Ameren
MISSOURI

Callaway Plant

October 11, 2016

ULNRC-06328

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

10 CFR 50.90

Ladies and Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
FACILITY OPERATING LICENSE NPF-30
REVISION OF TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," TO ALLOW THE USE OF THE PARAGON AND NEXUS CORE DESIGN METHODS (LDCN 16-0011)

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," Ameren Missouri (Union Electric Company) herewith transmits an application for amendment to Facility Operating License Number NPF-30 for the Callaway Plant. The proposed amendment would modify Technical Specification (TS) requirements to reference and allow use of the NRC-approved methodologies described in WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," and WCAP-10965-P-A, Addendum 2-A, "Qualification of the New Pin Power Recovery Methodology," for the Callaway Plant.

The Enclosure to this letter provides a detailed description and technical evaluation of the proposed changes, including a determination that the proposed changes involve no significant hazards. Additional information is provided in attachments to the enclosure. Specifically, attachment 1 provides the marked-up pages of the proposed TS 5.6.5.b changes, attachment 2 provides the retyped proposed TS pages, and attachment 3 provides the proposed FSAR changes, for information only.

It has been determined that this amendment application does not involve a significant hazards consideration as determined per 10 CFR 50.92, "Issuance of amendment." In addition, pursuant to 10 CFR 51.22, "Criterion categorical exclusion or otherwise not requiring environmental review," Section (b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment. It should also be noted that this submittal does not contain new commitments.

The Callaway Onsite Review Committee has reviewed and approved the proposed changes and has approved the submittal of this amendment application.

This amendment request is being submitted in support of plant operation following the conclusion of Callaway refuel outage 22, scheduled for October, 2017. Ameren Missouri therefore requests approval of the requested license amendment by October 7, 2017, to support core fuel reload activities for Cycle 23. Ameren Missouri further requests that the license amendment be made effective upon NRC issuance, to be implemented within 90 days from the date of issuance.

In accordance with 10 CFR 50.91 "Notice for public comment; State consultation," Section (b)(1), a copy of this amendment application is being provided to the designated Missouri State official.

If there are any questions, please contact Mr. Tom Elwood at 314-225-1905.

I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,

Roger C. Wink P/N 6381

Roger C. Wink
Manager, Regulatory Affairs

Executed on: October 11, 2016

Enclosure: Description and Assessment of the Proposed Change

Attachments to the Enclosure:

1. Technical Specification Page Markups
2. Retyped Technical Specification Pages
3. Proposed FSAR Changes

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Ms. Leanne Tippett-Mosby (DNR)

DESCRIPTION AND ASSESSMENT OF THE PROPOSED CHANGE

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1.0 SUMMARY DESCRIPTION

Ameren Missouri (Union Electric Company) is proposing to amend Operating License NPF-30 for Callaway Plant. The proposed amendment would modify Technical Specification (TS) requirements to reference and allow the use of NRC-approved methodologies described in the following topical reports (TRs):

- WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON" (Reference 1),
- WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology" (Reference 3) and
- WCAP-10965-P-A, Addendum 2-A, "Qualification of the New Pin Power Recovery Methodology" (Reference 5).

2.0 DETAILED DESCRIPTION

Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," Section b, lists the analytical methods used to determine the core operating limits.

The proposed change would add the following analytical methods to those listed in Section b of Specification 5.6.5:

- WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON."
- WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology."
- WCAP-10965-P-A, Addendum 2-A, "Qualification of the New Pin Power Recovery Methodology."

Due to the changes described above, the list of analytical methods in Specification 5.6.5 will be renumbered as applicable.

The proposed amendment will allow the methodology of WCAP-16045-P-A, WCAP-16045-P-A, Addendum 1-A, and WCAP-10965-P-A, Addendum 2-A to be utilized in providing future core designs, including the core design for the next refueling at Callaway Plant (scheduled for October 2017). The proposed changes are acceptable since they assure the core operating limits have been calculated in accordance with NRC approved methodologies.

The addition of the analytical methods by topical report number and title is consistent with Amendment No. 183. Amendment No. 183 implemented Technical Specification Task Force (TSTF) Traveler TSTF-363, "Revise Topical Report References in ITS 5.6.5, COLR," (References 7 and 8), wherein the NRC concluded in their safety evaluation that the proposed change to only list the NRC-approved methodology by topical report number and title is acceptable.

3.0 TECHNICAL EVALUATION

TS Section 5.6.5.b specifies that the analytical methods used to determine the core operating limits shall be previously reviewed and approved by NRC.

According to Westinghouse, the core reload design vendor for Callaway Plant, Advanced Nodal Code (ANC) is a computer code capable of two-dimensional and three-dimensional neutronics calculations. ANC is the reference model for certain safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. As such, ANC is one of the primary analytical methods utilized for establishing core operating limits per TS 5.6.5.

PHOENIX-P is the neutron transport code traditionally used to provide cross section data as input to the Advanced Nodal Code (ANC) code. The PARAGON computer code is a standalone neutron transport code based on collision probability techniques, and it is approved for use as a standalone lattice physics code and as a cross section generation tool for core simulators, such as ANC, for uranium-fuel pressurized water reactors (PWRs). ANC is a core simulator code system, which performs calculations based on nuclear data supplied by a code such as PARAGON or PHOENIX-P. The PARAGON nuclear data methodology was developed as a direct replacement to PHOENIX-P.

The NRC Safety Evaluation (Reference 2) for the PARAGON nuclear data methodology states, in part, "... the staff considers the new PARAGON code to be well qualified as a stand-alone code replacement for the PHOENIX-P lattice code, wherever the PHOENIX-P code is used in NRC-approved methodologies."

WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," is an improvement to the PARAGON computer code. The NEXUS methodology is a reparameterization of the PARAGON nuclear data output and a new reconstruction approach within the ANC core simulator code to simplify the use of this code system for design use. NEXUS has been implemented in the PARAGON/ANC code system for design use. Specifically, the NEXUS methodology has been implemented in the parameterization of PARAGON cross sections for input to ANC and also in ANC to reconstruct those cross sections at specific nodal conditions. Since the NEXUS methodology provides a linkage between PARAGON and ANC, establishing a new code system, while still using PARAGON, both WCAP-16045-P-A and WCAP-16045-P-A, Addendum 1-A are being added to Specification 5.6.5.b.

The NRC Safety Evaluation (Reference 4) for the NEXUS nuclear data methodology states, in part, "The NRC staff has reviewed the TR [Topical Report] submitted by Westinghouse and determined that the NEXUS/ANC code system is adequate to replace the PARAGON/ANC code system wherever the latter is used in NRC-approved methodologies."

