Entergy

Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72801
Tel 501-858-4888

Craig Anderson
Vice President
Operations ANO

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2CAN120001

U. S. Nuclear Regulatory Commission
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Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Application for License Amendment to Increase Authorized Power Level

Gentlemen:

Attached for your review and approval is an application for an amendment to Facility Operating License NPF-6 for Arkansas Nuclear One, Unit 2 (ANO-2). This license amendment application revises the operating license and technical specifications to authorize operation of the plant at a Nuclear Steam Supply System (NSSS) power level up to 3026 megawatts thermal (MWt). The proposed change represents a 7.5% increase above the currently licensed core power rating of 2815 MWt. The proposed change would increase the unit's design gross electrical output from approximately 958 megawatts electrical (MWe) to approximately 1048 MWe. This application is submitted in support of the ANO-2 project commonly referred to as Power Uprate.

Enclosure 1 contains the revised pages for the ANO-2 Operating License and Technical Specifications and associated Bases. Marked up pages are provided for information only in Enclosure 2. Enclosures 3 and 4 contain revised Core Operating Limits Report and Technical Requirement Manual pages, respectively, for information only. These pages are provided for completeness and to aid the NRC staff in their review. Enclosure 5 is the ANO-2 Power Uprate Licensing Report. It contains a summary of the analyses supporting the requested increase in authorized power level. These analyses include a review of the limiting facility systems, components and safety analyses under both loss of coolant accident (LOCA) and non-LOCA conditions at a power level of 3026 MWt over a complete range of operating temperature and pressure conditions, and establish that the facility is capable of operating safely at the requested power level.

The ANO-2 Power Uprate analyses and evaluations were completed using the following guidelines: 1) Westinghouse topical Report WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," January 1983, 2) GE Topical Report NEDC-31897P-A, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," May 1992 [approved by the NRC staff on February 8, 1996], and

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3) SECY-97-042, Section 3, "Power Uprate Review Process." Additional insights were derived from the NRC letter dated February 8, 1996, "Staff Position Concerning General Electric Boiling-Water Reactor Extended Power Uprate Program." Based on a review of the above documents, as well as several previous power uprate submittals, the balance of plant (BOP) information submitted in this letter has been minimized since NRC review and approval is not required. While extensive reviews of BOP systems have been completed, this submittal focuses only on those BOP systems that interface with NSSS systems. The remaining BOP analyses and evaluations are evaluated in accordance with the requirements of 10CFR50.59 and are on file at ANO.

To accommodate operation at the uprated power level, several plant modifications were necessary. These modifications are discussed in the attachment. 10CFR50.92 requires in part, that a construction permit be issued before the issuance of an amendment to the license if the application for that amendment involves the material alteration of a licensed facility. Based on past determinations regarding similar plant modifications, the modifications were determined not to be material because they do not change the plant operations or purpose as originally licensed. Additionally, although not a modification, a significant amount of work was required to reanalyze the ANO-2 Containment Building in order to increase its design pressure to 59 psig. This was necessary due to the increased peak accident pressure that resulted primarily from the larger water volume in the replacement steam generators which were increased in size to accommodate the uprate in power. The NRC staff approved the increase in the ANO-2 Containment Building design pressure in a letter dated November 13, 2000 (2CNA110002).

As a result of the new uprated power, accident analyses were either reanalyzed, not reanalyzed or not applicable. Of those events that were reanalyzed, four resulted in dose increases above that currently analyzed. Dose consequences from other events were not changed from those reported to the NRC previously. The details may be found in Section 7.0 of the ANO-2 Power Uprate Licensing Report (Enclosure 5). Therefore, as a result of the new analyses performed for Power Uprate, NRC review is requested for the Steam Generator Tube Rupture, Fuel Handling Accident and Control Element Assembly Ejection events. Review is requested in accordance with the requirements of 10CFR50.90, "Application for amendment of license or construction permit," per the requirements of the revised 10CFR50.59 Rule, "Changes, tests and experiments," dated October 4, 2000. The dose consequences for the fourth reanalyzed event, large break loss of coolant accident, increased by less than 10% of the remaining difference to the 10CFR100 limit and is not subject to NRC review as a license amendment in accordance with 10CFR50.90 pursuant to 10CFR50.59.

As stated in correspondence dated September 29, 1999 (2CAN099902) and supplemented by letter dated May 12, 2000 (2CAN050001), ANO utilized a revised Appendix K Evaluation Model for the large break loss-of-coolant-accident analysis for power uprate. The topical report, CENPD-132, Supplement 4-P, "Calculative Methods for the ABB CENP Large Break LOCA Evaluation Model," was approved by the NRC on December 15, 2000.

The proposed changes have been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that these changes involve no significant hazards considerations. The bases for these determinations are included in the attachment to this letter.

Additionally, pursuant to 10CFR51, Section 51.41, "Requirement to submit environmental information," additional information concerning the environmental effects of the ANO-2 Power Uprate is provided. The information provided in section 10 of Enclosure 5 demonstrates no individual or cumulative adverse effect upon the human environment. This information is provided to aid the NRC in complying with section 102(2) of the National Environmental Policy Act.

Entergy requests approval of this amendment by March 15, 2002, with an effective date prior to the commencement of heatup from refueling outage 2R15. The outage is currently scheduled to begin in March 2002. Although this request is neither exigent nor emergency, your prompt review is requested. Entergy realizes this submittal has a large scope and is available to meet with the staff to respond to NRC questions or concerns.

Should you have any questions or comments, please contact me. I declare under penalty of perjury that the foregoing is true and correct. Executed on December 19, 2000.

Very truly yours,

CGA/dwb
Attachments

cc: Mr. Ellis W. Merschoff
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

NRC Senior Resident Inspector
Arkansas Nuclear One
P.O. Box 310
London, AR 72847

Mr. Thomas W. Alexion
NRR Project Manager Region IV/ANO-2
U. S. Nuclear Regulatory Commission
NRR Mail Stop 04-D-03
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Mr. David D. Snellings
Director, Division of Radiation
Control and Emergency Management
Arkansas Department of Health
4815 West Markham Street
Little Rock, AR 72205

ATTACHMENT

TO

2CAN120001

PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT TWO

DOCKET NO. 50-368

DESCRIPTION OF PROPOSED CHANGES

The proposed changes to the Arkansas Nuclear One, Unit 2 (ANO-2) Technical Specifications (TS) are required to maintain consistency with the transient and accident analyses which evaluated the impact of the power uprate planned for Cycle 16 in combination with the replacement steam generators that were installed for Cycle 15 operation. The following changes are proposed:

Entergy Operations, Inc. is proposing that the Arkansas Nuclear One, Unit 2 (ANO-2) Operating License be amended to revise affected technical specification limits associated with power uprate. This license application, if approved, would revise the operating license and technical specifications to authorize operation of ANO-2 at a nuclear steam supply system power level up to 3026 megawatts thermal. The proposed change represents a 7.5% increase above the currently licensed core power rating of 2815 megawatts thermal. The new technical specification limits ensure safety margins remain acceptable during subsequent operations at uprated conditions.

