) analysis, the RCPs were assumed to begin coasting down 3 seconds after SLB initiation for the case without offsite power. The NRC staff requested the licensee to justify the validity of the use of different RCP coastdown delay times in the analysis for the LOAC and SLB events. In response, the licensee indicated (Ref. 24) that for both cases, it was assumed that loss of offsite power (LOOP) occurred as a consequence of instability on the power grid caused by the unit trip. The analysis assumed that for the cases with LOOP, the loss of power to RCPs caused RCPs coastdown which occurred 2 or 3 seconds following the turbine trip. The licensee stated that the RCP coastdown delay time is not an important parameter for the analysis of both LOAC and SLB events. Based on its evaluation, the licensee indicated that transient results, including DNBR, pressurizer pressure and the peak pressurizer water volume, will be negligibly affected if a zero delay time were assumed in the analysis. Also, the LOAC result in terms of the maximum pressurizer water volume was significantly less limiting than that for the LONF event in which continuous operation of the RCPs was assumed. For the SLB analysis, the result in terms of minimum DNBR for the case with LOOP was less limiting than the case with offsite power available. Therefore, the licensee stated and the staff agreed that the different RCPs coastdown delay times assumed in the analysis of LOAC and SLB events are acceptable.

The staff reviewed the non-LOCA analysis and summarized the computer codes used for each case and the results of the analysis for the most limiting cases in Table 1. The staff found that: (1) the licensee used the approved codes and methodologies to perform transient analyses; (2) the values used for the input parameters are conservative in predicting the worst consequences; (3) the computer codes are reliable in calculating core power, pressure, temperature, RCS flow and pressurizer water volume during the transient; and (4) the results of the analysis show that the acceptance criteria, satisfying the GDC 10 and 15 requirements regarding the fuel integrity and RCS pressure boundary integrity, for the analysis of record are met. Therefore, the NRC staff concludes that the analyses are acceptable.

3.2.1 ATWS Analysis

An ATWS event is defined as an anticipated operational occurrence (AOO) combined with an assumed failure of the reactor tip system to shutdown the reactor. For Westinghouse plants, the ATWS rule, 10 CFR 50.62, requires the installation of an ATWS mitigating system actuation circuitry (AMSAC) system to initiate a turbine trip and actuate AFW flow independent of the reactor protection system. The licensee has met the ATWS rule by installing an AMSAC at the PINGP. In addressing concerns regarding the operability of the AFW pump during an ATWS, the licensee subsequently installed an NRC-approved diverse scram system (DSS) at the PINGP. The DSS provides a reactor trip signal on a low SG wide range level signal and on RCP breaker position signal. In support of the request for NRC approval of the Westinghouse non-LOCA methods conversion, the licensee reanalyzed ATWS events to confirm that the AMSAC/DSS will provide adequate protection. The licensee presented its ATWS reanalysis in Reference 5. The licensee evaluated the ATWS analysis presented in the UFSAR and considered the need to reanalyze each AOO under ATWS conditions. As a result, the licensee identified five events that are not needed to explicitly reanalyze for ATWS conditions, because the events result in consequences bounded by other events. The events are: (1) uncontrolled RCCA withdrawal from a subcritical condition; (2) RCCA misalignment; (3) excessive heat removal due to feedwater system malfunctions, (4) excessive load incident; and (5) isolation of main condenser. The licensee performed ATWS reanalysis with the Westinghouse non-LOCA methods for the remaining USAR AOOs. The reanalyzed events include: (1) partial loss of reactor coolant flow; (2) LONF; (3) loss of AC to the station auxiliaries; (4) loss of external electrical load; (5) uncontrolled RCCA withdrawal at power; and (6) uncontrolled boron dilution. The licensee analyzed the ATWS events using the NRC-approved RETRAN-02 code.

Consistent with the NRC staff's position that was applied to the acceptable ATWS analyses, the licensee assumed nominal plant conditions in the reanalysis as initial plant boundary conditions. For example, the reactor power was assumed to be at 100 percent of the rated power, the initial pressure and temperature were assumed at their nominal values, the MSSVs and pressurizer power-operated relief valves were assumed to operate normally.