The NEXUS/ANC9 pin power reconstruction method, documented in WCAP-10965-P-A, Addendum 2-A, "Qualification of the New Pin Power Recovery Methodology," represents a departure from conventional pin power methodologies. It recasts the conventional pin factors into a cross section and flux component. One of the main advantages of this new method is that it can track the effects of control rod insertion on pin powers.

The NRC Safety Evaluation (Reference 6) for the New Pin Power Recovery Methodology states, in part, "The NRC staff has reviewed the TR submitted by Westinghouse and determined that WCAP-10965-P, Addendum 2/WCAP-10966-A, Addendum 2, is an adequate enhancement to replace the pin power recovery methodologies of NRC-approved methodologies WCAP-10965-P-A, and where appropriate, WCAP-10965-P-A, Addendum 1."

In conclusion, the proposed changes to TS 5.6.5.b would allow the use of only NRC-approved methodologies and are in accordance with NRC expectations (consistent with TSTF-363 and License Amendment 183).

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements / Criteria

The regulatory requirements and/or guidance documents associated with this amendment application include the following:

- 10 CFR 50.36(c)(5) requires that the TS include a category called "Administrative Control," that contains the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

  The proposed NRC-approved methodologies would be included in the Administrative Controls section of the TS and would be used to determine a core operating limit. They would continue to assure that the plant is operated in a safe manner. As such, the proposed change would be consistent with the Administrative Controls requirement of 10 CFR 50.36(c)(5).

- 10 CFR 50 Appendix A, Criterion 10, "Reactor Design" requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

  The proposed analytical methodologies are an improvement that allows more accurate modeling of core performance. The NRC has reviewed and approved the additional methodologies for use in lieu of the current methodology; thus, the margin of safety is not reduced due to this change.

- The regulatory basis for TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," is to ensure core operating limits are established in accordance with NRC-approved methodologies and are documented in the COLR.

  The NRC has reviewed and approved the additional methodologies that are being proposed to be added to TS 5.6.5.b. Therefore, this requirement continues to be met since the proposed methodologies have been approved by the NRC.

- Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," provides guidance for the removal of cycle-specific parameter limits from the TS, since processing cycle-specific parameter limit changes in the TS was an unnecessary burden on both licensees and the NRC. In light of the fact core operating limits are required to be developed with NRC-approved methodologies (regardless of whether they are specified in the TS or not), and in order to support the removal of cycle-specific parameter limits from the TS, the Generic Letter recommends that cycle-specific parameter limit values be placed in a CORE OPERATING LIMITS REPORT (COLR), thereby eliminating the need for many reload license amendments. The COLR would be submitted to the NRC to allow continued trending of information even though NRC approval of these limits would not be required. This was established per License Amendment 58, and then expanded per License Amendment 183.

  The proposed methodologies have been approved by the NRC, and will be placed in the COLR, in conformance with Generic Letter 88-16. Therefore, this requirement continues to be met.

There are no changes being proposed in this amendment application such that conformance or commitments to the regulatory requirements and/or guidance documents above would come into question. The evaluations documented herein confirm that Callaway Plant will continue to comply with all applicable regulatory requirements.

In conclusion, based on considerations discussed herein, (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) issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.2 Precedent

On August 7, 2014, the NRC issued Amendment No. 209 to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment revises TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," to replace the methodology of TR WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," with WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," and WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," to determine core operating limits.

On July 17, 2013, the NRC issued Amendment No. 191 to Renewed Facility Operating License No. NPF-2 and Amendment No. 187 to Renewed Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2, respectively. This amendment adds a reference to WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," and WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," to TS Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)."

On March 9, 2016, the NRC issued Amendment No. 224 to Facility Operating License No. DPR-80 and Amendment No. 226 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant (DCPP), Unit Nos. 1 and 2, respectively. The amendments revise TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," Section b, to replace Westinghouse proprietary TR WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," with NRG-approved proprietary TR WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," and NRG-approved proprietary TR WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology."

4.3 No Significant Hazards Consideration Determination

Ameren Missouri has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

Response: No

The proposed license amendment would revise TS 5.6.5.b to add additional TR references for NRC-approved methodologies used in core reload designs and the determination of core operating limits, thereby specifically approving the use of these methodologies for the Callaway Plant. The additional analytical methodologies are improvements over the current methodologies in use at Callaway Plant. The NRC staff reviewed and approved these methodologies and concluded that these analytical methods are acceptable as a replacement for the current analytical method.

This proposed license amendment does not involve any physical.changes to the Callaway Plant. Additionally, the core operating limits determined using the proposed analytical methods will continue to assure that the reactor operates safely. On that basis, the proposed changes do not involve an increase in the probability of an accident.

The proposed changes will not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended functions to mitigate the consequences of an initiating event within the assumed acceptance limits and therefore, does not increase the likelihood of any failure mechanisms or precursors to transients or accidents postulated and analyzed in the Callaway Plant FSAR. Operation of the reactor with core operating limits determined by use of the proposed analytical methods does not increase the reactor power level, does not increase the core fission product inventory, and does not change any radiological release assumptions. The proposed changes will not alter any accident analysis assumptions discussed in the FSAR, nor do they involve any changes to the requirement for Callaway Plant to operate within the power distribution limits and shutdown margins required by the TS and within the assumptions of the safety analyses described in the FSAR. Therefore the proposed methodology and TS changes do not involve a significant increase in the consequences of an accident.

Therefore, it is concluded that this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The proposed change provides revised analytical methods for determining core operating limits, and does not change any system functions or requirements. Acceptance criteria required to be met for analyzed core performance under normal, transient and accident conditions are not being changed, as the core operating limits will continue to be established in accordance with NRC-approved methods. The change does not involve physical alteration of the plant, as no new or different type of equipment will be installed. The change does not alter assumptions made in the safety analyses, but ensures that the core will operate within safe limits. Consequently, this change does not create new failure modes or mechanisms, and no new accident precursors are generated.

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

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

Response: No

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed changes do not physically alter safety-related systems, nor do they affect the way in which safety related systems perform their functions. The setpoints at which protective actions are initiated are not altered by the proposed changes. The availability of equipment required to be available to actuate upon demand for mitigating an analyzed event is unchanged by the proposed amendment. The proposed analytical methodologies are an improvement that allows more accurate modeling of core performance. The NRC has reviewed and approved the additional methodologies for use in lieu of the current methodology; thus, the margin of safety is not reduced due to this change.

Therefore, it is concluded that the proposed change does not involve a significant reduction in a margin of safety.

In consideration of all of the above, Ameren Missouri concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and on that basis, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

Based on the considerations discussed above, 1) there is a 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.

