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PG&E Letter DCL-06-136

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
License Amendment Request 06-05
Revision to Technical Specification 3.4.1, "Reactor Coolant System Pressure, Temperature, and Flow Departure from Nucleate Boiling Limits," and Technical Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)"

References: 1. Industry/Technical Specification Task Force (TSTF) Traveler No. TSTF-339, Revision 2, "Relocate TS Parameters to COLR," June 13, 2000

2. TSTF-363, Revision 0, "Revise Topical Report References in ITS 5.6.5, COLR," April 13, 2000

3. PG&E Letter DCL-06-006, "License Amendment Request 06-02 - Revision to Technical Specification 5.6.5, 'Core Operating Limits Report (COLR),'" dated January 13, 2006

Dear Commissioners and Staff:

In accordance with 10 CFR 50.90, enclosed is an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively. The enclosed license amendment request (LAR) proposes to revise Technical Specification (TS) 3.4.1, "Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)."

This LAR proposes to relocate the RCS DNB parameters for pressurizer pressure and RCS average temperature to the COLR. This relocation is consistent with Reference 1. TS 5.6.5, is revised to identify selected topical reports by title and number only. These changes are consistent with Reference 2. Also, TS 5.6.5 is revised to add topical reports WCAP-8567-P-A, "Improved Thermal Design Procedure," and WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores."

A001

EVALUATION

1.0 DESCRIPTION

This letter is a request to amend Operating Licenses DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively.

The proposed changes would revise the Operating Licenses to revise Technical Specification (TS) 3.4.1, "Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)."

This LAR proposes to relocate the RCS DNB parameters for pressurizer pressure and RCS average temperature to the COLR. This relocation is consistent with NRC-approved Industry/Technical Specification Task Force (TSTF) Traveler number TSTF-339, Revision 2, "Relocate TS Parameters to COLR," approved by the NRC staff on June 13, 2000 (Reference 1). TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," is revised to identify selected topical reports by title and number only. These changes are consistent with TSTF-363, Revision 0, "Revise Topical Report References in ITS 5.6.5, COLR" (Reference 2). Also TS 5.6.5 is revised to add topical reports WCAP-8567-P-A, "Improved Thermal Design Procedure," (Reference 3) and WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores" (Reference 9).

2.0 PROPOSED CHANGES

Limiting Condition for Operation (LCO) 3.4.1a and 3.4.1b are revised to state:

> a. *Pressurizer pressure is greater than or equal to the limit specified in the COLR;*
>
> b. *RCS average temperature is less than or equal to the limit specified in the COLR;*

Surveillance Requirements (SR) 3.4.1.1 and 3.4.1.2 are revised to state:

> *SR 3.4.1.1* *Verify pressurizer pressure is greater than or equal to the limit specified in the COLR.*
>
> *SR 3.4.1.2* *Verify RCS average temperature is less than or equal to the limit specified in the COLR;*

TS 5.6.5a. lists the core operating limits that are to be documented in the COLR and to be established prior to each reload cycle, or prior to any remaining portion of a reload cycle. The following limits are added to TS 5.6.5a. consistent with the TS changes proposed above.

9. *RCS pressure and temperature DNB limits in Specification 3.4.1.*

TS 5.6.5b identifies the approved topical reports and analytical methods used to determine the core operating limits. TS 5.6.5 is revised to add topical reports WCAP-8567-P-A, "Improved Thermal Design Procedure [ITDP]," and WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," as TSs 5.6.5b.9 and 10, respectively.

TS 5.6.5 is revised to specify five approved topical reports (5.6.5b.1, 2, 3, and new Nos. 9 and 10) by number and title only, consistent with TSTF-363.

Editorial corrections have been made by deleting the word "and" from the end of the topical reports listed in TS 5.6.5b.4, 5, 6, and 7, and replacing a period with a comma at the end of the topical report listed in TS 5.6.5b.8.