In the analysis, the licensee credited the DSS in conjunction with the AMSAC for event mitigation. The licensee assumed that upon actuation of the AMSAC/DSS signal, the turbine was tripped, the AFW pumps were started, and the DSS was actuated to insert the control rods. The following assumptions regarding the AMSAC/DSS were made for the analysis:

- The AMSAC/DSS SG WRL trip safety analysis setpoint is 35 percent WRL (as compared to 42.5 percent WRL for the normal trip setpoint).

- The AMSAC/DSS output signal is generated within 5 seconds.

- The AMSAC /DSS turbine trip delay is 10 seconds from the AMSAC/DSS setpoint is reached.

- The AMSAC/DSS AFW start delay time is 65 seconds from the time that AMSAC/DSS setpoint is reached.

The NRC staff found that the credit of the AMSAC/DSS for the control rod insertion, the turbine trip and the AFW pump actuation was assumed in the analysis of record. Therefore, the NRC staff concludes that the credit of the AMSAC/DSS in the reanalysis is acceptable.

In accordance with the NRC staff's review position on the ATWS analysis for the existing PWRs, such as the PINGP, the staff considers that an acceptable ATWS analysis shows that the unfavorable exposure time (UET), given the cycle design (including the moderator temperature coefficient (MTC), is not greater than 5 percent, or the ATWS pressure limit is met for at least 95 percent of the fuel cycle. The UET is the time during the cycle when reactivity feedback is insufficient to maintain pressure under 3200 psia for a given reactor site. During the review, the NRC staff requested the licensee to confirm that the selection of the values for MTC used in the ATWS reanalysis complied with the stated acceptance criterion (i.e., 5 percent UET). In response, the licensee indicated in Reference 24 that the most negative value of MTC used in the analysis was -4.2 pcm/ °F, which bounded 95 percent of those for a representative cycle. The NRC staff found that this 95 percent probability level for the captured cycle time is equivalent to the probability level to assure that UET will not be greater than 5 percent, and therefore, determined that it is acceptable.

The results of reanalysis showed that for six analyzed ATWS events, the event initiated from the loss of external electrical load is the limiting case with the maximum calculated RCS pressure of 2445 psia which is below the acceptable limit of 3200 psia. The NRC staff found that: (1) the ATWS analysis was performed with the NRC-approved RETRAN-02 code; (2) the values of MTC used in the analysis are more than 95 percent of those for a representative cycle; and (3) the calculated peak RCS pressure is within the acceptable limit. Therefore, the NRC staff concludes that the ATWS analysis is acceptable.

3.3 TS Changes

In the process of converting to Westinghouse non-LOCA methodologies, the licensee proposed the following TS changes:

3.3.1 TS 2.1.1 - Reactor Core SLs

Current TS 2.1.1, “Reactor Core SLs,” requires that in Modes 1 and 2, the combination of thermal power, RCS highest loop average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1, “Reactor Core Safety Limits vs. Boundary of Protection.” The proposed TS relocates Figure 2.1.1-1 to the COLR and replaces it with more specific requirements regarding the safety limits (i.e., the DNBR limit and the fuel centerline melt design basis). In support of COLR implementation, the licensee referenced in the proposed TS 5.6.5 the NRC-approved methodologies (Refs. 9, 10, 12, 22 and 23) used to determine the parameters in Figure 2.1.1-1.

The NRC staff concludes that the TS changes for COLR implementation are acceptable because: (1) the TS changes for COLR implementation meets the GL 88-16 (Ref. 7) guidance that allows the cycle-specific parameters to be relocated to the COLR provided that the parameters are calculated using the approved methods with these methods listed in Administrative Control Section of the TSs; and (2) the approach to relocate the reactor core SLs to the COLR is consistent with the NRC-approved method discussed in WCAP-14483-A (Ref. 13).

