Pacific Gas and Electric Company

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PG&E Letter DCL-04-178

U.S. Nuclear Regulatory Commission
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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 04-09
Relocation of Technical Specification Cycle Specific Parameters to the Core Operating Limits Report

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 proposed changes relocate reactor coolant system (RCS) related cycle-specific parameters from the Technical Specifications (TS) to the Core Operating Limits Report (COLR) for DCPP Units 1 and 2. The justification to implement this expansion of the COLR is provided in Westinghouse WCAP-14483-A, "Generic Methodology for Expanding Core Operating Limits Report," approved by the Nuclear Regulatory Commission (NRC) on January 19, 1999. The proposed changes are consistent with NRC-approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-339, Revision 2, "Relocate TS Parameters to COLR." Additionally, TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," is revised to identify three 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."

In accordance with TSTF-339, the proposed changes to TS 2.1.1, "Reactor Core Safety Limits," and associated Bases relocate Figure 2.1.1-1, "Reactor Core Safety Limit," to the COLR and replace it with the more specific fuel departure from nucleate boiling ratio and peak fuel centerline temperature safety limit requirements. The proposed changes to TS Table 3.3.1-1, "Reactor Trip System Instrumentation," relocate the Overtemperature ΔT and Overpower ΔT constant (K) values, the dynamic compensation time (τ) values, and the breakpoint and slope values for the

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f(ΔI) penalty functions to the COLR. Additionally, the proposed changes to TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and associated Bases relocate the pressurizer pressure, RCS average temperature, and RCS total flow rate values to the COLR. However, as discussed in WCAP-14483-A, the minimum limit for RCS total flow rate is retained in TS 3.4.1.

NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," dated October 4, 1988, provides guidance to licensees for the removal of cycle-dependent variables from the TS, provided that the values of these variables are included in a COLR and are determined with NRC-approved methodology which is referenced in the TS. The changes Pacific Gas and Electric (PG&E) Company proposes herein meet these criteria.

The changes proposed in this license amendment request (LAR) are consistent with similar changes approved by the NRC 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, and for Wolf Creek Generating Station by LA No. 144 issued on March 28, 2002.

Enclosure 1 contains a description of the proposed changes, the supporting technical analyses, and the no significant hazards consideration determination. Enclosures 2 and 3 contain marked-up and retyped (clean) TS pages, respectively. Enclosure 4 provides the marked-up TS Bases changes for information only. TS Bases changes will be implemented pursuant to TS 5.5.14, "Technical Specifications Bases Control Program," at the time this amendment is implemented.

PG&E has determined that this LAR does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

The changes proposed in this LAR are not required to address an immediate safety concern. PG&E requests approval of this LAR by December 31, 2005. PG&E requests the license amendments be made effective upon NRC issuance, to be implemented within 90 days from the date of issuance.

This communication contains no new or revised commitments.

If you have any questions or require additional information, please contact Stan Ketelsen at 805-545-4720.

Sincerely,

David H. Oatley
Vice President and General Manager

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Enclosures

cc:	Edgar Bailey, DHS
	Bruce S. Mallett
	David L. Proulx
	Diablo Distribution
cc/enc:	Girija S. Shukla

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

	)	Docket No. 50-275
In the Matter of	)	Facility Operating License
PACIFIC GAS AND ELECTRIC COMPANY	)	No. DPR-80
	)	
Diablo Canyon Power Plant	)	Docket No. 50-323
Units 1 and 2	)	Facility Operating License
	)	No. DPR-82

AFFIDAVIT

David H. Oatley, of lawful age, first being duly sworn upon oath says that he is Vice President and General Manager - Diablo Canyon of Pacific Gas and Electric Company; that he has executed license amendment request 04-09 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

David H. Oatley
Vice President and General Manager

Subscribed and sworn to before me this 28th day of December 2004.

Notary Public
County of San Luis Obispo
State of California

SANDRA L. RECTOR
Commission # 1339380
Notary Public - California
San Luis Obispo County
My Comm. Expires Jan 12, 2006

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 relocate reactor coolant system (RCS) related cycle-specific parameters from the Technical Specifications (TS) to the Core Operating Limits Report (COLR) for DCPP Units 1 and 2. The justification to implement this expansion of the COLR is provided in Westinghouse WCAP-14483-A, "Generic Methodology for Expanding Core Operating Limits Report," approved by the Nuclear Regulatory Commission (NRC) on January 19, 1999. The proposed changes are consistent with NRC approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-339, Revision 2, "Relocate TS Parameters to COLR." Additionally, TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," is revised to identify three 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."

