AmerenUE
Callaway Plant

PO Box 620
Fulton, MO 65251

September 20, 2006

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop P1-137
Washington, DC 20555-0001

Ladies and Gentlemen: ULNRC-05315

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
APPLICATION FOR LICENSE AMENDMENT
TO REVISE TECHNICAL SPECIFICATION 5.6.6;
REACTOR COOLANT SYSTEM (RCS)
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

Pursuant to 10CFR50.90, Union Electric, d.b.a. AmerenUE hereby requests an amendment to the Facility Operating License No. NPF-30 for Callaway Plant.

The proposed amendment would revise the Technical Specification (TS) consistent with the NRC-approved Revision 0 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-419, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR." The proposed change to reference only the Topical Report number and title in TS 5.6.6 does not alter the analytical methods used to determine the pressure-temperature (P/T) limits or cold over pressure mitigation system (COMS) setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of currently approved Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license.

Attachment 1 provides a detailed description of the proposed changes, a technical analysis of the proposed changes, AmerenUE's determination that the proposed changes do not involve a significant hazard consideration, a regulatory analysis of the proposed changes and an environmental evaluation. Attachment 2 provides the affected TS pages marked-up to reflect the proposed changes. Attachment 3 provides retyped TS pages which incorporate the requested changes. Attachment 4 provides a copy of the proposed TS Bases change in support of this License Amendment request.

The Callaway Onsite Review Committee and a subcommittee of the Nuclear Safety Review Board have reviewed and approved this amendment application.

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AmerenUE requests approval of the proposed License Amendment by February 28, 2007, to be implemented within 90 days of the issuance of the license amendment.

This communication contains no new or revised commitments.

Should you have any questions, please contact Dave Shafer (314) 554-3104.

In accordance with 10CFR50.91(b), AmerenUE is providing the State of Missouri a copy of this proposed amendment.

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

Very truly yours,

Executed on: September 20, 2006

Todd A. Moore

for Keith Young
Manager, Regulatory Affairs

PMB/jdg

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(Certrec receives ALL attachments as long as they are non-safeguards and public disclosed).

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ATTACHMENT I
ULNRC-05315

EVALUATION

EVALUATION

1.0 DESCRIPTION

The proposed license amendment revises the TS requirements related to the Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-419, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," Revision 0.

2.0 PROPOSED CHANGE

Consistent with Revision 0 of TSTF-419, the proposed TS changes include:

1. The definition of PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) is revised to delete the reference to the specifications containing the limits specified in the PTLR.

2. The requirement in TS 5.6.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," to use analytical methods to determine the RCS pressure and temperature and Cold Overpressure Mitigation System (COMS) PORV limits that have been previously reviewed and approved by the NRC specified by date and revision number is revised to allow the use of previously reviewed and approved analytical methods that are specified by number and title only.

   AmerenUE has reviewed the TSTF 419 traveler and the model safety evaluation and has concluded that the justifications presented in TSTF-419 Revision 0 and the model SE prepared by the NRC staff for the Standard Technical Specifications are applicable to this proposed change.

3.0 BACKGROUND

NRC Generic Letter 96-06, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996, allows a licensee to relocate the pressure-temperature (P/T) limit curves from their plant Technical Specifications (TS) to a Pressure Temperature Limits Report (PTLR) or a similar document. The COMS limits were also allowed to be relocated to the same document. The methodology used to determine the P/T and COMS limit parameters must comply with the specific requirements of Appendices G and H to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR), be documented in an NRC approved topical report or in a plant specific submittal, and be incorporated by reference into the TS. Subsequent changes in the methodology must be approved by a license amendment.

Callaway Plant License Amendment 134, dated March 24, 2000 revised TS Section 5.6.6, “Reactor Coolant System (RCS) Pressure and Temperature Limits Report

(PTLR)". This amendment added criticality to the list of conditions for the reactor coolant system pressure-temperature (RCS P/T) limits, expanded the PTLR to include the analytical methods for the low temperature overpressure protection (LTOP) power operated relief valve (PORV) lift settings, and changed the list of documents that provide analytical methods for the RCS P/T limits and PORV lift settings in TS Section 5.6.6. Callaway specific terminology for LTOP is the cold overpressure mitigation system (COMS).