The proposed TS specifies the safety DNBR and fuel centerline temperature limits in the added TSs 2.1.1.1 and 2.1.1.2, respectively. The NRC staff found that: (1) the changes to the TS 2.1.1 and associated Bases are consistent with the TS 2.1.1 guidance in NUREG-1431(Ref. 8), “Standard Technical Specifications - Westinghouse Plants;” and (2) the safety limits of DNBR and fuel centerline temperature are consistent with the limits specified in the

NRC-approved WCAP-14565-P-A (Ref. 22) and WCAP-8745-P-A (Ref. 11), respectively. Therefore, the NRC staff concludes that the proposed TS 2.1.1 is acceptable.

3.3.2 TS 3.1.8 - Physics Tests Exceptions - Mode 2

As discussed in Section 3.3.5 below, the licensee set the axial flux difference (f(ΔI)) value to zero for the overpower ΔT trip function specified in Table 3.3.1-1 Function 7 in the safety analysis. The licensee proposed (Ref. 2) to delete from TS 3.1.8 reference to Table 3.3.1-1, Function 7, and to surveillance requirements (SRs) 3.3.1.3 and 3.3.1.6. The changes are shown on TS page 3.1.8.-1 and Bases pages B 3.1.8-6 and B.3.3.1-21. Bases pages B 3.1.1-2, B 3.2.2-2 and B 3.2.4-1 were changed to specify the 200 cal/gm fuel energy deposition limit during a rod ejection event. The proposed changes adequately reflect the assumptions for f(ΔI) and the fuel energy deposition limit used in the transient analysis and are, therefore, acceptable.

3.3.3 TS 3.2.3 - Axial Flux Difference (AFD)

The axial flux difference (AFD) is a measure of axial power distribution skewing to the top or bottom half of the core. The limit on AFD assures that $F_Q(Z)$ is not exceeded during normal operation and operational transients. The current AFD control is achieved by use of the CAOC methodology. The current TS refers to a target AFD band located in the COLR and allows deviation outside of the target band for certain period of time depending on the power level and accrued penalty time. The licensee proposed to use the RAOC methodology and the associated TS to control AFD for the future cycles at the PINGP. The proposed TS will replace the current TS in its entirety. The new TS refers to AFD and the operational space that are located in the COLR. If the AFD is not within its limits, it must be brought into the limits within 30 minutes, or power must be reduced to less than 50 percent of rated thermal power. The SR requires the operator to verify compliance of AFD with its limits for each operable excore channel every 7 days. The NRC staff found that the use of the RAOC (discussed in Section 2.1.2 above) is acceptable for the PINGP, and that the content and format of the proposed TS are identical with those specified in Section 3.2.3B of Westinghouse Standard TSs in NUREG-1431 (Ref. 8). Therefore, the NRC staff concludes that the proposed TS is acceptable.

3.3.4 TS 3.2.1 - Heat Flux Hot Channel Factor - $F_Q(Z)$

The limit on the values of $F_Q(Z)$ is applied to the local power density. As discussed in TS Bases 3.2.1, F^W_Q (Z) is determined by multiplying the measured steady state value of $F_Q(Z)$ by a measurement uncertainty and by a factor W(Z) which accounts for plant maneuvers. F^W_Q (Z) is compared with the limit on $F_Q(Z)$. In the current TS, if F^W_Q (Z) exceeds the $F_Q(Z)$ limit, thermal power is reduced proportionately. The proposed changes to the TS and its associated Bases require reducing the AFD limits proportionately if F^W_Q (Z) exceeds the $F_Q(Z)$ limit. Specifically, Condition B Required Actions (RA) B.1, B.2, and B.3 of TS 3.2.1 were revised to require reduction of AFD limits and reduction of setpoints of the high power range neutron flux and overpower ΔT trips in accordance with the allowable power level of the AFD limit reduction. The Completion Time (CT) for RA B.4 was also changed to correlate to the allowable power level of the reduced AFD limits. The NRC staff found that the proposed TS changes are identical with those specified in Section 3.2.1B RA B.1, B.2, B.3 and CT for RA B.4 of

Westinghouse Standard TSs in NUREG-1431 (Ref. 8). Therefore, the NRC staff concludes that the proposed TS changes are acceptable.