2.0 PROPOSED CHANGE

<u>Reactor Core Safety Limits (SL)</u>

TS Figure 2.1.1-1, "Reactor Core Safety Limit," is being relocated to the COLR, and is being replaced with more specific fuel departure from nucleate boiling ratio (DNBR), and peak fuel centerline temperature safety limits based on megawatt days per metric ton of uranium (MWD/MTU).

Specifically TS 2.1.1, "Reactor Core SLs," is revised to state:

2.1.1 <u>Reactor Core SLs</u>

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥1.17 for the WRB-1/WRB-2 DNB correlations.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

The SLs proposed in TS 2.1.1.1 and TS 2.1.1.2 are described in the Final Safety Analysis Report Update (FSARU), Section 4.4, "Thermal and Hydraulic Design."

As documented in the Westinghouse Owners Group (WOG) response (Reference 1) to the NRC Request for Additional Information (RAI) (Reference 2) associated with Westinghouse WCAP-14483-A (Reference 3), because the reactor core safety limit figure is being relocated to the COLR, and is being replaced by more specific requirements regarding the safety limits (i.e., departure from nucleate boiling ratio limit and peak fuel centerline temperature limit), the "requirement" for the reactor core safety limit figure will be retained in the TS. The NRC approved methodology used to calculate the reactor core safety limit figure is contained in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology (Westinghouse Proprietary)," dated July 1985. (Reference 4) The existing TS 5.6.5, "COLR," Section b, specifically requires that the analytical methods used to determine the core operating limits be previously approved by the NRC. The analytical methods described in WCAP-9272-P-A are used to determine the core operating limits for DCPP. Therefore, the NRC's request (Reference 2) that the NRC-approved methodology used to derive the parameters in the figure should be referenced in the reporting requirements section of the TS, is currently in-place.

<u>Overtemperature (OT) ΔT and Overpower (OP) ΔT Parameters</u>

TS Table 3.3.1-1, Note 1: OTΔT, and Note 2: OPΔT, setpoint parameter constant (K_1 through K_6) values, dynamic compensation time (τ_1 through τ_5) values, T', T'', P', and the breakpoint and slope values for the f(ΔI) penalty functions for the trip setpoints are being relocated to the COLR.

Specifically, each of these parameters is replaced by an asterisk (*), with the following statement added to the bottom of each note: "The values denoted with * are specified in the COLR."

A typographical error is corrected in the Note 2 OPΔT equation replacing "ΔT^0" with "ΔT_0."

<u>RCS Pressure, Temperature and Flow Departure From Nucleate Boiling (DNB) Parameters</u>

The TS 3.4.1 limits specified for the RCS DNB parameters for pressurizer pressure and RCS average temperature are being relocated to the COLR.

Specifically the pressurizer pressure parameter is replaced and flow rate parameters are supplemented with the phrase: "is greater than or equal to the limit specified in the COLR." The RCS average temperature parameter is replaced with the phrase: "is less than or equal to the limit specified in the COLR."

The minimum limit for RCS total flow rates shown on Tables 3.4.1-1 for Unit 1 and Table 3.4.1-2 for Unit 2, based on NRC approved analysis methods, are being retained in TS 3.4.1. As documented in an NRC RAI, (Reference 2), a change in RCS flow from cycle-to-cycle is an indication of a physical change to the plant that should be reviewed by the NRC. To comply with this recommendation and the WOG response (Reference 1), the minimum limit for RCS total flow rate is being retained in TS 3.4.1 to assure that a lower flow rate than that reviewed by the NRC will not be used.

<u>COLR and COLR Analytical Methods</u>

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. *Reactor Core Safety Limits in Specification 2.1.1,*

10. *Overtemperature ΔT and Overpower ΔT trip setpoints in Specification 3.3.1, and*

11. *RCS pressure, temperature, and flow 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. Except as discussed in the following paragraph, this section is revised to specify three approved topical reports by number and title only consistent with TSTF-363, Revision 0, "Revise Topical Report References in ITS 5.6.5, COLR."

Three additional topical reports listed in TS 5.6.5b (Nos. 4, 5, and 6) are loss-of-coolant accident (LOCA) analysis methods. Consistent with recent NRC staff guidance regarding LOCA methodology (Reference 8), 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.

A typographical error is corrected by deleting the word "and" from the end of the topical report listed in TS 5.6.5b.4.

The TS changes proposed by this license amendment request (LAR) are noted on the markup TS pages provided in Enclosure 2. The proposed retyped TS pages are provided in Enclosure 3. Revisions to the TS Bases will be made *consistent with the above TS changes, and are shown 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 the removal of 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 was submitted to the NRC staff on December 1, 1995. This topical report provided justification to support the TS changes required to expand COLRs associated with Westinghouse plants. The NRC approved WCAP-14483 by letter dated January 19, 1999 (Reference 6). 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 on June 13, 2000.