The analyses supporting the uprate result in a change to the Operating License and several changes to the Technical Specifications/Bases. Certain Core Operating Limits Report (COLR) and Technical Requirements Manual (TRM) pages will be affected by the increase in power. These changes will be implemented in accordance with the requirements of 10CFR50.59 following the NRC staff's approval of this license amendment. However, since the changes are due to the power uprate, the affected COLR and TRM pages are included in this submittal for completeness.

Some improvements are made to the TSs and COLR documents, including minor format changes and grammar corrections, that are not related to power uprate. The power uprate was viewed as an appropriate opportunity to make these improvements. The following is a page-by-page list of changes. Minor changes in formatting are not discussed within the contents of this submittal. Improvements not related to power uprate are denoted in *italics*.

Operating License Change

- Page 3, Section 2.C.(1) - Change the maximum authorized reactor core power level from 2815 megawatts thermal to 3026 megawatts thermal.

Technical Specification Changes

- Page 1-1; Definition 1.3 - Change the definition of RATED THERMAL POWER from 2815 mwt to 3026 MWt. Note that "mwt" is changed to "MWt."

- Page 2-5; Table 2.2-1 - In the "Reactor Protective Instrumentation Trip Setpoints Limits" table, decrease the pressurizer pressure – low setpoint from ≥ 1675 psia to ≥ 1650 psia and its allowable value from ≥ 1643.9 psia to ≥ 1618.9 psia.

- Pages 3/4 1-17, 3/4 1-18 and 3/4 1-19; TS 3.1.3.1 - Clarify control element assembly (CEA) position requirements.

- *Page 3/4 1-25 and 3/4 1-26; TS 3.1.3.6 - Clarify the CEA position requirements in regard to the Regulating and Group P insertion limits.*

- *Pages 3/4 3-5b, and 3/4 3-5c; TS Table 3.3-1 - Clarify the CEA position requirements in the reactor protective instrumentation table.*

- Pages 3/4 3-16 and 3/4 3-17; Table 3.3-4 - Decrease the pressurizer pressure – low setpoint of item 1c and 5c of Table 3.3-4, "Engineered Safety Feature Actuation System Instrumentation Trip Values," from ≥ 1675 psia to ≥ 1650 psia and its allowable value from ≥ 1643.9 psia to ≥ 1618.9 psia. Additionally, Table 3.3-4 is revised to eliminate redundant requirements for both volume and indicated level, leaving indicated level as the controlling requirement.

- Page 3/4 5-7; TS 3.5.4 - Replace the refueling water tank indicated level requirements and corresponding volumes with a required minimum and maximum available volume of water which is used in the safety analyses.

- Page 3/4 7-2; TS Table 3.7-1 - "Maximum Allowable Linear Power Level-High Trip Setpoint With Inoperable Steam Line Safety Valves During Operation With Both Steam Generators," has been changed to decrease the allowable values for the linear power level – high trip setpoint during operation with one or more main steam safety valves (MSSVs) inoperable.

- Page 3/4 7-3; TS Figure 3.7-1 - "MTC Versus Maximum High Linear Power Level and Trip Setpoint," has been changed and retitled to decrease the maximum value of the linear power level – high trip setpoint allowed for a given moderator temperature coefficient (MTC) during periods when a MSSV(s) is inoperable.

- Page 6-21a; TS 6.9.5.1 - Item 17 was added to the list of analytical methods used to determine the Core Operating Limits Report. Item 17 reflects the change in emergency core cooling system (ECCS) evaluation methods being applied to ANO-2.

Technical Specification Bases Changes

- *Page 3/4 1-3 and 3/4 1-4; Bases Section 3/4.1.3 - Clarify CEA position requirements.*

- Page B 3/4 2-1; Bases Section 3/4.2.1 - Revised discussion associated with maintaining fuel pins within design limits.

- Page B 3/4 5-3; Bases Section 3/4.5.4 - Changes the basis consistent with the change to the limiting condition for operation which redefines the refueling water tank volume limits in terms of the minimum and maximum available volume assumptions used in the accident analyses.

- Page B 3/4 7-1; Bases Section 3/4.7.1.1 - This section is modified to more clearly link the total main steam safety valve capacity requirement to the loss of condenser vacuum transient analysis.

The following sections will provide information and bases for the changes listed above. Five enclosures are included in this submittal:

Enclosure 1: Revised Operating License and Technical Specification/Bases Pages

Enclosure 2: Marked up Operating License and Technical Specification/Bases Pages (for information only)

Enclosure 3: Revised COLR Pages (for information only)

Enclosure 4: Revised TRM Pages (for information only)

Enclosure 5: ANO-2 Power Uprate Licensing Report

BACKGROUND

The proposed power uprate consists of a number of changes that will permit power operation up to 3026 MWt for Arkansas Nuclear One, Unit 2. This represents the first power uprate on ANO-2 since original issuance of the operating license. This power level is 7.5% above the current maximum rated thermal power of 2815 MWt. A 7.5% uprate was selected based on several factors. Since the unit's steam generators required replacement due to various corrosion related phenomena that had occurred over the years of operation, an economic decision was made to design the replacement steam generators to accommodate an increase in rated thermal power. It was determined, based on economics, that a power uprate of at least 6.5% would be required in order to recover the capital investment of larger replacement steam generators and other modifications that would be necessitated by the installation of larger steam generators. Scoping efforts were then initiated to explore whether an uprate greater than 6.5% was possible based on five criteria:

- satisfactory safety analysis results,
- satisfactory margins on all safety-related systems, structures and components,
- satisfactory margins for reactor vessel head Alloy 600 nozzles,
- acceptable additional cost above the cost to achieve a 6.5% uprate, and
- the ability of the replacement steam generators to support a higher uprate.

Based on the above criteria and the physical limitations of the replacement steam generators (i.e., height and interface requirements), a 7.5% uprate was determined to be the optimum level. The replacement steam generators accommodate a power level of 107.5% while anything less would limit the possibility of maximizing MWe production.

To accommodate operation at the uprated power level several plant modifications were necessary. Changes in MWe output can vary not only with licensed reactor power but also with steam generator condition and turbine and cycle performance. ANO-2 had experienced degraded steam generator steam pressure and reduced MWe generation due to plugged steam generator tubes as a result of corrosion-related phenomena that have occurred over the years of operation. Replacement of the steam generators, turbine modifications and power uprate all contribute to regaining and improving the MWe output of ANO-2. These, and other, modifications are discussed further in Enclosure 5.