Bases changes were also made in support of the Westinghouse methods conversion. The changes include: reduction of the energy deposition to the fuel from 280 cal/gm to 200 cal/gm; removal of the statement that K(Z) is based on the small-break LOCA and exclusion of the top and bottom 15 percent of the core from F^W_Q (Z) evaluation. The changes adequately reflect the licensee's methods used for safety analysis and are acceptable.

3.3.5 TS TABLE 3.3.1-1 - Reactor Trip System Instrumentation Related to Overpower ΔT Trip Function, and Overtemperature ΔT Values

The Overpower ΔT (OPΔT) trip function is designed to ensure that no fuel melting will occur under all possible overpower conditions and the Overtemperature ΔT (OTΔT) trip function is designed to ensure that the design limit DNBR is met. The terms representing the ΔT trip functions are provided in the TS Table 3.3.1-1 (TS pages 7 and 8). Specifically, current TS Table 3.3.1-1 (TS page 8) defines for the OPΔT trip function the f(ΔI) input, which is a function of the axial flux difference and reduces the ΔT setpoint to account for adverse power distribution effects. The licensee indicated that PINGP has a single f(ΔI) function generator that is shared by both the OPΔT and OTΔT trip functions. Since an f(ΔI) function is required for OTΔT, the licensee stated that it is necessary to physically remove this input in order to eliminate it from OPΔT. Thus, the licensee proposed to completely remove the f(ΔI) term from the OPΔT equation specified in Table 3.3.1-1 Note 2. In response to an NRC staff's request for providing additional information to justify the removal of the OPΔT f(ΔI) for the proposed TS changes, the licensee indicated (Ref. 3) that Westinghouse had verified that via plant-specific analyses for several 14x14 and 15x15 plants that an OPΔT f(ΔI) function is not required to preclude fuel centerline melting. Specifically, the licensee performed safety analyses applicable to the PINGP, and confirmed (Ref. 5) that the OPΔT trip function will not exceed the fuel centerline melting point without the need for an f(ΔI) penalty function. Therefore, the NRC staff concludes that the changes are acceptable.

Current TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," contains Overpower ΔT (OPΔT) and Overtemperature ΔT (OTΔT) parameters values. These values are included in pages 7 and 8 of TS Table 3.3.1-1. The proposed TS removes the OPΔT and OTΔT parameter values to the COLR. The NRC staff found that the relocation of the proposed parameter values had been generically approved by the NRC on the basis defined in the previously approved report, WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report." The NRC staff also found that the licensee had satisfied the limitations specified in the staff's evaluation approving WCAP-14483-A (Ref. 13) by referencing the applicable NRC-approved setpoint methodology, WCAP-8745-P-A,"Design Bases for Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Function," in TS 5.6.5.b. Therefore, the NRC staff determines that the licensee proposal to relocate the OTΔT and OPΔT parameter values to the COLR is acceptable.

3.3.6 TS TABLE 3.3.1-1 - Reactor Trip System Instrumentation Related to Item 8.a, Pressurizer Pressure-Low

In response to the NRC staff's questions regarding the need to increase the allowable value for pressurizer pressure reactor trip from the existing value of 1760 psig to 1845 psig, the licensee provided the following explanation in their letter dated January 14, 2004.

The licensee states:
"The proposed change to this value is in the conservative direction. As described and illustrated in Appendix B of WCAP-8745-P-A, the low and high pressurizer reactor trips limit the range of conditions over which the OTΔT and OPΔT trips must provide protection. The more limited this range is, the less restrictive the setpoint constants need to be, which provides increased operational flexibility while still ensuring that safety limits are met. The current Allowable Value for pressurizer pressure - low is much lower than the actual plant setting. To credit and thus derive the benefit from this higher than actual plant setting in the ΔT trip setpoints it is necessary to increase the safety analysis setpoint, which makes it necessary to formally increase the Allowable Value for pressurizer pressure - low in the Technical Specifications."