5.0 ENVIRONMENTAL EVALUATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

6.0 REFERENCES

1. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004 (ADAMS Accession No. ML042250322).
2. Final Safety Evaluation for Westinghouse Topical Report WCAP-16045-P, Revision 0, "Qualification of the Two-Dimensional Transport Code PARAGON (TAC NO. MB8040)," March 18, 2004 (ADAMS Accession No. ML040780402).
3. WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology,". August 2007 (ADAMS Accession No. ML053460157).
4. Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP-16045-P-A, Addendum 1, "Qualification of the NEXUS Nuclear Data Methodology" (TAC NO. MC9606)," February 23, 2007 (ADAMS Accession No. ML070320398).
5. WCAP- 10965-P-A, Addendum 2/ WCAP- 10966-A, Addendum 2, "Qualification of the New Pin Power Recovery Methodology" (ADAMS Accession No. ML091560106).
6. Final Safety Evaluation for Westinghouse Electric Company Topical Report WCAP-10965-P-A, Addendum 2/WCAP-10966-Addendum 2, "Qualification of the New Pin Power Recovery Methodology," (TAC NO. ME1420) (ADAMS Accession No. ML102350044).
7. ULNRC-05322, " Relocation of Technical Specification Cycle-Specific Parameters to the Core Operating Limits Report, Adoption of Industry Travelers TSTF-339 And TSTF-363" (ADAMS Accession No. ML062360244).

8. NRC letter from J. R. Jolicoeur to TSTF, "Implementation of Travelers TSTF-363, Revision 0, 'Revise Topical Report References ITS 5.6.5, COLR,' TSTF-408, Revision 1, 'Relocation of LTOP Enable Temperature and PORV Lift Setting to the PTLR,' and TSTF-419, Revision 0, 'Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR,'" August 4, 2011 (ADAMS Accession No. ML110660285).

ATTACHMENT 1

TECHNICAL SPECIFICATION PAGE MARKUPS

No changes. Page included for completeness.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Moderator Temperature Coefficient limits in Specification 3.1.3,
2. Shutdown Bank Insertion Limit for Specification 3.1.5,
3. Control Bank Insertion Limits for Specification 3.1.6,
4. Axial Flux Difference Limits for Specification 3.2.3,
5. Heat Flux Hot Channel Factor, $F_Q(Z)$, F_Q^{RTP}, K(Z), W(Z) and F_Q Penalty Factors for Specification 3.2.1,
6. Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}^{N}$, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$, limits for Specification 3.2.2,
7. Shutdown Margin Limits for Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6, and 3.1.8,
8. Reactor Core Safety Limits Figure for Specification 2.1.1,
9. Overtemperature ΔT and Overpower ΔT Setpoint Parameters for Specification 3.3.1, and
10. Reactor Coolant System Pressure and Temperature DNB Limits for Specification 3.4.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY."
2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL AND FQ SURVEILLANCE TECHNICAL SPECIFICATION."
3. WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE."

(continued)

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT."

5. WCAP-11397-P-A, "REVISED THERMAL DESIGN PROCEDURE."

6. WCAP-14565-P-A, "VIPRE-01 MODELING AND QUALIFICATION FOR PRESSURIZED WATER REACTOR NON-LOCA THERMAL-HYDRAULIC SAFETY ANALYSIS."

7. WCAP-10851-P-A, "IMPROVED FUEL PERFORMANCE MODELS FOR WESTINGHOUSE FUEL ROD DESIGN AND SAFETY EVALUATIONS."

8. WCAP-15063-P-A, "WESTINGHOUSE IMPROVED PERFORMANCE ANALYSIS AND DESIGN MODEL (PAD 4.0)."

9. WCAP-8745-P-A, "DESIGN BASES FOR THE THERMAL OVERPOWER DT AND THERMAL OVERTEMPERATURE DT TRIP FUNCTIONS."

10. WCAP-10965-P-A, "ANC: A WESTINGHOUSE ADVANCED NODAL COMPUTER CODE."

insert A

12 ~~11.~~ WCAP-11596-P-A, "QUALIFICATION OF THE PHOENIX-P/ANC NUCLEAR DESIGN SYSTEM FOR PRESSURIZED WATER REACTOR CORES."

13 ~~12.~~ WCAP-13524-P-A, "APOLLO: A ONE DIMENSIONAL NEUTRON DIFFUSION THEORY PROGRAM."

14 ~~13.~~ WCAP-14565-P-A Addendum 2-P-A, "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications."

insert B

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

Tech Spec 5.6.5.b Markup

INSERT A

11. WCAP-10965-P-A Addendum 2-A, "Qualification of the New Pin Power Recovery Methodology."

INSERT B

15. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON."
16. WCAP-16045-P-A Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology."

ATTACHMENT 2

RETYPED TECHNICAL SPECIFICATION PAGES

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Not Used.

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period.

The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in a format similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Not used.

(continued)

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4. Axial Flux Difference Limits for Specification 3.2.3,

5. Heat Flux Hot Channel Factor, $F_Q(Z)$, F_Q^{RTP}, K(Z), W(Z) and F_Q Penalty Factors for Specification 3.2.1,

6. Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}^{N}$, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$, limits for Specification 3.2.2,

7. Shutdown Margin Limits for Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6, and 3.1.8,

8. Reactor Core Safety Limits Figure for Specification 2.1.1,

9. Overtemperature ΔT and Overpower ΔT Setpoint Parameters for Specification 3.3.1, and

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b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

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2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL AND FQ SURVEILLANCE TECHNICAL SPECIFICATION."

3. WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE."

(continued)

5.6 Reporting Requirements

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT."

5. WCAP-11397-P-A, "REVISED THERMAL DESIGN PROCEDURE."

6. WCAP-14565-P-A, "VIPRE-01 MODELING AND QUALIFICATION FOR PRESSURIZED WATER REACTOR NON-LOCA THERMAL-HYDRAULIC SAFETY ANALYSIS."

7. WCAP-10851-P-A, "IMPROVED FUEL PERFORMANCE MODELS FOR WESTINGHOUSE FUEL ROD DESIGN AND SAFETY EVALUATIONS."

8. WCAP-15063-P-A, "WESTINGHOUSE IMPROVED PERFORMANCE ANALYSIS AND DESIGN MODEL (PAD 4.0)."

9. WCAP-8745-P-A, "DESIGN BASES FOR THE THERMAL OVERPOWER DT AND THERMAL OVERTEMPERATURE DT TRIP FUNCTIONS."

10. WCAP-10965-P-A, "ANC: A WESTINGHOUSE ADVANCED NODAL COMPUTER CODE."

11. WCAP-10965-P-A Addendum 2-A, "Qualification of the New Pin Power Recovery Methodolgy."

12. WCAP-11596-P-A, "QUALIFICATION OF THE PHOENIX-P/ANC NUCLEAR DESIGN SYSTEM FOR PRESSURIZED WATER REACTOR CORES."

13. WCAP-13524-P-A, "APOLLO: A ONE DIMENSIONAL NEUTRON DIFFUSION THEORY PROGRAM."

14. WCAP-14565-P-A Addendum 2-P-A, "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications."

15. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON."