The TS bases will be updated to be consistent with the proposed TS changes, pursuant to TS 5.5.14, "Technical Specifications (TS) Bases Control Program," at the time this amendment is implemented.

The proposed TS changes are indicated on the marked-up TS pages provided in Enclosure 2. The proposed retyped TS pages are provided in Enclosure 3. The revised TS Bases are provided for information in Enclosure 4.

3.0 BACKGROUND

NRC Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications" (Reference 5), provides guidance to licensees for removing cycle-dependent variables from the TS, provided that the values of these variables are included in a COLR and are determined in accordance with NRC-approved methodology that is referenced in the TS. The changes proposed by this LAR meet these criteria.

WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," dated November 1995 (Reference 6) and approved by the NRC staff on January 19, 1999 (Reference 7), provides the justification to support the TS changes required to expand COLRs associated with Westinghouse plants. WCAP-14483-A includes justification to revise TS 3.4.1 to relocate the RCS DNB parameters for pressurizer pressure and RCS average temperature to the COLR.

TSTF-339, Revision 2, provided generic changes to NUREG-1431, Revision 1, "Standard Technical Specifications Westinghouse Plants," based on WCAP-14483-A, and was approved by the NRC staff on June 13, 2000.

The TS 3.4.1 limits on the DNB parameters assure that pressurizer pressure, RCS average temperature, and RCS total flow rate will be maintained within the limits of steady-state operation assumed in the Final Safety Analysis Report Update (FSARU) accident analyses. These limits are consistent with the initial full power conditions considered in the accident analyses. For Condition I and II events for which precluding DNB is the primary criterion, the safety analyses have demonstrated that the DNB design basis is satisfied, assuming that the plant is operating in compliance with the TS DNB parameter limits prior to initiation of the event. The DNB parameter limits are also based on the initial conditions assumed for Condition III and IV events for which precluding DNB is not a criterion. Given that the DNB parameter limits ensure that the DNB design basis and other safety criteria are satisfied, continuous plant operation at less than limiting conditions would result in margin to these safety criteria.

The limits for pressurizer pressure, RCS average temperature, and RCS total flow rate in TS 3.4.1 are evaluated in accordance with the methodologies described in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," dated July 1985 (Westinghouse Proprietary) (Reference 8). WCAP-9272-P-A is listed in TS 5.6.5b.2 as a topical report that describes analytical methods that have been previously reviewed and approved by the NRC that are used to determine the core operating limits.

4.0 TECHNICAL ANALYSIS

The justification to expand the COLR is provided in Westinghouse WCAP-14483-A. The changes to the Standard Technical Specifications were approved in TSTF-339, Revision 2.

The cycle-specific parameters being transferred from the TS to the COLR will continue to be controlled under existing programs and procedures. The FSARU accident analyses will continue to be examined with respect to changes in the cycle-dependent parameters obtained using NRC reviewed and approved reload design methodologies, ensuring that the transient evaluation of new reload designs are bounded by previously accepted analyses. This examination will continue to be performed pursuant to 10 CFR 50.59 requirements to ensure that future reload designs use NRC approved methodologies and do not involve more than a minimal increase in the probability or consequences of an accident previously evaluated in the FSARU.

TS 5.6.5 is revised to add topical reports WCAP-8567-P-A and WCAP-11596-P-A.

- WCAP-8567-P-A is the NRC reviewed and approved methodology topical report that describes the design method employed to meet the

DNB design basis. With the ITDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are combined statistically to obtain the overall DNB uncertainty factor that is used to define the design limit departure from nucleate boiling ratio (DNBR) that satisfies the DNB design criterion. DCPP FSARU Chapter 4.4, "Thermal and Hydraulic Design," discusses use of IDTP for DNBR calculations.