The pressurizer pressure-low trip function is designed to ensure that protection is provided against violating the DNB due to low pressure. Current TS defines the setpoint of greater than or equal to 1760 psig as Item 8.a in TS Table 3.3.1-1. The proposed TS increases the setpoint to greater than or equal to 1845 psig. The licensee further confirmed (Ref. 6) that the proposed TS value bounds the value (1835 psig) used in the PINGP safety analysis discussed Reference 5. The licensee also stated that the difference between the actual plant setting and the TS allowable value provides sufficient margin for calibration tolerances and measurable instrument uncertainties such as instrument accuracy, setting drift, and so on. The NRC staff has reviewed the calculation provided by the licensee for general adequacy and adherence to the PING setpoint methodology and agrees with the licensee's determination. Therefore, the NRC staff concludes that the change is acceptable.

3.3.7 TS 5.6.5 - COLR

TS 5.6.5.a provide a list of TSs for which the core operating limits are documented in the COLR. The proposed TS 5.6.5.a (Refs. 1 and 2) adds to list TS 2.1.1, "Reactor Core SLs," and LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation Overtemperature ΔT and Overpower ΔT Parameter Values for Table 3.3.1-1." The changes are acceptable because the added TS and LCO specify the cycle-specific parameters that are allowed to be relocated to the COLR based on the evaluation discussed in Sections 3.3.1 and 3.3.5 above.

TS 5.6.5.b provides a list of titles of TRs documenting the NRC-approved methodologies used to determine the values of cycle-specific parameters that are removed to the COLR. The proposed TS adds the following titles of TRs to the reference list (the Reference numbers refer to those in the proposed TS 5.6.5.b (Ref. 1)) :

- Reference 13 - WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control/F_Q Surveillance Technical Specifications;"
- Reference 14 - WCAP 8745-P-A, "Design Bases for the Thermal Overpower ΔT and Overtemperature ΔT Trip Functions;"

- Reference 15 - WCAP-11397-P-A, "Revised Thermal Design Procedure;" and
- Reference 16 - WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report."

In addressing the NRC staff's concern with regard to which TRs are be referenced in the TS, the licensee added to the revised TS 5.6.5.b (Ref. 3) the following TRs that are also used to calculate the values of cycle-specific parameters that are removed to the COLR (the Reference numbers refer to those in the proposed TS 5.6.5.b (Ref. 3)):

- Reference 17 - WCAP-7588 Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods."
- Reference 18 - WCAP-7908-A, "FACTRAN - A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod."
- Reference 19 - WCAP-7907-P-A, "LOFTRAN Code Description."
- Reference 20 - WCAP-7979-P-A, "TWINKLE - A Multidimensional Neutron Kinetics Computer Code."
- Reference 21 - WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code."
- Reference 22 - WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event."
- Reference 23 - WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurizer Water Reactor Cores."
- Reference 24 - WCAP-12910 Rev. 1-A, "Pressurized Safety Valve Set Pressure Shift."
- Reference 25 - WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurizer Water Reactor Non-LOCA Thermal Hydraulic Safety Analysis."
- Reference 26 - WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pessurized Water Reactor Non-LOCA Safety Analyses."

GL 88-16 (Ref. 7) allows licensees to remove cycle-dependent variables from TS provided that the values of these variables are included in a COLR and are determined with an NRC-approved methodology which is referenced in the TS. The NRC staff finds, as discussed in Sections 3.1 and 3.2 of this evaluation, that the methodologies documented in the referenced TRs are acceptable for use in support of the PINGP licensing applications, and the TS inclusion of the titles of the referenced TRs is consistent with the GL 88-16 guidance for TS inclusion of the approved methodologies. Therefore, the NRC staff concludes that the TS changes are acceptable.

The licensee also proposed (Ref. 2) to delete TR dates, references to NRC safety evaluations, volume and addendum in References 4, 5, 6, 7, 8, 9, 11 and 12 of TS 5.6.5.b. The NRC staff found that the licensee's proposed changes are consistent with the guidance of TS 5.6.5.b of the Westinghouse STS in NUREG-1431, Revision 2 (Ref. 8), which states that "...identify the Topical Report(s) by number and title or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date. The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements)." Further, the licensee indicated (Ref. 2) that it will include in the COLR the complete identification for each of the referenced TRs as required by NUREG-1431. Therefore, the NRC staff concludes that the proposed deletion of TR dates, references to NRC safety evaluations, volume and addendum is acceptable. The licensee proposed to remove References 6 from TS 5.6.5.b (Ref. 2) because

References 6 and 7 of TS 5.6.5.b are identical when their date, volume and addendum are removed. The NRC staff finds that the removal of Reference 6 is also acceptable.