16. WCAP-16045-P-A Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology."

(continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing and PORV lift setting as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and

2. Specification 3.4.12, "Cold Overpressure Mitigation System (COMS)."

b. The analytical methods used to determine the RCS pressure and temperature and COMS PORV limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves".

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not used.

5.6.8 PAM Report

When a report is required by Condition B or F of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Not used.

(continued)

5.6 Reporting Requirements (continued)

5.6.10 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG;

b. Degradation mechanisms found;

c. Nondestructive examination techniques utilized for each degradation mechanism;

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications;

e. Number of tubes plugged during the inspection outage for each degradation mechanism;

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator; and

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

ATTACHMENT 3

PROPOSED FSAR CHANGES
(for information only)

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TABLE 1.6-2 (Sheet 20)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review(1) Status
WCAP-10043	Steam Generator Tube Plugging Analysis for the Westinghouse Standardized Nuclear Power Plant (P) System	Rev. 0	5.4.2.5	12/3/82	U
WCAP-10297-P-A	"Dropped Rod Methodology for Negative Flux Rate Trip Plants"		15.4	6/83	
WCAP-10851-P-A	Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations	Rev. 0	3A, 4.2, 4.3, 4.4	6/85	A
WCAP-10961-P	Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment	Rev. 1	3B.4.2	1/17/86	A
SCP-97-116	Releases for Equipment Environmental Qualification Outside Containment				
WCAP-12472-P-A	BEACON Core Monitoring and Operations Support System	Rev. 0	4.3.2.2.7	4/90	A
WCAP-12476(3)	Evaluation of LOCA During Mode 3 and Mode 4 Operation for Westinghouse NSSS	Rev. 1	Table 15.0-8	11/27/91	Not approved-withdrawn on 4/28/99

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Westinghouse Topical Report No.	Title	Rev. No.	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-10965-P-A	ANC: A Westinghouse Advanced Nodal Computer Code	Rev. 0	4.3 15.0.11.10	9/86	A
WCAP-10965-P-A Addendum 2-A	Qualification of the New Pin Power Recovery Methodology	Rev. 0	4.3 15.0.11.10	9/10	A
WCAP-11596-P-A	Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores	Rev. 0	4.3	6/88	A

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TABLE 1.6-2 (Sheet 21)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-14040-NP-A	Methodology Used to Develop Cold Overpressure Mitigation System Setpoints and RCS Heatup and Cooldown Limit Curves	Rev. 4	5.3.1.6.1		A
WCAP-15151	Westinghouse Archived Reactor Vessel Materials	12/98	5.3.1.6.1		
WCAP-12472-P-A	Addendum 1-A	Rev. 0	4.3.2.2.7	1/00	A
WCAP-15400	Analysis of Capsule X from Callaway Unit 1 Reactor Vessel Radiation Surveillance Program	6/00	5.3.4 Table 5.3-10		A
WCAP-14565-P-A Addendum 2-P-A	Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications	4/08	4.4.1.1		A

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FSAR Table 1.6-2 Markup

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Westinghouse Topical Report No.	Title	Rev. No.	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-16045-P-A	Qualification of the Two-Dimensional Transport Code PARAGON	Rev. 0	4.3	8/04	A
WCAP-16045-P-A Addendum 1-A	Qualification of the NEXUS Nuclear Data Methodology	Rev. 0	4.3	8/07	A

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LIST OF TABLES

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LIST OF FIGURES (Continued)

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TABLE 4.1-2 ANALYTICAL TECHNIQUES IN CORE DESIGN

Analysis	Technique	Computer Code	Section Referenced
Mechanical design of core internals, loads, deflections, and stress analysis	Static and dynamic modeling	MULTIFLEX, FORCE 2, LATFORC, finite element, structural analysis code, and others	3.7(N).2.1 3.9(N).2 3.9(N).3
Fuel rod design			
Fuel performance characteristics (temperature, internal pressure, clad stress, etc.)	Semiempirical thermal model of fuel rod with consideration of fuel density changes, heat transfer, fission gas release, etc.	Westinghouse fuel rod design model	4.2.1.2 4.2.1.3 4.2.3.2 4.2.3.3 4.3.3.1 4.4.2.11
Nuclear design			
1. Cross sections and group constants	Microscopic data; macroscopic constants for homogenized core regions; group constants for control rods with self-shielding	ENDF/B-VI PHOENIX-P	4.3.3.2 4.3.3.2 4.3.3.2
2. X-Y power distributions, fuel depletion, critical boron concentrations, X-Y xenon distributions, reactivity coefficients	2-D and 3-D, 2-group diffusion theory	ANC	4.3.3.3

PARAGON/NEXUS

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TABLE 4.1-2 (Sheet 2)

Analysis	Technique	Computer Code	Section Referenced
3. Axial power distributions, control rod worths, and axial xenon distribution	1-D, 2-group diffusion theory	APOLLO	4.3.3.3
4. Fuel rod power	Integral transport theory	LASER	4.3.3.1
Effective resonance temperature	Monte Carlo weighting function	REPAD	
5. Criticality of reactor and fuel assemblies	2-D, multigroup-transport theory	PHOENIX-P PARAGON/NEXUS	4.3.2.6
6. Vessel irradiation	Multigroup spatial dependent transport theory	DOT	4.3.2.8
Thermal-hydraulic design			
1. Steady state	Subchannel analysis of local fluid conditions in rod bundles, including inertial and crossflow resistance terms, solution is based on a one-pass model which simulates the core.	VIPRE-01	4.4.4.5.2

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TABLE 4.1-2 (Sheet 3)

Analysis	Technique	Computer Code	Section Referenced
2. Transient departure from nucleate boiling analysis	Subchannel analysis of local fluid conditions in rod bundles during transients by including accumulation terms in conservation equations; solution is based on a one-pass model which simulates the core. Including the hot assembly and hot subchannel.	VIPRE-01	4.4.4.5.2

No changes. Page included for completeness.

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No changes, page included for completeness.

The maximum reactivity insertion rate due to withdrawal of rod cluster control assemblies at power or by boron dilution is limited. During normal at power operation, the maximum controlled reactivity insertion rate is less than 35 pcm/sec*. A maximum reactivity change rate of 85 pcm/sec* for accidental withdrawal of control banks at hot zero power is set such that peak heat generation rate and DNBR do not exceed the maximum allowable at over-power conditions. This satisfies GDC-25.