TSTF-363, Revision 0, proposed revising Section 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," of NUREG-1431, Revision 1, to allow topical reports to be identified by number and title. TSTF-363, Revision 0, was approved by the NRC on April 13, 2000.

RCS Pressure, Temperature and Flow DNB Parameters

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 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.

Overtemperature ΔT and Overpower ΔT Parameters

The OTΔT and OPΔT reactor trip functions ensure that during any Condition I or II transient, there is at least a 95 percent probability that the peak kW/ft fuel rods will not exceed the uranium dioxide, UO_2, melting temperature. To achieve this, a fuel centerline temperature limit has been established, based on the melting temperature for UO_2 of 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup. For design purposes, this fuel centerline temperature limit is significantly below the melting temperature to allow for fuel temperature calculation and other uncertainties. In addition, the DNB design basis is defined as the probability being at least 95 percent at a 95 percent confidence level that DNB will not occur on the limiting fuel rod(s). If DNB is precluded, adequate heat transfer is assured between the fuel cladding and the reactor coolant, and damage due to inadequate cooling is prevented.

The OTΔT reactor trip function, in conjunction with the OPΔT reactor trip function, ensures operation within the DNB design basis and within the hot-leg boiling limits. Since both of these limits are functions of the coolant temperature, pressure, and core thermal power, the OTΔT reactor trip function is correlated with the temperature difference across the vessel (ΔT), the RCS average temperature, and pressurizer pressure. A compensating term which is a function of ΔI is also factored into the OTΔT setpoint to account for the affected changes in the axial power shape. The setpoint is scaled to be consistent with the full power operating conditions.

The OPΔT reactor trip function, in conjunction with the OTΔT reactor trip function, ensures operation within the fuel temperature design basis. This is accomplished through the OPΔT reactor trip function by correlating the core thermal power with the vessel ΔT. Since the thermal power is not precisely proportional to ΔT, because of the effects of changes in coolant density and heat capacity, a compensation term, which is a function of vessel average temperature, is factored into the calculated OPΔT trip setpoint. The setpoint is set to be consistent with the nominal full power operating conditions.

Reactor Core Safety Limits

TS Figure 2.1.1-1, "Reactor Core Safety Limits," presents the limiting vessel average temperature conditions as a function of pressurizer pressure and fractional RATED THERMAL POWER. This figure is included in the TS to satisfy the requirements of 10 CFR 50.36 that states "safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity." The reactor core safety limits figure provides the relationship between RCS average temperature, pressurizer pressure and RATED THERMAL POWER level, and the DNB design basis.

Because the reactor core safety limits figure is used in the generation of the OTΔT and OPΔT reactor trip setpoints, it contains the hot-leg boiling limits, which are not true safety limits. The hot-leg boiling limits preclude saturated conditions to ensure that the measured ΔT remains proportional to thermal power. The DNB limits of the figure are based on the DNBR safety analysis limit and assume a specific RCS flow rate and a symmetrical reference axial power shape. Based on this figure, the gains (K_1 through K_6) of the OTΔT and OPΔT reactor trip setpoints are generated. For non-symmetrical power shapes that are more limiting than the reference axial power shape, the f(ΔI) penalty function of the OTΔT reactor trip setpoint reduces the corresponding trip setpoint. (The f(ΔI) penalty function of the OPΔT reactor trip setpoint is not applicable to DCPP). Thus, the OTΔT and OPΔT reactor trip setpoints ensure that the reactor core safety limits figure is satisfied during a Condition I or II event and ensure that for non-symmetrical axial power shapes, the DNB design basis is satisfied. Because the OTΔT and OPΔT reactor trip setpoints are based on the reactor core safety limits figure, the only way to violate the figure is under the postulated condition where both trains of the reactor protection system (RPS) do not function as designed. Operation of the RPS and the steam generator safety valves ensure that the DNB design basis is satisfied for any Condition I or II transient, independent of the reactor core safety limits figure.

COLR Analytical Methods

The analytical methods used to determine the core operating limits are required to be those previously reviewed and approved by the NRC, as specifically described in the documents listed in TS 5.6.5b.

This LAR proposes removal of dates and revision numbers from the first three topical reports listed in TS 5.6.5b., in accordance with TSTF-363. This change will not affect PG&E's commitments to follow the conditions specified in the NRC safety evaluations approving those topical reports or future revisions thereto. In a letter dated December 15, 1999 (Reference 7), the NRC indicated that it is acceptable for the references to topical reports in TS Section 5.6.5 to give only the topical report number and title as long as the complete citation is given in the COLR. Any conditions imposed by the NRC in approving the topical reports, in the past or in the future, are included in the NRC safety evaluation that is incorporated in the topical report when it is issued as an approved report.