- WCAP-11596-P-A, is the NRC reviewed and approved methodology topical report for an advanced nodal code (ANC) capable of two-dimensional and three-dimensional calculations. In this design, ANC is employed as the reference model for all safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, and reactivity coefficients. In addition, three-dimensional ANC is used to validate one and two-dimensional results and to provide information about radial peaking factors as a function of axial position. PHOENIX-P is a two-dimensional, multi-group transport theory code that utilizes a 70 energy-group cross-section library. It provides the capability for cell lattice modeling on an assembly level. In this design, PHOENIX-P is used to provide homogenized, two-group cross sections for nodal calculations and feedback models. It is also used in a special geometry to generate appropriately weighted constants for the baffle/reflector regions. DCPP FSARU Chapter 4.3, "Nuclear Design," describes use of the ANC and Phoenix codes for core neutronic calculations.

Relocating the DNB parameters limit values to the COLR allows the flexibility to utilize the available margins to increase cycle operating margins and improve core reload designs without the need for cycle-specific license amendments. The relocation of the DNB parameters to the COLR results in a more complete COLR, containing not only cycle-specific core reload-related parameters, but also cycle-specific operating condition parameters. Thus, the safety analyses could credit the actual cycle-specific operating conditions in the same way that the core reload designs currently do. The COLR and safety analysis will more closely reflect the cycle-specific conditions for which the plant control and protection systems are set.

The proposed change to TS 5.6.5b to reference only the topical report number and title for five of the topical reports listed in TS 5.6.5b (1, 2, 3, and new Nos. 9 and 10) is consistent with TSTF-363. The purpose for this change is to allow the use of current topical reports to support limits in the COLR without having to submit a request for an amendment to the operating license. Implementation of revisions to these topical reports would still be reviewed in accordance with 10 CFR 50.59 and, where required, prior NRC review and approval would be requested. The remaining topical reports listed in TS 5.6.5b (4, 5, 6, and 7) are loss of coolant accident (LOCA) analysis methods. Consistent with recent NRC staff guidance regarding LOCA methodology (Reference 4) and 10 CFR 50.46 reporting requirements, the revision number and dates for these LOCA-related topical reports are being retained in TS 5.6.5b.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

PG&E 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 changes are programmatic and administrative in nature, and do not physically alter safety-related systems or affect the way in which safety-related systems perform their functions. The proposed changes relocate cycle-specific parameters from Technical Specification (TS) 3.4.1 to the Core Operating Limits Report (COLR). This does not change plant design or affect system operating parameters. The proposed changes do not, by themselves, alter any of the parameters. Removal of the cycle-specific parameters from the TS does not eliminate existing requirements to comply with the parameters. Also, TS 5.6.5 is revised to add topical reports WCAP-8567-P-A, "Improved Thermal Design Procedure," and WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," as they are approved analytical methods for determining core operating limits.

Although relocation of the cycle-specific parameters to the COLR would allow revision of the affected parameters without prior NRC approval, there is no significant effect on the probability or consequences of an accident previously evaluated. Future changes to the COLR parameters could result in event consequences that are either slightly less or slightly more severe than the consequences for the same event using the present parameters. The differences would not be significant and would be bounded by the existing requirement of TS 5.6.5c to meet the applicable limits of the safety analyses.

The cycle-specific parameters being transferred from the TS to the COLR will continue to be controlled under existing programs and procedures. The Final Safety Analysis Report Update (FSARU) accident analyses will continue to be examined with respect to changes in the cycle-dependent parameters obtained using NRC reviewed and approved reload design methodologies to ensure that the transient evaluation of new reload designs are bounded by previously accepted analyses. This examination will continue to be performed pursuant to 10 CFR 50.59 requirements, ensuring that future reload designs use NRC-approved methodologies and do not involve more than a minimal increase in the probability or consequences of an accident previously evaluated in the FSARU.

The proposed changes do not allow for an increase in plant power levels, do not increase the production, and do not alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes do not change the type or increase the amount of effluents released offsite.