Additionally, although not a modification, a significant amount of work was required to reanalyze the ANO-2 Containment Building in order to increase its design pressure from 54 psig to 59 psig. This was necessary due to the increased peak accident pressure that resulted primarily from the larger water volume in the replacement steam generators. The NRC granted the design pressure increase in License Amendment #225 dated November 13, 2000 (2CNA110002). License Amendment #225 was granted in response to our application dated November 3, 1999 (2CAN119903), as supplemented by letters dated April 4 (2CAN040004), June 9 (2CAN060007), June 29 (2CAN060014), August 2 (2CAN080005), and August 16, 2000 (2CAN080010). This license amendment request will not reproduce the information contained in those submittals but will make reference to them as appropriate.

Similarly, the NRC granted License Amendment #222 dated September 29, 2000 (2CNA090002), in response to our request to revise the license and technical specifications to maintain consistency with the transient and accident analyses which evaluated the impact of the replacement steam generators beginning with Cycle 15 operation. This license amendment was granted in response to our application dated November 29, 1999 (2CAN119901), as supplemented by letters dated January 26 (2CAN010008), May 17 (2CAN050005 and 2CAN050006), May 31 (2CAN050009) and August 4, 2000 (2CAN080004). License Amendment #222 incorporates a new methodology employed in calculating radiological doses for some non-loss-of-coolant accident events. Also, TS changes were made to the reactor protection system (RPS) and engineered safety features actuation system (ESFAS) low pressurizer pressure setpoints, the RPS and ESFAS low steam generator pressure setpoints, the RPS and ESFAS low steam generator level setpoints, the reactor coolant system flow rate limit, and the high linear power trip setpoints with inoperable main steam safety valves (MSSVs). Many of these analyses were performed at the power uprated conditions. The power uprate license amendment will not reproduce the information contained in those submittals but will make reference to them as appropriate. The uprate analyses/evaluations were performed in accordance with the current ANO-2 licensing bases except where noted.

Also, ANO utilized a revised Appendix K Evaluation Model for the large break loss of coolant accident (LOCA) analysis for power uprate. Our letters dated September 29, 1999 (2CAN099902) and May 12, 2000 (2CAN050001), outlined our plans for utilizing Topical Report CENPD-132, Supplement 4-P, "Calculative Methods for the ABB CENP Large Break LOCA Evaluation Model." CE Nuclear Power, LLC (formerly ABB-CE), a division of Westinghouse Electric Company, LLC, submitted this topical report to the NRC for review and approval on April 30, 1999. NRC approval was received on December 15, 2000.

The analyses and evaluations supporting the requested 7.5% power uprate were performed by CE Nuclear Power, LLC (formerly ABB-CE) and Entergy Operations, Inc. Westinghouse Topical Report WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," (January 1983) was used as a guideline in preparation of the ANO-2 Power Uprate Licensing Report. Since its submittal to the NRC, the methodology has been used successfully as a basis for power uprate projects on over twenty pressurized water reactor (PWR) units, including Vogtle Units 1 and 2, Turkey Point Units 3 and 4, and Farley Units 1 and 2.

Additional guidance regarding the scope and content of an acceptable power uprate submittal was obtained from Licensing Topical Report NEDC-31897P-A, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," (May 1992). Although this topical report was written specifically for boiling water reactors, it contains useful information for PWR power uprate reports as well. SECY-97-042 (February 18, 1997), Section 3, "Power Uprate Review Process," cited both of these topical reports as documents the NRC should use to establish the basis for NRC review of power uprate submittals. Additional insights were derived from the NRC letter dated February 8, 1996, "Staff Position Concerning General Electric Boiling-Water Reactor Extended Power Uprate Program."

DISCUSSION OF CHANGE

To ensure protection of the health and safety of the public following power uprate to 3026 MWt, the ANO-2 design basis and accident analyses were reviewed with respect to associated changes to the plant operating conditions and characteristics. These reviews are described in Enclosure 5, and specifically support the proposed changes to the Operating License, TSs, TRM and COLR.

In particular, Section 7.1 of Enclosure 5 presents a summary of the emergency core cooling system (ECCS) performance analysis that demonstrates conformance to the ECCS acceptance criteria for light water nuclear power reactors, 10CFR50.46, for ANO-2 at uprate conditions. Analyses were performed for a spectrum of large break loss-of-coolant accidents (LOCAs) and for small break LOCAs. In addition, an evaluation of post-LOCA long term cooling was performed.

Section 7.3 of Enclosure 5 presents a summary of the ANO-2 SAR Chapter 15 event analyses that demonstrates conformance with the applicable acceptance criteria at power uprate conditions. A listing of these scenarios may be found in Table 7.3.0-1 of Enclosure 5 of this submittal. The table identifies how each event was evaluated in relation to power uprate. Section 7.3 contains information discussing the methods used in deciding whether the SAR events were reanalyzed, not reanalyzed or not applicable.

As discussed in Enclosure 5, the safety analyses and design reviews demonstrate that the acceptance criteria are met. However, as identified in the loss of condenser vacuum (LOCV) analysis presented in Section 7.3 of Enclosure 5, the maximum allowable operating power levels must be restricted per the proposed TS change to assure pressures remain below 110% of steam generator design pressure when one or more MSSVs are inoperable.

The revised technical specifications ensure the safety limits are maintained during subsequent operations at uprated conditions. The analyses supporting the uprate result in one change to the license and several changes to the TSs, TRM and COLR. The following changes are proposed.

Change to Operating License

Maximum Power Level - Page 3, Section 2.C.(1)

The definition of the maximum power level to which Entergy Operations, Inc. (EOI) is licensed to operate is increased from 2815 megawatts thermal to 3026 megawatts thermal. The Power Uprate

Licensing Report provided in Enclosure 5 documents the acceptability for operating at the increased power level.

Changes to Technical Specifications

Rated Power - Definition 1.3 on page 1-1

The TS definition of rated thermal power is increased from 2815 MWt to 3026 MWt. Additionally, "mwt" is changed to "MWt." MWt is the standard abbreviation for megawatts thermal. The justification for increasing the thermal power rating is provided in Enclosure 5.

Low Pressurizer Pressure Setpoint - Table 2.2-1 on page 2-5 and Table 3.3-4 on pages 3/4 3-16 and 3/4 3-17

The Low Pressurizer Pressure RPS and ESFAS setpoints were reduced from ≥ 1675 psia to ≥ 1650 psia, with the allowable values reduced from ≥ 1643.9 psia to ≥ 1618.9 psia. The reduction in setpoint is necessary due to the larger pressurizer pressure decrease caused by the larger hot full power to hot zero power temperature swing associated with power uprate, resulting in increased shrink phenomena and reduced reactor coolant system (RCS) pressures following a reactor trip. This setpoint is projected to be sufficiently reduced to prevent unnecessary ESFAS actuations following normal plant transients, and sufficiently above those values used in the analyses performed in support of power uprate. This setpoint change was supported by the use of a minimum setpoint of 1400 psia for the low pressurizer pressure safety injection actuation signal (SIAS) in the LOCA and steam line break analyses.