The remaining three topical reports listed in TS 5.6.5b (Nos. 4, 5, and 6) are LOCA analysis methods. Consistent with recent NRC staff guidance regarding LOCA methodology 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.

4.0 TECHNICAL ANALYSIS

The justification to implement the expansion of 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 ensuring that future reload designs will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The justification for allowing topical reports to be referenced by title and number was approved in TSTF-363, Revision 0. The proposed changes to TS 5.6.5b to reference only the topical report number and title for three of the topical reports 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 amend the operating license to use a revised NRC approved topical report. 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.

RCS Pressure, Temperature and Flow DNB Parameters

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.

Overtemperature ΔT and Overpower ΔT Parameters

Relocation of the OTΔT and OPΔT setpoint parameter values to the COLR minimizes the chance that a reload-related parameter change would necessitate a TS change. In addition, significant DNB and operating margin currently utilized in the setpoints that is unnecessarily allocated, and thus unavailable, could be utilized to enhance plant operating margins, enhance the OTΔT and OPΔT setpoints, and increase the flexibility of the core designs without any reduction in the margin of safety.

Reactor Core Safety Limits

The proper functioning of the RPS and steam generator safety valves prevents violation of the reactor core safety limits specified by current TS 2.1.1.

Replacement of the TS 2.1.1 reactor core safety limits figure with more specific fuel DNBR and peak fuel centerline temperature safety limits is acceptable, as these limits are criteria that must be satisfied for all Condition I and II transients. As with the current TS, appropriate functioning of the RPS and the steam generator safety valves will ensure that the safety limits are satisfied for any Condition I or II event. Replacing the reactor core safety limits figure with more specific fuel DNBR and peak fuel centerline temperature safety limits will reduce the chance of reaching an incorrect conclusion concerning whether or not a safety limit has been violated for a Condition I or II event. Violation of a safety limit could only result in the rare situation that the RPS was not functioning as designed.

COLR Analytical Methods

NRC GL 88-16 allows removal of cycle-dependent variables from the TS, provided that the values of these variables are included in a COLR and are determined with NRC reviewed and approved methodology, which is referenced in the TS. Safety limits, however, may not be placed in the COLR. The reactor core safety limit figure is being relocated to the COLR, and is being replaced by more specific requirements regarding the safety limits (i.e., departure from nucleate boiling ratio limit and peak fuel centerline temperature limit), The current reference to the NRC reviewed and approved methodology (Reference 4) used to calculate the reactor core safety limit figure is being retained in TS 5.6.5b.

TS 5.6.5b. identifies the approved topical reports and analytical methods used to determine the core operating limits. This section is revised to specify the approved topical reports by number and title only for three of the topical reports, consistent with TSTF-363, Revision 0. 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 an amendment to the facility operating license every time the topical report is revised. The COLR would provide specific information (i.e., report number, title, revision, date, and any supplements) identifying the particular approved topical report used to determine the core limits for the particular cycle in the COLR.

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, which do not physically alter safety related systems, nor affect the way in which safety related systems perform their functions. More specific requirements regarding the safety limits (i.e., departure from nucleate boiling ratio limit and peak fuel centerline temperature limit) are being imposed in Technical Specification (TS) 2.1.1, "Reactor Core SLs [Safety Limits]," which replace the reactor core safety limits figure and are consistent with the values stated in the Final Safety Analysis Report Update (FSARU). The proposed changes remove cycle-specific parameters from TS 3.4.1 and relocate them to the Core Operating Limits Report (COLR), which do not change the plant design or affect system operating parameters. In addition, the minimum limit for reactor coolant system (RCS) total flow rate is being retained in TS 3.4.1 to assure that a lower flow rate than reviewed by the NRC will not be used. The proposed changes do not, by themselves, alter any of the parameters. The removal of the cycle-specific parameters from the TS does not eliminate existing requirements to comply with the parameters.

The proposed changes to TS 5.6.5b to reference only the topical report number and title for three of the topical reports 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 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, receive NRC review and approval.

Although the 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 which 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 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, ensuring that future reload designs will not involve a significant increase in the probability or consequences of an accident previously evaluated. Additionally, the proposed changes do not allow for an increase in plant power levels, do not increase the production, nor 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 any effluents released offsite.

Therefore, the proposed changes do 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 retain the minimum limit for RCS total flow rate in the TS, and that relocate certain 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. There are no changes being made to the parameters within which the plant is operated, other than their relocation to the COLR. There are no setpoints affected by the proposed changes at which protective or mitigative actions are initiated. The proposed changes will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alteration in the procedures which ensure the plant remains within analyzed limits is being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced.

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 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.

Relocation of cycle-specific parameters has no influence or impact on, nor does it 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.