4.0 SUMMARY

Based on the review, the NRC staff has determined that the proposed Westinghouse methodologies and computer codes for non-LOCA analyses discussed in Sections 3.1 and 3.2 above are acceptable for PINGP, Units 1 and 2, because they were previously approved by the NRC for Westinghouse plants. The licensee's use of the methodologies and computer codes in performing reload analysis for the PINGP complies with the NRC staff's SE limitations imposed on the use of the approved methods or codes, and the results of the non-LOCA analyses using the Westinghouse methods satisfy the requirements of USAR Sections 3.1.2.1 and 4.3.1.2.1 with respect to integrity of the fuel and RCS pressure boundary.

This LAR referenced the Engineering Manual Section 3.3.4.1, "Engineering Design Standard for Instrument Setpoint/ Uncertainty Calculations," for the instrument setpoint and uncertainty calculation, which was accepted by the NRC staff during ITS conversion. The NRC staff's review of the submitted calculations determined that the PING setpoint methodology has followed the guidance provided in RG 1.105, Revision 3, and the requirements of 10 CFR 50.36 and, therefore, is acceptable.

The NRC staff also determined that the associated proposed changes to TSs 2.1.1, 3.1.8, 3.2.1, 3.2.3, TS Table 3.3.1-1, and 5.6.5 are acceptable because they meet the current regulation (10 CFR 50.36) and are consistent with the GL 88-16 guidance and NUREG-1431 (Revision 2), "Standard Technical Specifications - Westinghouse Plants," discussed in Section 3.3.

Table 1 - The NON-LOCA Analysis Results and Computer Codes and Critical Heat Flux Correlations Used in the Analysis

Events Analyzed	Codes Used	Result parameters	Results for limiting case
Uncontrolled RCCA Withdrawal from a Subcritical Condition	TWINKLE FACTRAN VIPRE	Min. DNBR below first mixing grid (W-3 correlation, thimble/typical) Min. DNBR above first mixing grid (WRB-1 correlation, thimble/typical) Max. fuel centerline temp., °F	1.703/1.849 2.047/2.075 2,480
Uncontrolled RCCA Withdrawal at Power	RETRAN	Min. DNBR (WRB-1) Peak RCS pressure, psia Peak SG pressure, psia	1.432 2,572.7 1,185.4

Events Analyzed	Codes Used	Result parameters	Results for limiting case
Rod Cluster Control Assembly Misalignment	LOFTRAN ANC VIPRE	Min. DNBR (WRB-1) Peak linear heat generation, kW/ft Peak uniform cladding strain, %	>1.34 <22.54 <1.0
Chemical and Volume Control System Malfunction Mode 1 with manual rod control Mode 1 with auto. rod control Mode 2 Modes 3, 4 , 5 and 6	N/A	Min. time to loss of shutdown margin, minutes	 >20 >21 >17 >24
Excessive Heat Removal due to Feedwater System Malfunctions	RETRAN VIPRE	Min. DNBR (WRB-1)	1.41
Excessive Load Increase Incident	RETRAN VIPRE	Min. DNBR (WRB-1) Peak core heat flux, %	1.49 117.1
Loss of Reactor Coolant Flow - Partial Loss of Flow - Complete Loss of Flow Locked-Rotor Accident	RETRAN VIPRE	 Min. DNBR (WRB-1) Min. DNBR (WRB-1) Max. percent rods in DNB, % Peak RCS pressure, psia Peak cladding temp., °F Max. Zirc-water reaction, %	 >1.607 >1.333 18.4 2,562 2,010 0.6
Loss of External Electrical Load	RETRAN VIPRE	Min. DNBR (WRB-1) Peak RCS pressure, psia Peak SG pressure, psia	1.77 2,701 1,207.3
Loss of Normal Feedwater	RETRAN	Max. pressurizer mixture volume, ft^3	934.1
Loss of All AC Power to the Station Auxiliaries	RETRAN	Max. pressurizer mixture volume, ft^3	653.0