Additionally, Table 3.3-4 is revised to eliminate redundant requirements for both refueling water tank volume and indicated level, leaving indicated level as the controlling requirement. In this instance indicated level, rather than volume, is appropriate because only the indicated level has any significance for calibration of the setpoint and surveillance monitoring of the setpoint.

High Linear Power Trip Setpoint with MSSVs Inoperable - Table 3.7-1 on page 3/4 7-2, Figure 3.7-1 on page 3/4 7-3 and associated Bases section 3/4.7.1.1 on page B 3/4 7-1

The loss of condenser vacuum (LOCV) analysis provided in Section 7.3.6 of Enclosure 5 determined that a reduction in the High Linear Power Trip setpoint, below the current TS requirement, is appropriate when MSSVs are inoperable. As a result, the setpoints specified in TS Table 3.7-1 have been reduced in cases where one MSSV is inoperable on one or both steam headers, and TS Figure 3.7-1 is modified accordingly. Additionally, the headers on Figure 3.7-1 were also changed. The x axis header was changed from "Percent Power" to "High Linear Power Level Trip Setpoint (%)." The y axis header was revised from "MTC (x 10E-04)" to "MTC (1E-04 $\Delta k/k/^{\circ}F$)." The figure has been retitled, "Maximum High Linear Power Level And Trip Setpoint Versus MTC." These changes more accurately describe the information displayed on the graph.

Figure 3.7-1 provides a graph of moderator temperature coefficient (MTC) versus the setpoint for the Linear Power Level – High Trip. The figure provides guidance for configurations with up to one MSSV per steam header being inoperable.

The lower limit of the plot on Figure 3.7-1 was deleted to accommodate the shift in the end-of-life MTC since the maximum end-of-life MTC for Cycle 16 could analytically reach a value more negative than -3.5 Δk/k/°F. MTC values more negative than –2.5 Δk/k/°F restrict operation to ≤ 87% rated thermal power with one MSSV inoperable on each main steam header. MTC values more negative than –2.1 Δk/k/°F restrict operation to ≤ 91% rated thermal power with a total of one MSSV inoperable. Since the maximum permissible power remains at the previously mentioned values as MTC becomes more negative, a lower limit is not necessary. Therefore, the aforementioned lower limit deletion does not alter the previous restrictions in power for the inoperable MSSV cases.

The application of MTC versus rated thermal power Figure 3.7-1 has been found acceptable for conditions where not more than one MSSV is inoperable per steam header. The proposed changes associated with the High Linear Power Trip setpoints relating to cases where MSSVs are inoperable are acceptable.

The Bases to Technical Specification 3/4.7.1.1 have been modified to more clearly link the total MSSV capacity requirement to the loss of condenser vacuum transient analysis. The comparison of valve capacity to full power steam flow has been deleted. This comparison is not a requirement of the ASME Code and can lead to misinterpretation since steam conditions at full power operation are different from those at which valve capacity is normally defined. According to the ASME Code, a safety valve must be sized large enough to prevent overpressurization of its associated piping. This capacity is not necessarily equal to the full power steam flow.

<u>CEA Position Requirements - TS 3.1.3.1 on pages 3/4 1-17 through 3/4 1-19, TS 3.1.3.6 on pages 3/4 1-25 and 3/4 1-26, Table 3.3-1 on pages 3/4 3-5b and 3/4 3-5c, and associated Bases section 3/4.1.3 on page B 3/4 1-3 and B 3/4 1-4</u>

The following changes to TS 3.1.3.1 modify / clarify CEA position requirements to assure the plant continues to operate with power distributions and CEA positions that are consistent with the safety analyses. While the changes to the CEA TSs at first appear to be substantial, the majority of the changes simply provide clarification or wording/format improvements. The remaining revisions implement conservative changes and implement operational enhancements that are supported by the power uprate analysis.

- Current ACTION (a) wording in regard to the requirements for HOT STANDBY is clarified to state "within the next 6 hours" instead of "within 6 hours."

- Current ACTIONS (b) and (c) have been consolidated into a new ACTION (b) by directly referring to insertion limits imposed by TS 3.1.3.5 and 3.1.3.6, rather than restating some of the requirements of those two specifications within this specification.

- Current ACTION (d) has been modified and relabeled ACTION (c) to clarify that multiple CEAs that are trippable but within their specified alignment requirements must be restored within 72 hours. This clarification differentiates between multiple CEA inoperability requirements from those of new ACTION (d) regarding CEAs that are misaligned by more than 7 inches.

- Current ACTIONS (e) and (f) have been modified and relabeled ACTION (d), providing limits to single CEA misalignments (versus multiple misalignments allowed by the current TS), and clarifies that the time period for specified ACTIONS associated with misalignments shall be measured from the time of misalignment. The use of the COLR limits associated with required power reductions have been clarified as necessary for inward deviations only, consistent with the intent of the NRC's safety evaluation in License Amendment No. 70 dated November 12, 1985. The time provided for CEA misalignment is increased to 2 hours consistent with the COLR figure limit and supported by the safety analysis and NUREG-1432, Revision 1, "Standard Technical Specifications Combustion Engineering Plants." The reference to an upper deviation limit of 19 inches has been deleted since actions are the same as those for CEA deviations of 7 inches. The requirements of this action are more restrictive than those of new ACTION (b); therefore, the redundant reference to CEA operability is deleted. If the CEA or its associated group is repositioned to recover alignment within 7 inches of the remainder of the group and the CEA is known to be inoperable, new ACTION (b) would then be applied. Reference to TS 3.1.3.5 has been added as a human factors enhancement. Reference to TS 3.1.3.6 is deleted from ACTION (d.2.a) since it is redundant to that found in ACTION (d.2).

- Current ACTION (g) has been modified and relabeled Action (e) to ensure a plant shutdown if more than one CEA is misaligned by more than 7 inches. The 7-inch lower limit is more restrictive than the present 19-inch action limit. The phrase "trippable" is removed from this action since a plant shutdown is required regardless of the trippable status of the CEA when more than one CEA is misaligned by >7 inches.

TS 3.1.3.6 is restructured to:

- Add CEA insertion limits that apply when both CEACs are inoperable. These restrictions were imposed by ACTION 6 of the current TS Table 3.3-1, but not directly addressed in TS 3.1.3.6.

- Add ACTION a.2.a.2 (i.e., Hot Standby in 8 hours) for the situation when insertion limits are violated with at least one CEAC operable. To prevent unnecessarily entering TS 3.0.3, this action was added to provide guidance to the operators if the CEAs are not restored within their limits.

- Add ACTION a.2.b (i.e., Hot Standby in 8 hours) for the situation when insertion limits are violated with both CEACs inoperable. To prevent unnecessarily entering TS 3.0.3, this action was added to provide guidance to the operators if the CEAs are not restored within their limits.

- Clarify that the time period for specified ACTIONS shall be measured from the time the insertion limit was exceeded for ACTION (a), and from the time the interval limits are exceeded for ACTION (b).