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited so as to preclude rupture of the coolant pressure boundary or disruption of the core internals to a degree which would impair core cooling capacity due to a rod withdrawal or ejection accident (see Chapter 15.0).

Following any Condition IV event (rod ejection, steam line break, etc.) the reactor can be brought to the shutdown condition, and the core will maintain acceptable heat transfer geometry. This satisfies GDC-28.

Discussion

Reactivity addition associated with an accidental withdrawal of a control bank (or banks) is limited by the maximum rod speed (or travel rate) and by the worth of the bank(s). For this reactor, the maximum control rod speed is 45 inches per minute, and the maximum rate of reactivity change considering two control banks moving is 85 pcm/sec* at hot zero power. During normal operation at power and with control rod overlap, the maximum reactivity change rate is less than 35 pcm/sec*.

The reactivity change rates are conservatively calculated, assuming unfavorable axial power and xenon distributions. The peak xenon burnout rate is 25 pcm/min, significantly lower than the maximum reactivity addition rate of 35 pcm/sec for normal operation and 85 pcm/sec for accidental withdrawal of two banks at hot zero power.

4.3.1.5 Shutdown Margins

Basis

Minimum shutdown margin as specified in the COLR is required at any power operating condition, in the hot standby shutdown condition, and in the cold shutdown condition.

In all analyses involving reactor trip, the single, highest worth rod cluster control assembly is postulated to remain untripped in its fullout position (stuck rod criterion). This satisfies GDC-26.

Discussion

* 1 pcm = $10^{-5}\ \Delta\rho$ (see footnote to Table 4.3-2A).

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Two independent reactivity control systems are provided: control rods and soluble boron in the coolant. The control rod system can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full-load to no-load. In addition, the control rod system provides the minimum shutdown margin under Condition I events and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits (very small number of rod failures), assuming that the highest worth control rod is stuck out upon trip.

The boron system can compensate for all xenon burnout reactivity changes and will maintain the reactor in the cold shutdown. Thus, backup and emergency shutdown provisions are provided by a mechanical and a chemical shim control system which satisfies GDC-26.

Basis

When fuel assemblies are in the pressure vessel and the vessel head is not in place, k_{eff} will be maintained at or below 0.95 with control rods and soluble boron. Further, the fuel will be maintained sufficiently subcritical that removal of all rod cluster control assemblies will not result in criticality.

Discussion

ANSI Standard NI8.2 specifies a k_{eff} not to exceed 0.95 in spent fuel storage racks and transfer equipment flooded with pure water and a k_{eff} not to exceed 0.98 in normally dry new fuel storage racks, assuming optimum moderation. No criterion is given for the refueling operation. However, a 5-percent margin, which is consistent with spent fuel storage and transfer and the new fuel storage, is adequate for the controlled and continuously monitored operations involved.

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The boron concentration required to meet the refueling shutdown criteria is specified in the Technical Specifications. Verification that this shutdown criteria is met, including uncertainties, is achieved using standard Westinghouse design methods such as the ~~PHOENIX-P (Reference 36) and ANC (Reference 34) codes.~~ The subcriticality of the core is continuously monitored, as described in the Technical Specifications.

4.3.1.6 Stability

Basis

The core will be inherently stable to power oscillations at the fundamental mode. This satisfies GDC-12.

Spatial power oscillations within the core with a constant core power output, should they occur, can be reliably and readily detected and suppressed.

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... PHOENIX-P (Reference 36) or PARAGON/NEXUS (References 41 and 42) codes and the ANC (References 34 and 43) code. ...

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4.3.2.3 Reactivity Coefficients

The kinetic characteristics of the reactor core determine the response of the core to changing plant conditions or to operator adjustments made during normal operation, as well as the core response during abnormal or accidental transients. These kinetic characteristics are quantified in reactivity coefficients. The reactivity coefficients reflect the changes in the neutron multiplication due to varying plant conditions, such as power, moderator or fuel temperatures, or pressure or void conditions, although the latter are relatively unimportant in the SNUPPS reactors. Since reactivity coefficients change during the life of the core, ranges of coefficients are employed in transient analysis to determine the response of the plant throughout life. The results of such simulations and the reactivity coefficients used are presented in Chapter 15.0. The reactivity coefficients are calculated on a corewise basis by radial and axial diffusion theory methods. The effect of radial and axial power distribution on core average reactivity coefficients is implicit in those calculations and is not significant under normal operating conditions. For example, a skewed xenon distribution which results in changing axial offset by 5 percent changes the moderator and Doppler temperature coefficients by less than 0.01 pcm/F and 0.03 pcm/F, respectively. An artificially skewed xenon distribution which results in changing the radial $F^N_{\Delta H}$ by 3 percent changes the moderator and Doppler temperature coefficients by less than 0.03 pcm/F and 0.001 pcm/F, respectively. The spatial effects are accentuated in some transient conditions, for example, in postulated rupture of the main steam line break and rupture of a rod cluster control assembly mechanism housing described in Sections 15.1.5 and 15.4.8, and are included in these analyses.

The analytical methods and calculational models used in calculating the reactivity coefficients are given in Section 4.3.3. These models have been confirmed through extensive testing of more than 30 cores similar to the plant described herein; results of these tests are discussed in Section 4.3.3.

Quantitative information for calculated reactivity coefficients, including fuel-Doppler coefficient, moderator coefficients (density, temperature, pressure, and void) and power coefficient is given in the following sections.

4.3.2.3.1 Fuel Temperature (Doppler) Coefficient

The fuel temperature (Doppler) coefficient is defined as the change in reactivity per degree change in effective fuel temperature and is primarily a measure of the Doppler broadening of U-238 and Pu-240 resonance absorption peaks. Doppler broadening of other isotopes is also considered but their contribution to the Doppler effect is small. An increase in fuel temperature increases the effective resonance absorption cross-sections of the fuel and produces a corresponding reduction in reactivity.

The fuel temperature coefficient is calculated by performing two-group, two or three dimension calculations, using the ANC Code ~~(Ref. 34)~~ (Refs. 34 and 43). Moderator temperature is held constant, and the power level is varied. Spatial variation of fuel temperature is taken into

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account by calculating the effective fuel temperature as a function of power density, as discussed in Section 4.3.3.1.

A typical Doppler temperature coefficient is shown in Figure 4.3-27 as a function of the effective fuel temperature (at BOL and EOL conditions). The effective fuel temperature is lower than the volume averaged fuel temperature, since the neutron flux distribution is nonuniform through the pellet and gives preferential weight to the surface temperature. A typical Doppler-only contribution to the power coefficient, defined later, is shown in Figure 4.3-28 as a function of relative core power. The integral of the differential curve on Figure 4.3-28 is the Doppler contribution to the power defect and is shown in Figure 4.3-29 as a function of relative power. The Doppler coefficient becomes more negative as a function of life as the Pu-240 content increases, thus increasing the Pu-240 resonance absorption, but the overall value becomes less negative since the fuel temperature changes with burnup, as described in Section 4.3.3.1. The upper and lower limits of Doppler coefficient used in accident analyses are given in Figure 15.0-2.