Events Analyzed	Codes Used	Result parameters	Results for limiting case
Rupture of a Steam Pipe-Core Response - Zero Power - Full Power	RETRAN VIPRE	Min. DNBR (WRB-1) Min. DNBR below first mixing vane grid (W-3) (thimble/typical) Min. DNBR above first mixing vane grid (WRB-1) (thimble/typical) Peak linear heat generation (kW/ft)	2.535 1.448/1.681 1.537/1.578 <22.54
RCCA Ejection	TWINKLE FACTRAN	Max. fuel pellet average enthalpy, cal/g	< 168.6

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (68 FR 22750). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Letter from J. M. Solymossy of Nuclear Management Company to US Nuclear Regulatory Commission, “License Amendment Request (LAR) dated March 25, 2003, Safety Analyses Transition,” dated March 25, 2003.

2. Letter from J. M. Solymossy of Nuclear Management Company to US Nuclear Regulatory Commission, “Supplement to License Amendment Request (LAR) dated March 25, 2003, Safety Analyses Transition,” dated June 16, 2003.

3. Letter from J. M. Solymossy of Nuclear Management Company to US Nuclear Regulatory Commission, “Supplement to License Amendment Request (LAR) dated March 25, 2003, Safety Analyses Transition,” dated January 14, 2004.

4. Attachments 2 through 4 to Reference 3, the Licensee Responses to the Staff Request for Additional information.

5. Attachment 5 to Reference 3, A Licensing Report Documenting the Non-LOCA Analysis Performed with Westinghouse Methods.

6. Attachment 6 to Reference 3, the Licensee Assessment in Addressing its Compliance with the SE(s) Restrictions on Applicability of TRs.

7. Generic Letter (GL) 88-16, “Removal of Cycle Specific Parameter Limits from Technical Specifications,” October 4, 1988.

8. NUREG-1431, Revision 2, “Standard Technical Specifications - Westinghouse Plants,” April 30, 2001.

9. WCAP-9272-P-A, “Westinghouse Reload Safety Evaluation Methodology,” July 1985.

10. WCAP-10216-P-A, Revision 1A, “Relaxation of Constant Axial Offset Control/F_Q Surveillance Technical Specification,” February 1994.

11. WCAP-8745-P-A, “Design Bases for Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions,” September 1986.

12. WCAP-11397-P-A, “Revised Thermal Design Procedure,” April 1987.

13. WCAP-14483-A, “Generic Methodology for Expanded Core Operating Limits Report,” January 1999.

14. WCAP-7588 Rev. 1-A, “An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods,” January 1975.

15. WCAP-7907-P-A, “LOFTRAN Code Description,” April 1984.

16. WCAP-7908-A, “FACTRAN - A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod,” December 1989.

17. WCAP-7979-P-A, "TWINKLE - A Multidimensional Neutron Kinetics Computer Code," January 1975.

18. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986.

19. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," January 1990.

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21. WCAP-12910 Rev. 1-A, "Pressurized Safety Valve Set Pressure Shift," May 1993.

22. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurizer Water Reactor Non-LOCA Thermal Hydraulic Safety Analysis," dated October 1999.

23. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pessurized Water Reactor Non-LOCA Safety Analyses," April 1999.

24. Letter from J. M. Solymossy of Nuclear Management Company to US Nuclear Regulatory Commission, "Supplement to License Amendment Request (LAR) dated March 25, 2003, Safety Analyses Transition (TAC Nos. MB8128 and MB8129)" February 11, 2004.

25. Letter from J. M. Solymossy of Nuclear Management Company to US Nuclear Regulatory Commission, "Supplement to License Amendment Request (LAR) dated March 25, 2003, Safety Analyses Transition, (TAC Nos. MB8128 and MB8129)" February 23, 2004

26. Letter from J. M. Solymossy of Nuclear Management Company to US Nuclear Regulatory Commission, "Revised Page for License Amendment Request (LAR) Dated March 25, 2003, Safety Analysis Transition (TAC NOs. MB8128 and MB8129)" April 7, 2004

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