Correspondingly, Table 3.3-1, ACTION 6.b is modified to add references to TS 3.1.3.5 and 3.1.3.6.b, rather than specify the insertion limits applicable when both CEACs are inoperable and to clarify actions during surveillance testing.

The Bases for 3/4.1.3 (Pages B 3/4 1-3 and B 3/4 1-4) are modified to clarify that continued/restricted operation is only allowed for a single CEA, clarify that a core protection calculator penalty applies only to outward deviations, remove the 19-inch misalignment restriction, change the time to realign a CEA from one hour to two hours and remove a redundant portion of the second paragraph which is already stated in the paragraph that succeeds it.

Minimum Boric Acid Concentration and Elimination of Volume Requirement - TS 3.5.4 on page 3/4 5-7, section 3/4.5.4 on page B 3/4 5-3 and associated TRM changes

The normal amount of boric acid stored solution contained in the combined volumes of the boric acid makeup tanks and the refueling water tank is sufficient to bring the plant to a cold shutdown condition at any time during plant life. The cooldown without letdown analysis determines the minimum boric acid storage requirements for the boric acid makeup tank by demonstrating that the minimum required shutdown margin can be maintained for a slow cooldown on natural circulation from hot standby to cold shutdown conditions at end of core life. The boric acid makeup tank and refueling water tank must be able to increase the RCS boron concentration to offset the positive reactivity insertion from the cooldown and from the decay of xenon. The increase in boron concentration must be accomplished without the benefit of letdown from the RCS. Letdown facilitates boron concentration control via feed and bleed.

The changes in the core design to support the power uprate change the rate of positive reactivity insertion during the analyzed transient. The positive reactivity insertion rate representative of the most restrictive core design anticipated for the next few cycles has increased slightly. This increase results in a tightening of the range of boric acid makeup tank boric acid concentrations that will produce acceptable rates of boron concentration increase to offset the positive reactivity insertion.

The lower limit of boric acid concentration has thus been increased from 2.5% to 3.0% for Cycle 16, based on a revised cooldown without letdown analysis. The revised analysis also determined new minimum boric acid makeup tank volumes. These new requirements will be incorporated into TS Figure 3.1-1. In conjunction with the cooldown without letdown analysis, a cooldown analysis to determine boron storage requirements for modes 5 and 6 was also completed. This would establish new boric acid makeup tank volume limits in Technical Specification 3.1.2.7. The minimum indicated level would increase from 31% to approximately 36%. A request to relocate all of TS Section 3/4.1.2, "Boration Systems" including TS Figure 3.1-1 and TS 3.1.2.7 to the ANO-2 Technical Requirements Manual (TRM) has been submitted to the NRC in a letter dated November 30, 2000 (2CAN110002). Consistent with the discussion above, the following changes will be reflected in the TRM pages following NRC approval to relocate the TS pages (see Enclosure 4):

- TS Page 3/4 1-13, TS 3.1.2.7 - the minimum indicated boric acid makeup tank level will be increased to 36% from 31% and the boric acid concentration range of between 2.5 and 3.5 WT% will be changed to 3.0 and 3.5 WT%. Also, the specification will be revised to eliminate redundant requirements for both volume and indicated tank level, redefining the limits in terms of the indicated level as the controlling requirement.

- TS Page 3/4 1-14, Figure 3.1-1 - will be revised to increase the lower limit of boric acid concentration from 2.5% to 3.0% and to incorporate new minimum boric acid makeup tank volumes based on a revised cooldown without letdown analysis. The TS figure currently shows a dual y-axis legend and scale. Following its relocation to the TRM, the y-axis legend and scale will be modified to show only indicated percent level for available volume. The scale values will be design verified in accordance with the ANO 10CFR50.59 program.

- TS Page 3/4 1-15 and 3/4 1-16, TS 3.1.2.8 - consistent with TS 3.1.2.7, the refueling water tank specification will be revised to eliminate redundant requirements for both volume and indicated tank level, redefining the limits in terms of the indicated tank level as the controlling requirement.

- TS Bases Pages B 3/4 1-2 and 3/4 1-3; Bases Section 3/4.1.2 - Revised leaving the available volume as the controlling requirement and eliminate the redundant volume and level requirement. Specific limits will be removed since this level of detail is not normally contained within the bases.

Specification 3.5.4 changes the allowable range for the borated water volume in the refueling water tank from between 464,900 and 500,500 gallons (91.7% to 100% indicated level) to a required range of available volumes between 384,000 and 503,300 gallons. The new range no longer applies directly to water contained in the tank; rather, the new range represents the minimum and maximum volume of water available to be transferred from the refueling water tank into containment via the ECCS and containment spray before the pump suction is transferred to the containment sump by the recirculation actuation signal. The values for the new range are based on the existing refueling water tank maximum and minimum volume/level limits and the recirculation actuation signal setpoint. The new minimum and maximum available volume limits are the values assumed in those accident analyses that are dependent on post LOCA containment sump level and/or the time required to reach the recirculation actuation signal setpoint; for example, sump pH calculations, pump net positive suction head, equipment submergence and the containment pressure/temperature analyses. The operating limits used to demonstrate compliance with this specification, in indicated refueling water tank level, will be implemented in procedures and will be the same as those in the current technical specification, i.e., 91.7% to 100%.

Additionally, Enclosure 1 contains one TS page (TS 3.5.4 on page 3/4 5-7) and one Bases page (section 3/4.5.4 on page B 3/4 5-3) that are being modified to eliminate redundant requirements for both the refueling water tank volume and indicated tank level, redefining the limits in terms of the available volume as the controlling requirement. This change is proposed because it eliminates potential confusion that can result from having a specification based on two potentially conflicting requirements.

Analytical Methods Used to Determine the Core Operating Limits - TS 6.9.5.1 on page 6-21a

TS 6.9.5.1 which lists the analytical methods used to determine the Core Operating Limits is updated to reflect the change in ECCS evaluation methods being applied to ANO-2. The topical report was approved by the NRC on December 15, 2000. Following re-issuance of CENPD-132-P, Supplement 4-P as an approved document, this reference will be updated to reflect the approved status and corresponding re-issuance date.

Changes to Core Operating Limits Report (COLR)

Section IV.6, linear heat rate, Figure 1, and Figure 2 of the COLR will be revised as a result of power uprate. These changes will be implemented in accordance with the requirements of 10CFR50.59 following the NRC staff's approval of this license amendment. However, since the changes are due to the power uprate, the affected COLR pages are discussed below and included in Enclosure 3 for completeness. Power uprate is planned for implementation at the beginning of Cycle 16; therefore, these changes, along with any others necessary to support Cycle 16, will be made at that time.

- Page 6; Item 6 – Increased the linear heat rate limit from 13.5 kW/ft to 13.7 kW/ft for burnups up to 200 effective full power days (EFPD), and restricted to 13.0 kW/ft for higher burnups.

- Page 8; Figure 1 – Modified to decrease the lower limit for moderator temperature coefficient from -3.4×10^{-4} Δk/k/°F to -3.8×10^{-4} Δk/k/°F.