4.3.2.3.2 Moderator Coefficients

The moderator coefficient is a measure of the change in reactivity due to a change in specific coolant parameters, such as density, temperature, pressure, or void. The coefficients so obtained are moderator density, temperature, pressure, and void coefficients.

Moderator Density and Temperature Coefficients

The moderator temperature (density) coefficient is defined as the change in reactivity per degree change in the moderator temperature. Generally, the effects of the changes in moderator density as well as the temperature are considered together.

The soluble boron used in the reactor as a means of reactivity control also has an effect on moderator density coefficient, since the soluble boron poison density as well as the water density is decreased when the coolant temperature rises. A decrease in the soluble poison density introduces a positive component in the moderator coefficient. If the concentration of soluble poison is large enough, the net value of the coefficient may be positive.The effect of control rods is to make the moderator coefficient more negative since the thermal neutron mean free path, and hence the volume affected by the control rods, increases with an increase in temperature.

With burnup, the moderator coefficient becomes more negative, primarily as a result of boric acid dilution, but also to a significant extent from the effects of the buildup of plutonium and fission products.

The moderator coefficient is calculated for a range of plant conditions by performing two-group X-Y calculations, in which the moderator temperature (and density) is varied by about ±5°F about each of the mean temperatures. The moderator coefficient is shown as a function of core temperature and boron concentration for a typical unrodded

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on time-averaged equilibrium cycle reactor core parameters and power distributions; and, thus, are suitable for long-term nvt projections and for correlation with radiation damage estimates.

As discussed in Section 5.3, the irradiation surveillance program utilizes actual test samples to verify the accuracy of the calculated fluxes at the vessel.

4.3.3 ANALYTICAL METHODS

Calculations required in nuclear design consist of three distinct types, which are performed in sequence:

a. Determination of effective fuel temperatures

b. Generation of macroscopic few-group parameters

c. Space-dependent, few-group diffusion calculations

These calculations are carried out by computer codes which can be executed individually. However, at Westinghouse most of the codes required have been linked to form an automated design sequence which minimizes design time, avoids errors in transcription of data, and standardizes the design methods.

4.3.3.1 Fuel Temperature (Doppler) Calculations

Temperatures vary radially within the fuel rod, depending on the heat generation rate in the pellet, the conductivity of the materials in the pellet, gap, and clad, and the temperature of the coolant.

The fuel temperatures for use in most nuclear design Doppler calculations are obtained from a simplified version of the Westinghouse fuel rod design model described in Section 4.2.1.3 which considers the effect of radial variation of pellet conductivity, expansion-coefficient and heat generation rate, elastic deflection of the clad, and a gap conductance which depends on the initial fill gap, the hot open gap dimension, and the fraction of the pellet over which the gap is closed. The fraction of the gap assumed closed represents an empirical adjustment used to produce good agreement with observed reactivity data at BOL. Further gap closure occurs with burnup and accounts for the decrease in Doppler defect with burnup which has been observed in operating plants. For detailed calculations of the Doppler coefficient, such as for use in xenon stability calculations, a more sophisticated temperature model is used which accounts for the effects of fuel swelling, fission gas release, and plastic clad deformation.

Radial power distributions in the pellet as a function of burnup are obtained from LASER (Ref. 22) calculations.

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The effective U-238 temperature for resonance absorption is obtained from the radial temperature distribution by applying a radially dependent weighting function. The weighting function was determined from REPAD (Ref. 23) Monte Carlo calculations of resonance escape probabilities in several steady state and transient temperature distributions. In each case, a flat pellet temperature was determined which produced the same resonance escape probability as the actual distribution. The weighting function was empirically determined from these results.

The effective Pu-240 temperature for resonance absorption is determined by a convolution of the radial distribution of Pu-240 densities from LASER burnup calculations and the radial weighting function. The resulting temperature is burnup dependent, but the difference between U-238 and Pu-240 temperatures, in terms of reactivity effects, is small.

The effective pellet temperature for pellet dimensional change is that value which produces the same outer pellet radius in a virgin pellet as that obtained from the temperature model. The effective clad temperature for dimensional change is its average value.

Delete

The temperature calculational model has been validated by plant Doppler defect data, as shown in Table 4.3-7, and Doppler coefficient data, as shown in Figure 4.3-42. Stability index measurements also provide a sensitive measure of the Doppler coefficient near full power (see Section 4.3.2.7). It can be seen that Doppler defect data is typically within 0.2 percent Δρ of prediction.

4.3.3.2 Macroscopic Group Constants

Macroscopic few-group constants and consistent microscopic cross-sections (needed for feedback and microscopic depletion calculations) are generated for fuel cells by ~~a recent version of the PHOENIX-P code (Ref. 36).~~

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The PHOENIX-P computer code is a two-dimensional, multi-group, transport lattice code and capable of providing all necessary data for PWR analysis. Being a dimensional lattice code, PHOENIX-P does not rely on pre-determined spatial/spectral interaction assumptions for a heterogeneous fuel lattice, hence, will provide a more accurate multi-group flux solution than versions of historical codes. The PHOENIX-P computer code is approved by the USNRC as the lattice code for generating macroscopic and microscopic few group cross-sections for PWR analysis.

The solution for the detailed spatial flux and energy distribution is divided into two major steps in PHOENIX-P. In the first step, a two-dimensional fine energy group nodal solution is obtained which couples individual sub-cell regions (pellet, cladding, and moderator) as well as surrounding pins. PHOENIX-P uses a method based on the Carlvik's collision probability approach and heterogeneous response fluxes which preserves the heterogeneity of the pin cells and their surroundings. The nodal solution

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... two available lattice codes. They are the PHOENIX-P code (Ref. 36) and PARAGON/NEXUS code set (Ref. 41 and Ref. 42). A detailed description of each follows.

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provides accurate and detailed local flux distributions, which is then used to spatially homogenize the pin cells to fewer groups.

The second step in the solution process solves for the angular flux distribution using a standard S4 discrete ordinates calculation. This step is based on the group-collapsed and homogenized cross-sections obtained from the first step of the solution. The S4 fluxes are then used to normalize the detailed spatial and energy nodal fluxes. The normalized nodal fluxes are used to compute reaction rates, power distribution and to deplete the fuel and burnable absorbers. A standard B1 calculation is employed to evaluate the fundamental mode critical spectrum and to provide an improved fast diffusion coefficient for the core spatial codes.