The proposed changes to TS 5.6.5b to reference only the topical report number and title for five of the topical reports do not alter the analytical methods that have been previously reviewed and approved by the NRC. This method of referencing topical reports would allow the use of current topical reports to support limits in the COLR without having to submit a request for an amendment to the operating license. Implementation of revisions to these topical reports would still be reviewed in accordance with 10 CFR 50.59 and, where required, revisions would be submitted to the NRC for approval prior to implementation.

Therefore, the proposed 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 accident from any accident previously evaluated?

Response: No.

The proposed changes that relocate cycle-specific parameters from the TS to the COLR, thus removing the requirement for prior NRC approval of revisions to those parameters, do not involve a physical change to the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. No changes are being made to the parameters within which the plant is operated, other than their relocation to the COLR. No protective or mitigative action setpoints are affected by the proposed changes. The proposed changes will not alter the manner in which equipment operation is initiated, nor will the functional demands on credited equipment be changed. No change to procedures that ensure the plant remains within analyzed limits are being proposed, and no change is being made to procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced.

Relocation of cycle-specific parameters does not influence, impact, or contribute in any way to the possibility of a new or different kind of accident. The relocated cycle-specific parameters will continue to be calculated using the NRC-reviewed and approved methodology. The proposed changes do not alter assumptions made in the safety analysis, and operation within the core operating limits will continue.

The proposed changes to reference only the topical report number and title do not alter the use of the analytical methods that have been previously reviewed and approved by the NRC. This method of referencing topical reports would allow the use of current topical reports to support limits in the COLR without having to submit a request for an amendment to the operating license. Implementation of revisions to topical reports would still be reviewed in accordance with 10 CFR 50.59 and, where required, would receive NRC review and approval.

The addition of WCAP-8567-P-A and WCAP-11596-P-A to TS 5.6.5 is a clarification to provide a complete listing of approved analytical methods used for determining core operating limits.

Therefore, the proposed change does not create the possibility of a new or different 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. No protective or mitigative action setpoints are affected by the proposed changes. Therefore, sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. As the proposed changes to relocate cycle-specific parameters to the COLR will not affect plant design or system operating parameters, there is no detrimental impact on any equipment design parameter, and the plant will continue to be operated within prescribed limits.

The development of cycle-specific parameters for future reload designs will continue to conform to NRC-reviewed and approved methodologies, and will be performed pursuant to 10 CFR 50.59 to assure that the plant operates within cycle-specific parameters.

The proposed changes to reference only the topical report number and title do not alter the use of the analytical methods used to determine core operating limits that have been reviewed and approved by the NRC. This method of referencing topical reports would allow the use of current NRC-approved topical reports to support limits in the COLR without having to submit a request for an amendment to the operating license. Implementation of revisions to topical reports would still be reviewed in accordance with 10 CFR 50.59 and, where required, receive NRC review and approval.

The addition of WCAP-8567-P-A and WCAP-11596-P-A to TS 5.6.5 is a clarification to provide a complete listing of approved analytical methods used for determining core operating limits.

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

Based on the above evaluation, PG&E concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.36, "Technical Specifications," provides criteria for which limiting conditions for operation must be established. GL 88-16, which provides guidance for removal of cycle-specific parameter limits from TS, addresses conformance with 10 CFR 50.36 as follows:

> *The current method of controlling reactor physics parameters to assure conformance to 10 CFR 50.36 is to specify the specific value(s) determined to be within specified acceptance criteria (usually the limits of the safety analyses) using an approved calculation methodology. The alternative contained in this guidance controls the values of cycle-specific parameters and assures conformance to 10 CFR 50.36, which calls for specifying the lowest functional performance levels acceptable for continued safe operation, by specifying the calculation methodology and acceptance criteria. This permits operation at any specific value determined by the licensee, using the specified methodology, to be within the acceptance criteria. The Core Operating Limits Report will document the specific values of parameter limits resulting from*

licensee's calculations including any mid-cycle revisions to such parameter values.