By letter dated May 23, 2002, the Westinghouse Owners Group (WOG) submitted Topical Report (TR) WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for NRC staff review and approval. This TR was developed to define a methodology for reactor pressure vessel (RPV) pressure-temperature (P/T) limit curve development and, consistent with the guidance provided in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," for the development of plant-specific Pressure-Temperature Limit Reports (PTLRs). A prior revision, WCAP-14040, Revision 2, had previously been approved as a PTLR methodology by the NRC staffs safety evaluation dated October 16, 1995. WCAP-14040, Revision 3, was submitted for NRC staff approval to reflect recent changes in the WOG methodology. Given the scope of the changes incorporated in WCAP-14040, Revision 3, and a significant amount of rewriting which was done to improve clarity of some sections, the NRC staff reviewed the TR in its entirety and published the NRC staff's safety evaluation dated February 27, 2004

The WOG subsequently republished this approved TR as WCAP-14040, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," incorporating the NRC staffs final safety evaluation and adding an Appendix B; "Correspondence with the NRC."

4.0 TECHNICAL ANALYSIS

The current definition of PTLR identifies the specifications in which the pressure and temperature limits are addressed. Specification 5.6.6.a requires that the individual specifications that address RCS pressure and temperature limits be referenced. The proposed changes to the definition will eliminate the duplication between the definition of PTLR and Section 5.6.6.a.

The revision to TS 5.6.6 to allow the Topical Reports to be identified by number and title will allow AmerenUE to use a current NRC-approved Topical Report to support limits in the PTLR without having to submit an amendment to the operating license each time the Topical Report is revised. The PTLR would provide the specific information identifying the particular approved Topical Report(s) used to determine the pressure-temperature (P/T) limits or COMS limits. This arrangement still provides the assurance that only the approved versions of the referenced Topical Reports will be used for the determination of the P/T limits or COMS limits since the complete citation will be provided in the PTLR.

The requirement to operate within the limits in the PTLR is specified in and controlled by the Technical Specifications. The figures, values, and parameters associated with the P/T limits and COMS setpoints are located in the PTLR and the methodology for their development must be reviewed and approved by the NRC. The proposed changes do not change the requirements associated with the review and approval of the methodology or the requirement to operate within the limits specified in the PTLR.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration Determination

AmerenUE has reviewed and concurs with the determination of whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10CFR50.92, Issuance of amendment, as discussed below:

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

   Response: No

   The proposed changes to reference the Topical Report number and title do not alter the use of the analytical methods used to determine the P/T limits or COMS setpoints 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 PTLR without having to submit an amendment to the operating license. Implementation of revisions to Topical Reports would still be reviewed and where required receive NRC review and approval. The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

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

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

   Response: No

   The proposed changes to reference the Topical Report number and title do not alter the use of the analytical methods used to determine the P/T limits or COMS setpoints 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 PTLR without having to submit 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 changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

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

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No

   The proposed changes to reference the Topical Report number and title do not alter the use of the analytical methods used to determine the P/T limits or COMS setpoints 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 PTLR without having to submit 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 proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The setpoints at which protective actions are initiated are not altered by the proposed changes. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. Therefore, it is concluded that this change does not involve a significant reduction in the margin of safety.

Based on the above evaluations, AmerenUE concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10CFR50.92(c) and, accordingly, a finding of no significant hazards consideration is justified.

5.2 Applicable Regulatory Requirements/Criteria

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The acceptability of a facility's proposed PTLR methodology is based on the NRC regulations and guidance as discussed below.

Appendix G to 10 CFR Part 50 requires that facility pressure temperature (P/T) limit curves for the reactor pressure vessel (RPV) be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

Appendix H to 10 CFR Part 50 establishes requirements related to facility RPV material surveillance programs.

Regulatory Guide 1.99, Revision 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation.

Generic Letter 92-01, Revision 1, requested that licensees submit their RPV data for their plants to the staff for review, and Generic Letter 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code.

The attributes of the vessel fluence methodology are described in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Regulatory Guide 1.190 is based on General Design Criteria (GDC) 14, 30 and 31 of Appendix A to 10 CFR Part 50. In this context, GDC-14 relates to an extremely low probability of leakage from the pressure coolant boundary; GDC-30 relates to the design of the reactor coolant boundary; and GDC-3 1 relates to material embrittlement and the effect of irradiation.

The review requirements for the COMS transients are described in SRP Section 5.2.2. SRP Section 5.2.2 is based on GDC-15 as it relates to the reactor coolant boundary design margin and GDC-3 1 as it relates to embrittlement and the effect of irradiation.