- Page 9; Figure 2 – Increased from 1 hour to 2 hours the time allowed to complete a core power reduction following misalignment of one or more CEAs.

Linear Heat Rate

An increase in the linear heat rate limit is desirable to maintain adequate margin to the limit during plant operation under the new power uprate conditions; especially at beginning-of-cycle when radial power peaks are highest. The LOCA and non-LOCA accident analysis presented in Sections 7.1 and 7.3 of Enclosure 5 explicitly account for increasing the linear heat rate limit to 13.7 kW/ft. This limit, in combination with restricting burnups above 200 EFPD to 13.0 kW/ft, assures maximum rod internal pressure does not exceed acceptable levels as described in Section 8.3.1.5. [Note: The fuel rod pressure calculation described in Section 8.3.1.5 actually assumes that the linear heat rate limit will be reduced to 13.0 kW/ft before any rod exceeds an average burnup of 50 gigawatt days per metric ton of uranium (GWD/MTU). Limiting cycle burnup to 200 EFPD assures that no rod will exceed 50 GWD/MTU prior to the step-down.]

The Bases for TS 3/4.2.1 (page B 3/4 2-1) are modified to identify that the linear heat rate limit ensures the maximum fuel rod pressures do not exceed acceptable levels, as well as ensuring the LOCA peak cladding temperatures do not exceed 2200°F.

Moderator Temperature Coefficient (MTC)

A new negative MTC limit is required to accommodate the more negative end-of-cycle MTC that will be characteristic of cores designed for power uprate conditions and the cycle energy requirements. As identified in Section 7.3.0.2, the accident analyses explicitly covered changing the lower limit for MTC from -3.4×10^{-4} Δk/k/°F to -3.8×10^{-4} Δk/k/°F.

CEA Position

The increased time allowed for completing a core power reduction following CEA misalignment and reduced power rampdown rate (from 20% in one hour to 28% in two hours) allows better operator control of the ramp and reduces the risk of a reactor trip. The increased time allowed for completing the core power reduction has been factored into the CEA misoperation analysis described in Section 7.3.3 of Enclosure 5. This analysis establishes the departure from nucleate boiling ratio (DNBR) margin that must be reserved by the core operating limiting supervisory system (COLSS), or core protection calculators (CPCs) when COLSS is inoperable such that the DNB specified acceptable fuel design limit (SAFDL) is not violated during CEA misoperation events.

Changes in Dose Consequences

As a result of the new uprated power, accident analyses were either reanalyzed, not reanalyzed or not applicable. Of those events that were reanalyzed, four resulted in dose increases above that currently analyzed. Dose consequences from other events were not changed from those reported in the license amendment titled "Proposed Technical Specification Changes and Resolution of Unreviewed Safety Question Associated with Applicable Limits and Setpoints Supporting Steam Generator Replacement" dated November 29, 1999 (2CAN119901), as supplemented. The four events with increased dose consequences are:

LOCA – The method and results of the LOCA dose analysis are discussed in Section 7.3.10 of the attached Power Uprate Licensing Report. The dose consequences determined for Power Uprate have increased by a small amount. The increase is less than 10% of the remaining difference to the 10CFR100 limits and NRC review is not required as a license amendment under 10CFR50.90, pursuant to the revised 10CFR50.59 rule.

Steam Generator Tube Rupture – The steam generator tube rupture event is discussed in Section 7.3.13 of the attached Power Uprate Licensing Report. The dose consequences determined for Power Uprate have increased, but are still within the regulatory limits. This increase is primarily due to the conservative application of iodine spiking beyond our existing licensing basis.

CEA Ejection – The CEA Ejection event is discussed in Section 7.3.14 of the attached Power Uprate Licensing Report. To accommodate a range of future core designs, the dose consequences from a CEA ejection event with as much as 14% fuel pin failures has been determined. The dose consequences from this event have increased as a result of the assumed fuel failures, power uprate, and more conservative analysis methodology. The dose consequences are still within the regulatory limits.

Fuel Handling Accident – The fuel handling accident is discussed in Section 7.3.15 of the attached Power Uprate Licensing Report. The dose consequences determined for Power Uprate have increased, but are still within the regulatory limits. The dose consequences from this event have increased as a result of power uprate and more conservative assumptions.

Therefore, as a result of the new analyses performed for Power Uprate, NRC review is requested for the Steam Generator Tube Rupture, Fuel Handling Accident and Control Element Assembly

Ejection events. Review is requested in accordance with the requirements of 10CFR50.90, "Application for amendment of license or construction permit," per the requirements of the revised 10CFR50.59 Rule, "Changes, tests and experiments," dated October 4, 1999.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Entergy Operations, Inc. is proposing that the Arkansas Nuclear One, Unit 2 (ANO-2) Operating License be amended to revise affected Technical Specification limits associated with power uprate. This license application, if approved, would revise the operating license and technical specifications to authorize operation of ANO-2 at a nuclear steam supply system power level up to 3026 megawatts thermal (MWt). The proposed change represents a 7.5% increase above the currently licensed core power rating of 2815 megawatts thermal. The new technical specification limits ensure safety margins remain acceptable during subsequent operations at uprated conditions. The analyses supporting the uprate result in one change to the Operating License and several changes to the ANO-2 Technical Specifications.

The following Operating License/Technical Specification changes are required as a result of the power uprate:

- Change the definition of RATED THERMAL POWER from 2815 MWt to 3026 MWt.
- Decrease the Pressurizer Pressure – Low setpoint from ≥ 1675 psia to ≥ 1650 psia and its allowable value from ≥ 1643.9 psia to ≥ 1618.9 psia.
- Replace the refueling water tank level/volume requirements with an allowable range for the available volume in the refueling water tank.
- Decrease the allowable values for the Linear Power Level – High Trip setpoint during operation with one or more main steam safety valves inoperable.
- Decrease the maximum value of the Linear Power Level – High Trip setpoint allowed for a given moderator temperature coefficient during periods when a main steam safety valve(s) is inoperable.
- Revise the administrative control section which lists the analytical methods used to determine the core operating limits to reflect the change in emergency core cooling system evaluation methods being applied to ANO-2.
- Modify the Moderator Temperature Coefficient limit consistent with the accident analysis.
- Modify the linear heat rate limit consistent with the accident analysis.
- Clarify control element assembly (CEA) position requirements.

- Increase in dose consequences which result in a license amendment in accordance with the revised 10CFR50.59 Rule, "Changes, tests, and experiments," dated October 4, 1999.

Changes in MWe output can vary not only with licensed reactor power but also with steam generator condition, turbine and Rankine cycle performance. The replacement steam generators and turbine modifications that were installed during Refueling Outage 2R14 (fall 2000) and power uprate all contribute to regaining and improving the MWe output of ANO-2.