The regulatory basis for TS 5.6.5 is to ensure that core operating limits are established in accordance with NRC-approved methodologies and documented in the COLR. GL 88-16 provided guidance for the removal of cycle-specific parameters from the TS, since processing cycle-specific limit changes was an unnecessary burden on both licensees and the NRC. The GL was intended to apply to those TS changes that were developed with NRC-approved methodologies. To support the removal of cycle-specific parameters, the GL recommended that cycle-specific parameter limit values be placed in a COLR, thereby eliminating the need for 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.

In conclusion, based on the considerations discussed above, (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.

6.0 ENVIRONMENTAL CONSIDERATION

PG&E has evaluated the proposed amendment and has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment 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 amendment.

7.0 REFERENCES

7.1 References

1. TSTF-339, Revision 2, "Relocate TS Parameters to COLR," approved by the NRC Staff on June 13, 2000

2. TSTF-363, Revision 0, "Revise Topical Report References in ITS 5.6.5, COLR," approved by the NRC Staff on April 13, 2000

3. WCAP-8567-P-A, "Improved Thermal Design Procedure," February 1989

4. NEI Summary of NRR Reactor Systems Branch Public Meeting on November 30, 2004, Regarding Proposed Regulatory Issue Summary on 10 CFR 50.46 Reporting Requirements and Changes to Chapter 15 Methodologies

5. NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 4, 1988

6. WCAP-14483-A, "Generic Methodology for Expanding Core Operating Limits Report," November 1995

7. T. H. Essig (NRC) Letter to A. P. Drake (WOG), "Acceptance for Referencing of Licensing Topical Report WCAP-14483-P-A, Generic Methodology for Expanded Core Operating Limits Report (TAC No. M94338)," January 19, 1999

8. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (Westinghouse Proprietary)

9. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988

7.2 Precedent

The changes proposed in the LAR for relocating RCS-related cycle-specific parameters are consistent with similar changes approved by the NRC for other nuclear power plants. These include changes approved for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, by License Amendment (LA) Nos. 106 and 113, respectively, issued on May 15, 2000; for Wolf Creek Generating Station by LA No. 144, issued on March 28, 2002; for Catawba Nuclear Station, Units 1 and 2 by LA Nos. 210 and 204, respectively, issued on December 19, 2003; for McGuire Nuclear Station, Units 1 and 2 by LA Nos. 219 and 201, respectively, issued on January 14, 2004; for Millstone Power Station, Unit No. 3 by LA No. 218 issued on March 9, 2004; and for Indian Point Nuclear Generating Unit No. 3 by LA No. 225, issued on March 24, 2005.

Proposed Technical Specification Changes (marked-up) is greater than or equal to the limit specified in the COLR

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ~~≥ 2197.3 psig~~;

b. RCS average temperature ~~≤ 584.3°F~~; and

c. RCS total flow rate within limits shown on Table 3.4.1-1 for Unit 1 and Table 3.4.1-2 for Unit 2.

is less than or equal to the limit specified in the COLR

APPLICABILITY: MODES 1.

-----NOTE-----
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

greater than or equal to the limit specified in the COLR

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ~~≥ 2197.3 psig~~.	12 hours
SR 3.4.1.2	Verify RCS average temperature is ~~≤ 584.3°F~~.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is within limits.	12 hours
SR 3.4.1.4	Verify measured RCS total flow rate is within limits.	24 months

less than or equal to the limit specified in the COLR

5.6 Reporting Requirements (continued)

5.6.3 Radioactive Effluent Release Report

---NOTE---

A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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 50, Appendix I, Section IV.B.1.