Therefore, the proposed changes do not create 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. 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 operate 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.

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

Based on the above evaluation, PG&E concludes that the proposed changes present 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

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 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 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.

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, record keeping, review and audit, and reporting, necessary to assure operation of the facility in a safe manner.

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 effluent 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. L. F. Liberatori Jr. (WOG) Response to Request for Additional Information to NRC Document Control Desk, November 25, 1998

2. P. C. Wen (NRC) Request for Additional Information to A. P. Drake (WOG), September 2, 1998

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

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

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

6. T. H. Essig (NRC) Letter to A. P. Drake (WOG), "Acceptance For Referencing Of Licensing Topical Report WCAP-14483, 'Generic Methodology for Expanded Core Operating Limits Report,'" January 19, 1999

7. S. A. Richards (NRC) Letter to J. F. Malley (Siemens Power Corporation), "Acceptance For Siemens References to Approved Topical Reports in Technical Specifications," December 15, 1999

8. 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

9. TSTF-339, Revision 2, "Relocate TS Parameters to COLR"

10. TSTF-363, Revision 0, "Revise Topical Report References in ITS 5.6.5, COLR."

7.2 Precedent

There are precedents for relocating RCS-related cycle-specific parameters from the Technical Specifications to the COLR. These precedents are based on the justification provided in WCAP-14483-A (Reference 3). The changes proposed in this LAR are consistent with similar changes approved by the NRC 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, and for Wolf Creek Generating Station by LA No. 144, issued on March 28, 2002.

Proposed Technical Specification Changes (mark-up)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the ~~SLs~~ limits specified in ~~Figure 2.1.1-1~~ the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for the WRB-1/WRB-2 DNB correlations.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

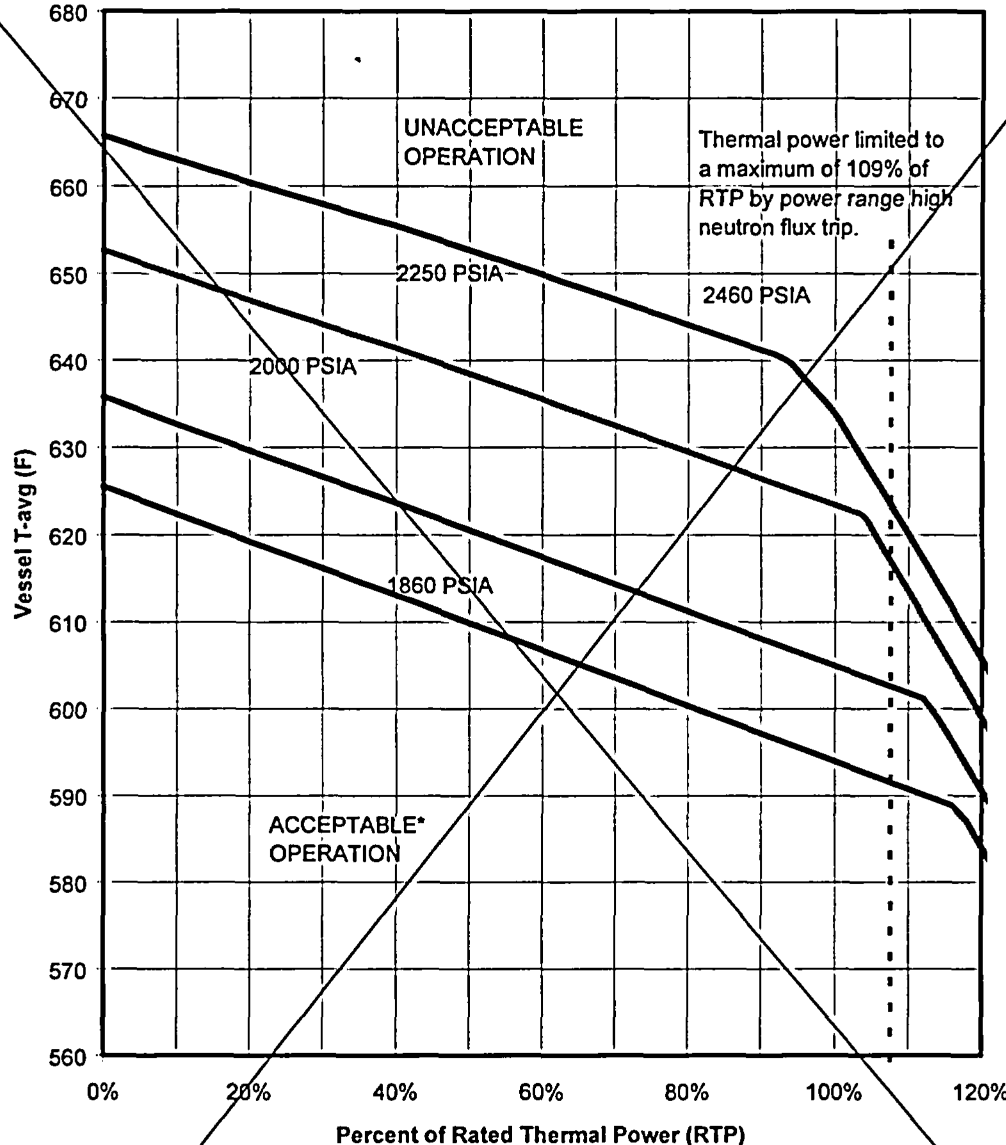
2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