Generic Letter 96-03 addresses the technical information necessary for a licensee's implementation of a PTLR. Generic Letter 96-03 establishes the information which should be included in an acceptable PTLR methodology (which will be used to develop the PTLR), and the information which should be included within the PTLR itself. These information criteria are principally addressed in a table contained in Attachment 1 of Generic Letter 96-03 entitled, "Requirements for Methodology and PTLR," and are subdivided into seven technical elements which must be addressed by the licensee. Generic Letter 96-03 also addresses the appropriate modifications to the administrative controls section of a facility's TS which are necessary to implement a PTLR. TSTF-419 provides guidance on an alternative set of TS administrative control section changes which may be made to implement a PTLR in accordance with Attachment I of Generic Letter 96-03.

6.0 ENVIRONMENTAL CONSIDERATION

AmerenUE has evaluated the proposed change and has determined that the change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure.
Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), an environmental assessment of the proposed change is not required.

7.0. PRECEDENTS

This change is generally consistent with the changes to the Improved Technical Specifications described in TSTF-419, Revision 0 (Reference 8.1) and the NRC staff's model SE (Reference 8.2). Plants which have received approval for similar changes, in whole or in part, are listed below:

* Fort Calhoun (ADAMS Accession number ML032300305)
* Sequoyah (ADAMS Accession number ML042600465)
* Diablo Canyon (ADAMS Accession number ML041400243)

8.0 REFERENCES

8.1 Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler, TSTF-419, "Revise PTLR Definition and References in ISTS *5.6.6,* RCS PTLR," Revision 0 (ADAMS Accession number L012690234).

8.2 NRC letter of March 21, 2002 to NEI approving traveler TSTF-419 and providing the model safety evaluation (ADAMS Accession number ML020800488).

8.3 NRC letter of February 27, 2004 to NEI; "Final Safety Evaluation for Topical Report WCAP-14040, Revision 3, "Methodology Used To Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (ADAMS Accession number ML040620297).

ATTACHMENT II
ULNRC-05315

MARKUP TO
TECHNICAL SPECIFICATIONS

OL 1274
No Changes This Page

1.1 Definitions (continued)

MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14 of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and REPORT (PTLR) cooldown rates, the power operated relief valve (PORV) lift settings, and the Cold Overpressure Mitigation System (COMS) arming temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence

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OL1274

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)	period in accordance with Specification 5.6.6. ~~Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)."~~
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3565 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing all slave relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity

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OL 1274
No Changes
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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-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).

2. WCAP-10216-P-A, REV. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL AND FQ SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).

3. WCAP-10266-P-A, REV. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987 (W Proprietary).

4. NRC Safety Evaluation Reports dated July 1, 1991, "Acceptance for Referencing of Topical Report WCAP-12610 'VANTAGE + Fuel Assembly Reference Core Report' (TAC NO 77268)," and September 15, 1994, "Acceptance for Referencing of Topical Report WCAP-12610, Appendix B, Addendum 1, 'Extended Burnup Fuel Design Methodology and ZIRLO Fuel Performance Models' (TAC No. M86416)" (WCAP-12610-P-A).

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

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

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

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

(continued)

5.6 Reporting Requirements

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

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

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

b. The analytical methods used to determine the RCS pressure and temperature and COMS PORV limits shall be those previously reviewed and approved by the NRC, specifically those described in ~~the following documents:~~

~~1. NRC letter, CALLAWAY PLANT, UNIT 1 – ISSUANCE OF AMENDMENT RE. PRESSURE TEMPERATURE LIMITS REPORT (TAC NOS. MA5631 and MA7287), dated March 24, 2000.~~

~~2.~~ WCAP-14040-NP-A, ~~Revision 2,~~ "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves~~, January, 1996~~".

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

5.6.7 Not used.

5.6.8 PAM Report

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

5.6.9 Not used.

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ATTACHMENT III
ULNRC-05315

RE-TYPED
TECHNICAL SPECIFICATION PAGES

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)	period in accordance with Specification 5.6.6.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3565 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing all slave relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity

(continued)

5.6 Reporting Requirements

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

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

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

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

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

5.6.7 Not used.

5.6.8 PAM Report

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

5.6.9 Not used.

5.6.10 Steam Generator Tube Inspection Report

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

a. The scope of inspections performed on each SG;

b. Active degradation mechanisms found;

c. Nondestructive examination techniques utilized for each degradation mechanism;

(continued)

5.6 Reporting Requirements (continued)

5.6.10 Steam Generator Tube Inspection Report (continued)

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

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

f. Total number and percentage of tubes plugged to date; and

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

ATTACHMENT IV
ULNRC-05315

COPY OF PROPOSED
TECHNICAL SPECIFICAQTION
BASES CHANGE

BASES

ACTIONS

C.1 and C.2 (continued)

the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. WCAP-14040-NP-~~A, Rev. 2, January 1996~~
2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

(continued)