5.6.4 Not Used

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. Shutdown Bank Insertion Limits for Specification 3.1.5,

2. Control Bank Insertion Limits for Specification 3.1.6,

3. Axial Flux Difference for Specification 3.2.3,

4. Heat Flux Hot Channel Factor, K(Z) and W(Z) - $F_Q(z)$ (F_Q^{RTP} Specification 3.2.1),

5. RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$ ($F^{RTP}_{\Delta H}$ and $PF_{\Delta H}$ for

6. SHUTDOWN MARGI .6, and 3.1.8,

7. Moderator Temperature Coefficient limits in Specification 3.1.3, ~~and~~

8. Refueling Boron Concentration limits in Specification 3.9.1~~.~~ , and

9. RCS pressure and temperature DNB limits in Specification 3.4.1.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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-10216-P-A, ~~Revision 1A,~~ Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification, ~~February 1994~~ (Westinghouse Proprietary),

2. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, ~~July 1985~~ (Westinghouse Proprietary),

3. WCAP-8385, Power Distribution Control and Load Following Procedures, ~~September 1974~~ (Westinghouse Proprietary),

4. WCAP-10054-P-A, Westinghouse Small Break LOCA ECCS Evaluation Model Using the NOTRUMP Code, August 1985. (Westinghouse Proprietary), ~~and~~

5. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and COSI Condensation Model," July 1997 (Westinghouse Proprietary), ~~and~~

6. WCAP-12945-P-A, Westinghouse Code Qualification Document for Best-Estimate Loss of Coolant Analysis, June 1996. (Westinghouse Proprietary), ~~and~~

7. WCAP-12945-P-A, Addendum 1-A, Revision 0, “Method for Satisfying 10 CFR 50.46 Reanalysis Requirements for Best Estimate LOCA Evaluation Models,” December 2004. (Westinghouse Proprietary) (Unit 1 Only), ~~and~~

8. WCAP-16009-P-A, Revision 0, Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), January 2005. (Westinghouse Proprietary) (Unit 2 Only)~~.~~,

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)

9. WCAP-8567-P-A, “Improved Thermal Design Procedure,” and

10. WCAP-11596-P-A, “Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores.”

Proposed Technical Specification Changes (retyped)

Remove Page	Insert Page
3.4-1	3.4-1
5.0-26	5.0-26
5.0-27	5.0-27

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure is greater than or equal to the limit specified in the COLR;

b. RCS average temperature is less than or equal to the limit specified in the COLR; and

c. RCS total flow rate within limits shown on Table 3.4.1-1 for Unit 1 and Table 3.4.1-2 for Unit 2.

APPLICABILITY: MODES 1.

---NOTE---

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1	Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is less than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is within limits.	12 hours
SR 3.4.1.4	Verify measured RCS total flow rate is within limits.	24 months

5.6 Reporting Requirements (continued)

5.6.3 Radioactive Effluent Release Report

-----NOTE-----

A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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 50, Appendix I, Section IV.B.1.

5.6.4 Not Used

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. Shutdown Bank Insertion Limits for Specification 3.1.5,

2. Control Bank Insertion Limits for Specification 3.1.6,

3. Axial Flux Difference for Specification 3.2.3,

4. Heat Flux Hot Channel Factor, K(Z) and W(Z) - $F_Q(z)$ (F_Q^{RTP} Specification 3.2.1),

5. RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$ ($F^{RTP}_{\Delta H}$ and $PF_{\Delta H}$ for Specification 3.2.2),