Delete

UNITS 1 & 2



*When operating in the reduced RTP region of Technical Specification LCO 3.4.1 (Table 3.4.1-1 for Unit 1 and Table 3.4.1-2 for Unit 2) the restricted power level must be considered 100% for this Figure.

Figure 2.1.1-1
REACTOR CORE SAFETY LIMIT

Table 3.3.1-1 (page 6 of 7)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.46% of ΔT span for hot leg or cold leg temperature inputs, 0.14% ΔT span for pressurizer pressure input, 0.19% ΔT span for ΔI inputs.

$$\Delta T\frac{(1+\tau_4 s)}{(1+\tau_5 s)} \le \Delta T_0 \left\{K_1 - K_2\frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left[T-T'\right] + K_3(P-P') - f_1(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.

ΔT_0 is the loop specific indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.

T' is the nominal loop specific indicated T_{avg} at RTP, ≤ ~~577.3~~ (Unit 1) & ~~577.6~~ (Unit 2)°F. $\boxed{*}$

P is the measured pressurizer pressure, psig

P' is the nominal RCS operating pressure, = ~~2235~~ psig

$\boxed{*}$ K_1 = ~~1.20~~	K_2 = ~~0.0182~~/°F $\boxed{*}$	K_3 = ~~0.000831~~/psig $\boxed{*}$
τ_1 = ~~30~~ sec	τ_2 = ~~4~~ sec	
τ_4 = ~~0~~ sec	τ_5 = ~~0~~ sec	
$f_1(\Delta I)$ =	~~-0.0275~~{ ~~19~~ + (q_t - q_b)}	when q_t - q_b ≤ ~~-19~~% RTP
	0% of RTP $\boxed{*}$	when ~~-19~~% RTP < q_t - q_b ≤ ~~7~~% RTP
	~~0.0238~~{(q_t - q_b) - ~~7~~}	when q_t - q_b > ~~7~~% RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

The values denoted with * are specified in the COLR

$\boxed{\Delta T_0}$

Table 3.3.1-1 (page 7 of 7)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.46% of ΔT span for hot leg or cold leg temperature inputs.

$$\Delta T\frac{(1+\tau_4 s)}{(1+\tau_5 s)} \leq \Delta T^0\left\{K_4 - K_5\frac{\tau_3 s}{1+\tau_3 s}T - K_6[T - T'] - f_2(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.

ΔT_0 is the loop specific indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.

$\boxed{*}$

T" is the nominal loop specific indicated T_{avg} at RTP, ≤ ~~577.3~~ (Unit 1) & ~~577.6~~ (Unit 2)°F.

$\boxed{*}$ K_4 = ~~1.072~~ τ_3 = ~~10~~ sec

$\boxed{*}$ K_5 = ~~0.0174~~/°F for increasing T_{avg} ~~0~~/°F for decreasing T_{avg} τ_4 = ~~0~~ sec

$\boxed{*}$ K_6 = ~~0.00145~~/°F when T >T" ~~0~~/°F when T ≤ T" τ_5 = ~~0~~ sec

$f_2(\Delta I)$ = ~~0% RTP for all ΔI~~.

Note 3: Steam Generator Water-Level Low Low Time Delay

The Steam Generator Water Level-Low Low time delay function power allowable value shall not exceed the following trip setpoint power by more than 0.7% RTP.

TD = $B1(P)^3 + B2(P)^2 + B3(P) + B4$

Where: P = RCS Loop ΔT Equivalent to Power (%RTP), P ≤ 50% RTP

TD = Time delay for Steam Generator Water Level Low-Low Reactor Trip (in seconds).

B1 = -0.007128 sec/$(RTP)^3$

B2 = +0.8099 sec/$(RTP)^2$

B3 = -31.40 sec/(RTP)

B4 = +464.1 sec

The values denoted with * are specified in the COLR.