The replacement steam generator design included increased tubing surface area to accommodate power uprate. Additional steam generator tubing surface area was accomplished by increasing the diameter of the lower shell, and therefore the tubesheet, by four inches. The tubesheet diameter increase was a major factor that allowed for a greater number of tubes, and the tubing surface area in each replacement steam generator is approximately 109,000 ft^2 compared to approximately 87,000 ft^2 in the original steam generators, an increase of about 25% in tubing surface area. This surface area permits a 107.5% power uprate while maintaining prudent design for replacement steam generator parameters such as adequate tubesheet structural strength. The replacement steam generators were installed prior to Cycle 15. Power uprate is planned for Cycles 16 and beyond. Cycle 16 is scheduled to begin in April 2002.

An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the standards in 10CFR50.92(c). A discussion of these standards as they relate to this amendment request follows:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

All previously evaluated accidents have been reviewed for the proposed power uprate. The results of both large and small break loss of coolant accident (LOCA) analyses demonstrate continued conformance to the emergency core cooling system acceptance criteria of 10CFR50.46. Non-LOCA safety analyses supporting the proposed changes have been performed and have demonstrated conformance with all applicable licensing basis acceptance criteria. The analyses were performed considering the proposed Safety Limits and the Limiting Safety Settings of the technical specifications. The results of the LOCA and non-LOCA analyses demonstrate that operation at power uprate conditions is acceptable. Reactor Protection System/Engineered Safety Features Actuation System setpoint changes provide functionally equivalent protection at uprated conditions as the previous setpoint values. Proposed changes in regard to High Linear Power Trip setpoints associated with conditions where main steam safety valves are inoperable represent appropriate restrictions that have resulted from the analyses performed in support of power uprate. As a result of the analyses and evaluations performed in support of the power uprate, the ANO specific safety parameters and regulatory limits are protected. The proposed changes to control element assembly position requirements and insertion limits are requested to make wording changes that provide clarification and improved readability, implement conservative changes, and implement operational enhancements which are supported by the power uprate analysis.

The impacts of the proposed power uprate to plant operations were reviewed against the unit's design capability and it was determined that following the completion of the required plant modifications to support the uprate, no system, structure, or component would exceed design conditions or limits. Operation at the power uprated conditions will not unacceptably affect the reliability of plant equipment. Current technical specification surveillance requirements ensure adequate monitoring of system operability. Systems will continue to be operated in accordance with current design requirements under uprated conditions; therefore, no new components or system interactions were identified that could lead to an increase in accident probability. Changes to reactor trip setpoints are such that no significant increase in trip frequency due to operation at uprated conditions will occur.

Challenges to the containment building have been evaluated, and the containment and its associated support systems will continue to meet the requirements of 10CFR50, Appendix A General Design Criterion 38, "Long Term Cooling," and Criterion 50, "Containment." Radiological release events have been evaluated and shown to be within the limits of 10CFR100.

The spectrum of analyzed postulated accidents and transients was investigated for increases in dose consequences. Dose consequences for the steam generator tube rupture, control element assembly ejection and the fuel handling accident meet the criteria for NRC review, but are within the regulatory limits of 10CFR Part 100. In the area of core design, the fuel operating limits will continue to be met at the higher power level, and fuel reload analyses will show plant transients meet NRC accepted criteria. The evaluation of accident consequences was performed consistent with the proposed changes to the plant technical specifications. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed changes to the ANO-2 Operating License and Technical Specifications are analytically based and require changing plant setpoints and procedural limits; therefore, no physical modifications are required as a result of the proposed technical specification changes. Normal plant operations will not be substantially impacted by increasing the rated thermal power to 3026 MWt. Training will be provided to communicate impacts resulting from the uprate and the plant's simulator will be updated consistent with the changes in the plant. Operating procedures will remain largely unchanged; therefore, human performance issues are not introduced as a result of the power uprate.

No new fission product release paths or failure modes are created as a result of an increase in power level. The fission product barriers -- fuel cladding, reactor coolant pressure boundary and the containment building -- remain unchanged. Fuel rod cladding integrity is ensured by operating within thermal, mechanical, and exposure design limits, and is demonstrated by the transient and accident analyses performed for the power uprate. Analysis of the reactor coolant pressure and containment boundaries concludes that the power uprate will not unacceptably affect these fission product barriers. The proposed technical specification changes are consistent with the analyses, and assure transient and accident mitigation capability in compliance with regulatory requirements.

The license and technical specification changes needed to implement the power uprate require only parameter value changes. No new systems will be needed to support implementation of the power uprate; therefore, adding new systems of a different design which could possibly introduce new failure modes or accident sequences is not a concern. The power uprate-related accident considerations defined in Chapter 15, "Accident Analysis," of the ANO-2 Safety Analysis Report have been evaluated and no new or different kind of accident has been identified. The license and technical specification changes have been evaluated and are acceptable.

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

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

Containment, LOCA, and non-LOCA safety analyses supporting the proposed changes have been performed and have demonstrated that the associated margins of safety are within acceptable limits. The transient and accident analyses to support the power uprate were performed at 3026 megawatts thermal and increased by an additional 2% in accordance with regulatory guidance, when required. The analyses demonstrate that sufficient margins of safety exist.

With the increase in Rated Thermal Power, the bases for the setpoints in the ANO-2 Technical Specifications are affected. Based on the new analyses and evaluations conducted in support of this license amendment, the new technical specification setpoints provide adequate margin to protect established safety and regulatory limits. Additionally, no NRC acceptance criteria as required by our licensing basis are exceeded.

The calculated loads on all affected structures, systems, and components remain within their design allowables for all design basis event categories.

Fuel operating limits will continue to be met at the higher power level, and fuel reload analyses will show that plant transients meet the NRC-accepted criteria as specified in the plant's technical specifications. Challenges to fuel and emergency core cooling system performance were evaluated and shown to meet the criteria of 10CFR50.46 and 10CFR50, Appendix K. Challenges to the containment building were evaluated and the integrity of the fission product barrier was confirmed. Radiological release events were evaluated and shown to meet the guidelines of 10CFR100.

The proposed changes to the operating license and technical specifications are consistent with the power uprate evaluations. The evaluations demonstrate compliance with the acceptance criteria contained in applicable codes and regulations. Therefore, this change does not involve a significant reduction in the margin of safety.

Therefore, based on the reasoning presented above and the previous discussion of the amendment request, Entergy Operations, Inc. has determined that the requested changes do not involve a significant hazards consideration.

ENCLOSURE 1

TO

2CAN120001

<u>PROPOSED REVISIONS TO ANO-2 OPERATING LICENSE AND TECHNICAL SPECIFICATIONS/BASES</u>

(4) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) EOI, pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

c. This amended license shall be deemed to contain and is subject to conditions specified in the following Commission regulations in 10 CFR Chapter I; Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

EOI is authorized to operate the facility at steady state reactor core power levels not in excess of 3026 megawatts thermal. Prior to attaining this power level EOI shall comply with the conditions in Paragraph 2.C.(3).