The PHOENIX-P code employs a 70 energy group library (Ref. 37) which as been derived mainly from ENDF/B-VI (base) files. The PHOENIX-P cross sections library was designed to properly capture integral properties of the multi-group data during group collapse, and enabling proper modeling of important resonance parameters. The library contains all neutronic data necessary for modeling fuel, fission products, cladding and structural, coolant, and control/burnable absorber materials present in Light Water Reactor cores.

Group constants for burnable absorber cells, guide thimbles, instruments thimbles, control rod cells, and other non-fuel cells are also generated by PHOENIX-P.

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4.3.3.3 Spatial Few-Group Diffusion Calculations

Spatial few-group diffusion calculations historically consisted of 2-group X-Y calculations using an updated version of the TURTLE code (Ref. 10), and 2-group axial calculations using APOLLO (Ref. 18), an updated version of the PANDA code. However, with the advent of VANTAGE 5 fuel, and hence, axial features such as axial blankets and part length burnable absorbers, there has been a greater reliance on three-dimensional nodal codes such as 3D ANC (Advanced Nodal Code) ~~(Ref. 34)~~. The three-dimensional nature of the nodal codes provides both the radial and axial power distributions.

(Refs. 34 and 43).

Nodal three-dimensional calculations are carried out to determine the critical boron concentrations and power distributions. The moderator coefficient is evaluated by varying the inlet temperature in the same calculations used for power distribution and reactivity predictions.

Validation of ANC reactivity calculations is associated with the validation of the group constants themselves, as discussed in Section 4.3.3.2. Validation of the Doppler calculations is associated with the fuel temperature validation discussed in Section 4.3.3.1. Validation of the moderator coefficient calculations is obtained by comparison with plant measurements at hot zero power conditions as shown in Table 4.3.11.

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PARAGON/NEXUS has been approved by the NRC as the next generation of Westinghouse lattice code (References 41 and 42). PARAGON/NEXUS is a replacement for PHOENIX-P and its primary use will be to provide the same types of input data that PHOENIX-P generates for use in three-dimensional core simulator codes. This includes macroscopic cross-sections, microscopic cross-sections, pin factors for pin-power reconstruction calculations, discontinuity factors for a nodal method solution, and other data needed for safety analysis or other downstream applications.

PARAGON/NEXUS is based on collision probability-interface current cell coupling methods. PARAGON/NEXUS provides flexibility in modeling not available in PHOENIX-P, including exact cell geometry representation instead of cylinderization, multiple rings and regions within the fuel pin and the moderator cell geometry, and variable cell pitch. The solution method permits flexibility in choosing the quality of the calculation through both increasing the number of regions modeled within the cell and the number of angular current directions tracked at the cell interfaces.

The calculation scheme in PARAGON/NEXUS is based on the conventional lattice modules: resonance calculation, flux solution, leakage correction, and depletion. The detailed theory of these modules is described in References 41. The cross-section resonance calculation module is based on the space-dependent Dancoff method (Reference 41); it is a generalization of the PHOENIX-P methodology that permits subdivision of the fuel pin into many rings and, therefore, generates space-dependent self-shielded isotopic cross-sections. The flux solution module uses the interface current collision probability method and permits a detailed representation of the fuel cells (Reference 41). The other two modules (leakage and depletion) are similar to the ones used in PHOENIX-P.

The current PARAGON/NEXUS cross-section library is a many energy group library, based on the ENDF/B basic nuclear data, with the same group structure as the library currently used with PHOENIX-P. The PARAGON/NEXUS qualification library has been improved through the addition of more explicit fission products and fission product chains (Reference 41). PARAGON is, however, designed to employ any number of energy groups. PARAGON data is parameterized using the NEXUS cross-section generation system (Reference 42) prior to use by the 3D nodal diffusion code [ANC (Reference 34)].

PARAGON/NEXUS allows the use of ANC to perform three-dimensional explicit pin-by-pin calculations that account for history effects associated with control rod motion (e.g., fuel pin power recovery). Traditional pressurized-water reactors (PWRs) operate without significant insertion of control rod banks. The advent of new PWR core designs, wherein some control rod insertion is a typical mode of operation, has presented the likelihood of introducing significant heterogeneities, the cumulative effect of which

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will not be captured in fuel pin powers using conventional pin power methodology. To remedy these deficiencies of the conventional methodology of pin power reconstruction in ANC, an improved methodology has been introduced that follows the history of each individual fuel pin in ANC, and computes the fuel pin macroscopic cross-sections based on the fuel pin history and the local spectrum (Reference 43). The efficacy of the improved methodology for pin power recovery with the ANC code is critically dependent on the capabilities and results of the codes PARAGON and NEXUS. NEXUS parameterizes cross sections calculated by PARAGON for input to ANC. It is this capability that is a key to the improved methodology for pin power reconstruction.

The improved pin power methodology in ANC was qualified by comparisons of pin powers from single assembly ANC calculations to corresponding pin powers calculated by the pin by pin transport theory lattice code PARAGON at identical conditions. A wide range of control rod insertion and withdrawal scenarios was used for these comparisons including very challenging control rod history cases beyond those anticipated at actual core operating conditions. In addition to the single assembly control rod history scenarios, comparison of results between the improved and conventional pin power methodology for traditional unrodded PWR core simulations demonstrated that the improved method is as accurate as the conventional method for unrodded cases.

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No changes. Page included for completeness.

ANC is used in two-dimensional and three-dimensional calculations. ANC can be used for safety analyses and to calculate critical boron concentrations, control rod worths, reactivity coefficients, etc.

Axial calculations are used to determine differential control rod worth curves (reactivity versus rod insertion) and axial power shapes during steady-state and transient xenon conditions (flyspeck curve). Group constants are obtained from ANC three-dimensional calculations homogenized by flux volume weighting.

Validation of the spatial codes for calculating power distributions involves the use of incore and excore detectors and is discussed in Section 4.3.2.2.7.

Based on comparison with measured data it is estimated that the accuracy of current analytical methods is:

± 0.2% $\Delta\rho$ for Doppler defect
± 2 x 10-5 $\Delta\rho$/°F for moderator coefficient
± 50 ppm for critical boron concentration with depletion
± 3% for power distributions
± 0.2% $\Delta\rho$ for rod bank worth
± 4 pcm/step for differential rod worth
± 0.5 pcm/ppm for boron worth
± 0.1% $\Delta\rho$ for moderator defect

4.3.4 REFERENCES

1. "Westinghouse Anticipated Transients Without Reactor Trip Analysis," WCAP-8330, August 1974.

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TABLE 4.3-7 ~~COMPARISON OF MEASURED AND CALCULATED DOPPLER DEFECTS~~ DELETED.