3.4 REACTOR COOLANT SYSTEM (RCS)

is

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~~;

is

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.

and

greater 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

SURVEILLANCE REQUIREMENTS

and

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

Table 3.4.1-1 (page 1 of 1)

Reduction in Percent RATED THERMAL POWER for Reduced RCS Flow Rate

Unit 1

RCS total Flow [a] (10^4 GPM)	Acceptable Operating Region [b] (% RTP)
≥ 35.9	≤ 100%
≥ 35.6	≤ 98%
≥ 35.2	≤ 96%
≥ 34.8	≤ 94%
≥ 34.5	≤ 92%
≥ 34.1	≤ 90%

(a) For RCS Total Flow < 341,000 GPM, entry into LCO 3.4.1 Condition A is required.

(b) When operating in the restricted power limits, the restricted power level shall be considered 100% RTP for Figure 2.1.1-1.

No changes to this page

Table 3.4.1-2 (page 1 of 1)
Reduction in Percent RATED THERMAL POWER for Reduced RCS Flow Rate
Unit 2

RCS Total Flow[(a)] (10^4 GPM)	Acceptable Operating Region[(b)] (% RTP)
≥ 36.3	≤ 100%
≥ 35.9	≤ 98%
≥ 35.5	≤ 96%
≥ 35.2	≤ 94%
≥ 34.8	≤ 92%
≥ 34.4	≤ 90%

(a) For RCS Total Flow < 344,000 GPM, entry into LCO 3.4.1 Condition A is required.

(b) When operating in the restricted power limits, the restricted power level shall be considered 100% RTP for Figure 2.1.1-1.

No changes to this page

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
Control Program and in conforma
Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statis
monthly basis no later than the 15
covered by the report.

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, ~~and~~
8. Refueling Boron Concentration limits in Specification 3.9.1~~.~~ ,
9. Reactor Core Safety Limits in Specification 2.1.1,
10. Overtemperature ΔT and Overpower ΔT trip setpoints in Specification 3.3.1, and
11. RCS pressure, temperature, and flow 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).

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)

Proposed Technical Specification Changes (retyped)

Remove Page	Insert Page
2.0-1	2.0-1
2.0-2	-----
3.3-17	3.3-17
3.3-18	3.3-18
3.4-1	3.4-1
5.0-26	5.0-26
5.0-27	5.0-27

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for the WRB-1/WRB-2 DNB correlations.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

Table 3.3.1-1 (page 6 of 7)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.46% of ΔT span for hot leg or cold leg temperature inputs, 0.14% ΔT span for pressurizer pressure input, 0.19% ΔT span for ΔI inputs.

$$\Delta T \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \le \Delta T_0 \left\{ K_1 - K_2 \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[T - T' \right] + K_3(P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.

ΔT_0 is the loop specific indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.

T' is the nominal loop specific indicated T_{avg} at RTP, ≤ * (Unit 1) & * (Unit 2)°F.

P is the measured pressurizer pressure, psig

P' is the nominal RCS operating pressure, = * psig

K_1 = *	K_2 = */°F	K_3 = */psig
τ_1 = * sec	τ_2 = * sec	
τ_4 = * sec	τ_5 = * sec	
$f_1(\Delta I)$ =	* { * + (q_t - q_b)}	when $q_t - q_b \le$ *% RTP
	0% of RTP	when *% RTP < $q_t - q_b \le$ *% RTP
	* {(q_t - q_b) - * }	when $q_t - q_b$ > *% RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

The values denoted with * are specified in the COLR.

Table 3.3.1-1 (page 7 of 7)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.46% of ΔT span for hot leg or cold leg temperature inputs.

$$\Delta T\frac{(1+\tau_4 s)}{(1+\tau_5 s)} \leq \Delta T_0\left\{K_4 - K_5\frac{\tau_3 s}{1+\tau_3 s}T - K_6\left[T - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.

ΔT_0 is the loop specific indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.

T" is the nominal loop specific indicated T_{avg} at RTP, ≤ * (Unit 1) & * (Unit 2)°F.

K_4 = *	K_5 = */°F for increasing T_{avg} */°F for decreasing T_{avg}	K_6 = */°F when T > T" */°F when T ≤ T"
τ_3 = * sec	τ_4 = * sec	τ_5 = * sec

$f_2(\Delta I)$ = *.

The values denoted with * are specified in the COLR.

Note 3: Steam Generator Water-Level Low Low Time Delay

The Steam Generator Water Level-Low Low time delay function power allowable value shall not exceed the following trip setpoint power by more than 0.7% RTP.

TD = $B1(P)^3 + B2(P)^2 + B3(P) + B4$

Where: P = RCS Loop ΔT Equivalent to Power (%RTP), P ≤ 50% RTP

TD = Time delay for Steam Generator Water Level Low-Low Reactor Trip (in seconds).