6. SHUTDOWN MARGIN values in Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6, and 3.1.8,

7. Moderator Temperature Coefficient limits in Specification 3.1.3,

8. Refueling Boron Concentration limits in Specification 3.9.1, and

9. RCS pressure and temperature DNB limits in Specification 3.4.1.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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-10216-P-A, Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification, (Westinghouse Proprietary),
2. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, (Westinghouse Proprietary),
3. WCAP-8385, Power Distribution Control and Load Following Procedures, (Westinghouse Proprietary),
4. WCAP-10054-P-A, Westinghouse Small Break LOCA ECCS Evaluation Model Using the NOTRUMP Code, August 1985 (Westinghouse Proprietary),
5. WCAP-10054-P-A, Addendum 2, Revision 1, Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and COSI Condensation Model, July 1997 (Westinghouse Proprietary),
6. WCAP-12945-P-A, Westinghouse Code Qualification Document for Best-Estimate Loss of Coolant Analysis, June 1996 (Westinghouse Proprietary),
7. WCAP-12945-P-A, Addendum 1-A, Revision 0, "Method for Satisfying 10 CFR 50.46 Reanalysis Requirements for Best Estimate LOCA Evaluation Models," December 2004. (Westinghouse Proprietary) (Unit 1 Only),
8. WCAP-16009-P-A, Revision 0, Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), January 2005. (Westinghouse Proprietary) (Unit 2 Only),
9. WCAP-8567-P-A, "Improved Thermal Design Procedure," and
10. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores."

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)

Changes to Technical Specification Bases Pages
(For information only)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND	These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed. The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits. The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits. The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses and is variable with reactor thermal power down to 90% RTP as shown on Tables 3.4.1-1 and 3.4.1-2. Flow rate indications from the plant computer or RCS flow rate indicators are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the DNB limits to be approached. Operation for significant periods of time outside the limits on RCS flow, pressurizer pressure and average RCS temperature increases the likelihood of a fuel cladding failure if a DNB limited event were to occur.
APPLICABLE SAFETY ANALYSES	The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR correlation limit ~~of ≥ 1.17~~ (Ref. 2 and 3). This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criterion. The analyzed transients include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank

(continued)

specified in the COLR

the

in the safety

BASES

APPLICABLE SAFETY ANALYSES (continued)

Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit ~~of 2197.3 psig~~ and ~~the~~ RCS average temperature limit ~~of 584.3°F~~ correspond to ~~nominal~~ analytical limits ~~of 2250 psia and 577.6°F for Unit 2 (the limiting unit)~~ used ~~for the DNB calculation in the reload~~ analyses with allowance for ~~analysis initial consideration~~ uncertainty ~~(38 psi and 6.7°F)~~.

measurement

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables-pressurizer pressure, RCS average temperature, and RCS total flow rate to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow limits are provided for a RTP range of 90% to 100% on Tables 3.4.1-1 and 3.4.1-2 for Unit 1 and Unit 2 respectively.

The RCS total flow rate limit allows for a measurement error of 2.4% flow. Both the precision flow calorimetric method and the cold leg elbow tap method used to measure RCS flow meet the 2.4% flow uncertainty allowance.

The precision flow calorimetric method normalizes the RCS flow rate indicators to a precision flow calorimetric performed at the beginning of cycle. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision flow calorimetric in a non-conservative manner. A bias error of 0.1% for undetected fouling of the feedwater venturi is included in the measurement error analysis.

Any fouling that might significantly bias the feedwater flow rate input to the flow calorimetric measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The pressurizer pressure and RCS average temperature variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle.

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LCO (continued)	Use of the cold leg elbow tap method to measure RCS flow at approximately 100% RTP at the beginning of cycle results in a measurement uncertainty of ± 2.3% flow using the control board RCS flow rate indicators (which bounds the use of the plant process computer). This method is based on the utilization of twelve RCS cold leg elbow taps correlated to the four baseline precision heat balance measurements during Cycles 1 and 2 for each unit. Correlation of the flow indication channels with the flow calorimetric measurements performed during Cycles 1 and 2 is documented in WCAP-15113, Revision 1. Use of the cold leg elbow tap method provides an alternative to performance of a precision flow calorimetric to measure RCS flow and was approved by the NRC in amendments 161/162. The LCO numerical values for ~~pressure, temperature, and~~ flow rate has ~~have~~ not been adjusted for instrument error. The numerical values for pressure and temperature specified in the COLR have been adjusted for instrument error.
APPLICABILITY	In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure the DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

(continued)