Plant	Fuel Type	Core Burnup (MWD/MTU)	Measured (pcm)*	Calculated (pcm)
1	Air-filled	1800	1700	1710
2	Air-filled	7700	1300	1440
3	Air and helium-filled	8460	1200	1210

~~* pcm = $10^5 \times \ln (k_1/k_2)$~~

Table 4.3-7 is Deleted.

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FIGURE 4.3-42 HAS BEEN DELETED

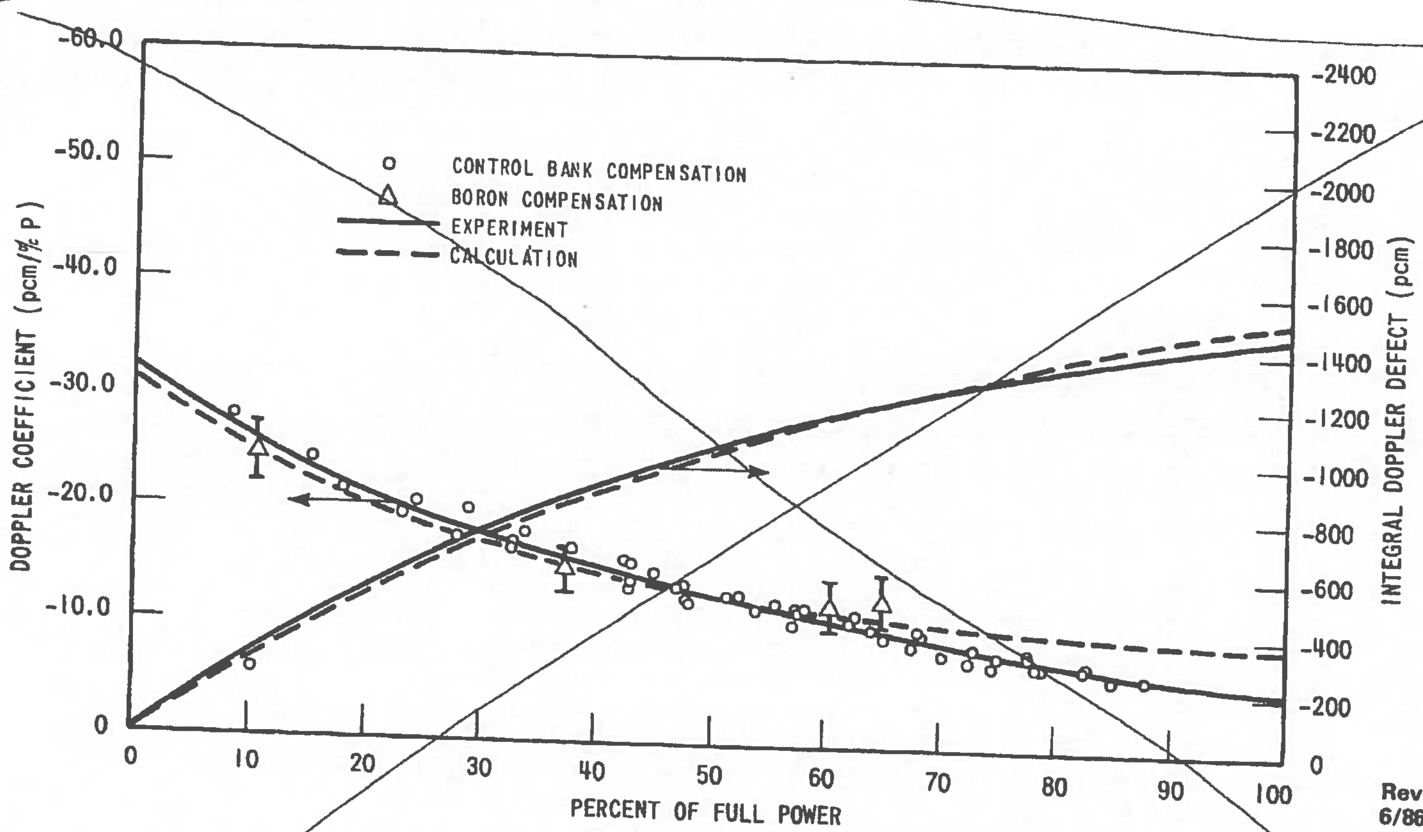

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FIGURE 4.3-42
CALCULATED AND MEASURED DOPPLER DEFECT AND COEFFICIENTS AT BOL, 2-LOOP PLANT, 121 ASSEMBLIES, 12 FOOT CORE

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local condition heat transfer. Component models include a two region nonequilibrium pressurizer, centrifugal and jet pumps, valves, non-conducting heat exchangers, steam separators, and turbine. An automatic steady state initialization procedure is also available.

The RETRAN code is discussed in Reference 18.

15.0.11.9 VIPRE

The VIPRE computer program performs thermal-hydraulic calculations. The code calculates coolant density, mass velocity, enthalpy, void fractions, static pressure and DNBR distributions along flow channels within a reactor core.

The VIPRE code is described in Reference 19.

15.0.11.10 ANC

ANC is an advanced nodal code capable of two-dimensional and three-dimensional neutronics calculations. ANC is the reference model for certain safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, three-dimensional ANC validates one-dimensional and two-dimensional results and provides information about radial (x-y) peaking factors as a function of axial position. It can calculate discrete pin powers from nodal information as well.

The ANC code is described in ~~Reference 20.~~ References 20 and 22.

15.0.12 LIMITING SINGLE FAILURES

The most limiting single failure as described in Section 3.1 of safety-related equipment, where one exists, is identified in each analysis description, and the consequences of this failure are described therein. In some instances, because of redundancy in protection equipment, no single failure which could adversely affect the consequences of the transient has been identified. The failure assumed in each analysis is listed in Table 15.0-7.

15.0.13 OPERATOR ACTIONS

For most of the events analyzed in Chapter 15.0 the plant will be in a safe and stable hot standby condition following the automatic actuation of reactor trip. This condition will, in fact, be similar to plant conditions following any normal, orderly shutdown of the reactor. At this point, the actions taken by the operator would be no different than normal operating procedures. The exact actions taken, and the time at which these actions would occur, will depend on what systems are available (e.g., turbine bypass system, main feedwater system, etc.) and the plans for further plant operation. As a minimum, to maintain the hot stabilized condition, decay heat must be removed via the steam

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FSAR 15.0.14

1) Each Level channel is tested one-at-a-time during the level channel testing with zero time delay as describd in the WCAP.

2) The TTD function and timers discussed in Reference 15 are no longer applicable in Callaway.

3) Section 3.6.2.2 is titled OUTAGE TESTING. The PROM logic modules and EAM testing described under this section may be performed on-line and not restricted to performance during outages.

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