B1 = -0.007128 sec/$(RTP)^3$

B2 = +0.8099 sec/$(RTP)^2$

B3 = -31.40 sec/(RTP)

B4 = +464.1 sec

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 and greater 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

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 and greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.4	Verify measured RCS total flow rate is within limits and greater than or equal to the limit specified in the COLR.	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 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

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,

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

9. Reactor Core Safety Limits in Specification 2.1.1,

10. Overtemperature ΔT and Overpower ΔT trip setpoints in Specification 3.3.1, and

11. RCS pressure, temperature, and flow DNB limits in 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-10216-P-A, Revision 1A, 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), and

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

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 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

No changes on this page.

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

RCS flow, ΔI,

The Reactor Trip System setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

appropriate operation of the RPS and the

Protection for these reactor core SLs is provided by the steam generator safety valves ~~and the following automatic reactor trip functions:~~ .

~~a. High pressurizer pressure trip;~~

~~b. Low pressurizer pressure trip;~~

~~c. Overtemperature Δ T trip;~~

~~d. Overpower Δ T trip;~~

~~e. Power Range Neutron Flux trip~~

~~The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the Δ T measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.~~

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. ~~5~~) provide more restrictive limits to ensure that the SLs are not exceeded.

3

(continued)

BASES (continued)

SAFETY LIMITS

The figure provided in the COLR shows ~~The safety limit is based upon~~ the loci of points of THERMAL POWER, RCS pressure, and average temperature below which the calculated DNBR is not less than the design DNBR value, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

Bases Insert 1

~~The curves are based on enthalpy rise hot channel factor limits provided in the COLR.~~

~~The SL is higher than the limit calculated when the AFD is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Refs. 3 and 4).~~

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 (associated with 1967 GDC 6 per FSAR Appendix 3.1A).
2. FSAR, Chapter 7.2.
~~3. WCAP-8746-A, March 1977.~~
~~4. WCAP-9273-NP-A, July 1985.~~
3 ~~5~~. FSAR, Chapter 15.

Bases Insert 1

The reactor core SLs are established to preclude the violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

Bases Insert 2

The pressurizer pressure limit and the RCS average temperature limit specified in the COLR correspond to the analytical limits used in the safety analyses, with allowance for measurement uncertainty.

Bases Insert 3

These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, usually based on maximum analyzed steam generator tube plugging, is retained in the TS LCO.

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

criterion

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)

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)."

Bases Insert 2

~~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).~~

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.

Bases Insert 3

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 penalty for undetected fouling of the feedwater venturi

(continued)

BASES

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.

specified in the COLR are given for the measurement location and

The ~~LCO~~ numerical values for pressure, temperature, and flow rate have ~~not~~ been adjusted for instrument error.

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)

BASES

APPLICABILITY (continued)

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational pressure transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

The DNBR limit

~~Another set of limits on DNB related parameters~~ is provided in SL 2.1.1, "Reactor Core SLs." ~~Those limits~~ are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

The conditions which define the DNBR limit

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and reduce the potential for violation of the accident analysis limits.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition reduces the potential for violation of the accident analysis limits. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

(continued)

No changes on this page.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

The 12 hour Surveillance Frequency for the indicated RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions. The term "indicated RCS total flow" is used to distinguish between the "measured RCS total flow" determined in SR 3.4.1.4.

SR 3.4.1.4

SR 3.4.1.4 has two surveillance requirements, one for the CHANNEL CALIBRATION of the RCS flow indicators and the other for measurement of RCS total flow rate. Measurement of RCS total flow rate by performance of a precision flow calorimetric or by using the cold leg elbow tap methodology once every 24 months allows the installed RCS flow instrumentation to be normalized and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.4 (continued)

The second part of this surveillance is the routine CHANNEL CALIBRATION of the RCS flow indication instrumentation. The routine calibration of the flow instrumentation ensures that the channels are within the necessary range and accuracy for proper flow indication. The routine CHANNEL CALIBRATION of the RCS flow indication instrumentation is performed every 24 months.

The Frequency of 24 months for the measurement of RCS total flow rate reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance. Flow verification demonstrates that setpoints are relevant and RCS flow resistance is within limits. The frequency of 24 months for the routine CHANNEL CALIBRATION of the flow indication instrumentation is based on operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. FSAR, Section 15.

~~2. Diablo Canyon Power Plant Unit 1 Cycle 9 Reload Safety Evaluation, August 1995.~~

~~3. Diablo Canyon Power Plant Unit 2 Cycle 8 Reload Safety Evaluation, Rev.1, April 1996.~~

2 → 4. FSAR, Table 4.1-1.

3 → 5. WCAP-15113, Revision 1, "RCS Flow Measurement Using Elbow Tap Methodology at Diablo Canyon Units 1 and 2," April, 2002.