(2) Technical Specifications

Amendment #226 11/13/00

The Technical Specifications contained in Appendix A, as revised through Amendment No. 226 are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications.

Exemptive 2nd paragraph of 2.C.2 deleted per Amendment 20, 3/3/81.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following issuance of the license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission.

2.C.(3)(a) Deleted per Amendment 24, 6/19/81.

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable, throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3026 MWt.

OPERATIONAL MODE - MODE

1.4 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Section 50.73 to 10CFR Part 50.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
a. Four Reactor Coolant Pumps Operating	≤ 110% of RATED THERMAL POWER	≤ 110.712% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.75%	≤ 0.819%
4. Pressurizer Pressure - High	≤ 2362 psia	≤ 2370.887 psia
5. Pressurizer Pressure - Low	≥ 1650 psia (2)	≥ 1618.9 psia (2)
6. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
7. Steam Generator Pressure - Low	≥ 751 psia (3)	≥ 738.6 psia (3)
8. Steam Generator Level - Low	≥ 22.2% (4)	≥ 21.5% (4)

3/4.1.3 CONTROL ELEMENT ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All CEAs shall be OPERABLE with each CEA of a given group positioned within 7 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more CEA(s) inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within the next 6 hours.

b. With one CEA trippable but inoperable due to causes other than addressed by ACTION (a) above, but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specifications 3.1.3.5 and 3.1.3.6.

c. With more than one CEA trippable but inoperable due to causes other than addressed by ACTION (a) above, but within the above specified alignment requirements, restore the inoperable CEA(s) to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours.

d. With one CEA trippable but misaligned from any other CEA in its group by more than 7 inches, operation in MODES 1 and 2 may continue, provided that, for inward deviations, core power is reduced in accordance with the limits specified in the CORE OPERATING LIMITS REPORT and, for all deviations, within 2 hours either:

   1. Restore the misaligned CEA to within its above specified alignment requirements, or

   2. Verify the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. Operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.5 and 3.1.3.6 provided:

      a) Within two hours following the misalignment the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT, and

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

   Otherwise, be in at least HOT STANDBY within the next 6 hours.

*See Special Test Exceptions 3.10.2 and 3.10.4.

ACTION: (Continued)

e. With more than one CEA misaligned from any other CEA in its group by more than 7 inches (indicated position), be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each CEA shall be determined to be within 7 inches (indicated position) of all other CEAs in its group at least once per 12 hours.

4.1.3.1.2 Each CEA not fully inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 92 days.

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REGULATING AND GROUP P CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups and Group P CEAs shall be maintained within the following limits:

a. One or more CEACs operable:

1. The regulating CEA groups and Group P CEAs shall be limited to the withdrawal sequence and to the insertion limits specified in the CORE OPERATING LIMITS REPORT. CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limit is restricted to:

a) ≤ 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

b) ≤ 14 Effective Full Power Days per calendar year.

2. CEA insertion between the Short Term Steady State Insertion Limit and the Transient Insertion Limit shall be restricted to ≤ 4 hours per 24 hour interval.

b. Both CEACs inoperable:

Regulating CEA Group 6 may be inserted no further than 127.5 inches withdrawn which is the Transient Insertion Limit when both CEACs are inoperable. All other CEAs must be maintained fully withdrawn.

APPLICABILITY: MODES 1* and 2*#

ACTION:

a. With the regulating CEA groups or Group P CEAs inserted beyond the Transient Insertion Limit, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours of exceeding the Transient Insertion Limit either:

1. Restore the regulating CEA groups or Group P CEAs to within the limits, or

2. Reduce THERMAL POWER as follows:

a) One or more CEACs Operable:

1) Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position specified in the CORE OPERATING LIMITS REPORT, or

2) Be in at least HOT STANDBY within 8 hours of exceeding the Transient Insertion Limit.

* See Special Test Exceptions 3.10.2 and 3.10.4

With K_{eff} ≥1.0.

ACTION: (Continued)

b) Both CEACs Inoperable:

Be in at least HOT STANDBY within 8 hours of exceeding the Transient Insertion Limit.

b. With the regulating CEA groups or Group P CEAs inserted between the Long Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, either:

1. Restore the regulating groups or Group P CEAs to within the Long Term Steady State Insertion Limit within two hours, or

2. Be in at least HOT STANDBY within the next 6 hours.

c. With the regulating CEA groups or Group P CEAs inserted between the Short Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 4 hours per 24 hour interval, operation may proceed provided any subsequent increase in thermal power is restricted to ≤ 5% of rated thermal power per hour.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group and Group P CEAs shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Alarm is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups or Group P CEAs are inserted beyond the Long Term Steady State Insertion Limit or the Short Term Steady State Insertion Limit but within the Transient Insertion Limit shall be determined at least once per 24 hours.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, place the reactor trip breakers of the inoperable channel in the tripped condition within 1 hour or be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1.

ACTION 6 -

a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEAs in its group. After 7 days, operation may continue provided that ACTION 6.b is met.

b. With both CEACs inoperable, operation may continue provided that:

1. Within 1 hour the margin required by Specification 3.2.4.b (COLSS in service) or Specification 3.2.4.d (COLSS out of service) is satisfied.

2. Within 4 hours:

a) All CEA groups are withdrawn within the limits of Specifications 3.1.3.5 and 3.1.3.6.b, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2.

b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to both CEACs inoperable.

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "OFF" mode except during CEA motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

3. At least once per 4 hours, all CEAs are verified fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, or as permitted by Specification 3.1.3.6.b, then verify at least once per 4 hours that the inserted CEAs are aligned within 7 inches (indicated position) of all other CEAs in their group.

ACTION 7 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or open the affected reactor trip breakers within the next hour. The trip breakers associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
c. Pressurizer Pressure - Low	≥ 1650 psia	≥ 1618.9 psia
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons	Not Applicable	Not Applicable
b. Containment Pressure - High-High	≤ 23.3 psia	≤ 23.490 psia
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual (Trip Buttons	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. MAIN STEAM AND FEEDWATER ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	≥ 751 psia (2)	≥ 738.6 psia (2)
5. CONTAINMENT COOLING (CCAS)		
a. Manual (Trip Buttons	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
c. Pressurizer Pressure - Low	≥ 1650 psia	≥ 1618.9 psia
6. RECIRCULATION (RAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Tank - Low	6.0 ± 0.5% indicated level	5.111% and 6.889% indicated level
7. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage	(4)	2300 ± 699 volts with a 0.64 ± 0.34 second time delay
b. 460 volt Emergency Bus Undervoltage	423 ± 2.0 volts with an 8.0 ± 0.5 second time delay	423 ± 4.0 volts with an 8.0 ± 0.8 second time delay

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. An available borated water volume of between 384,000 and 503,300 gallons

b. Between 2500 and 3000 ppm of boron,

c. A minimum solution temperature of 40°F, and

d. A maximum solution temperature of 110°F

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water tank inoperable, restore tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature.

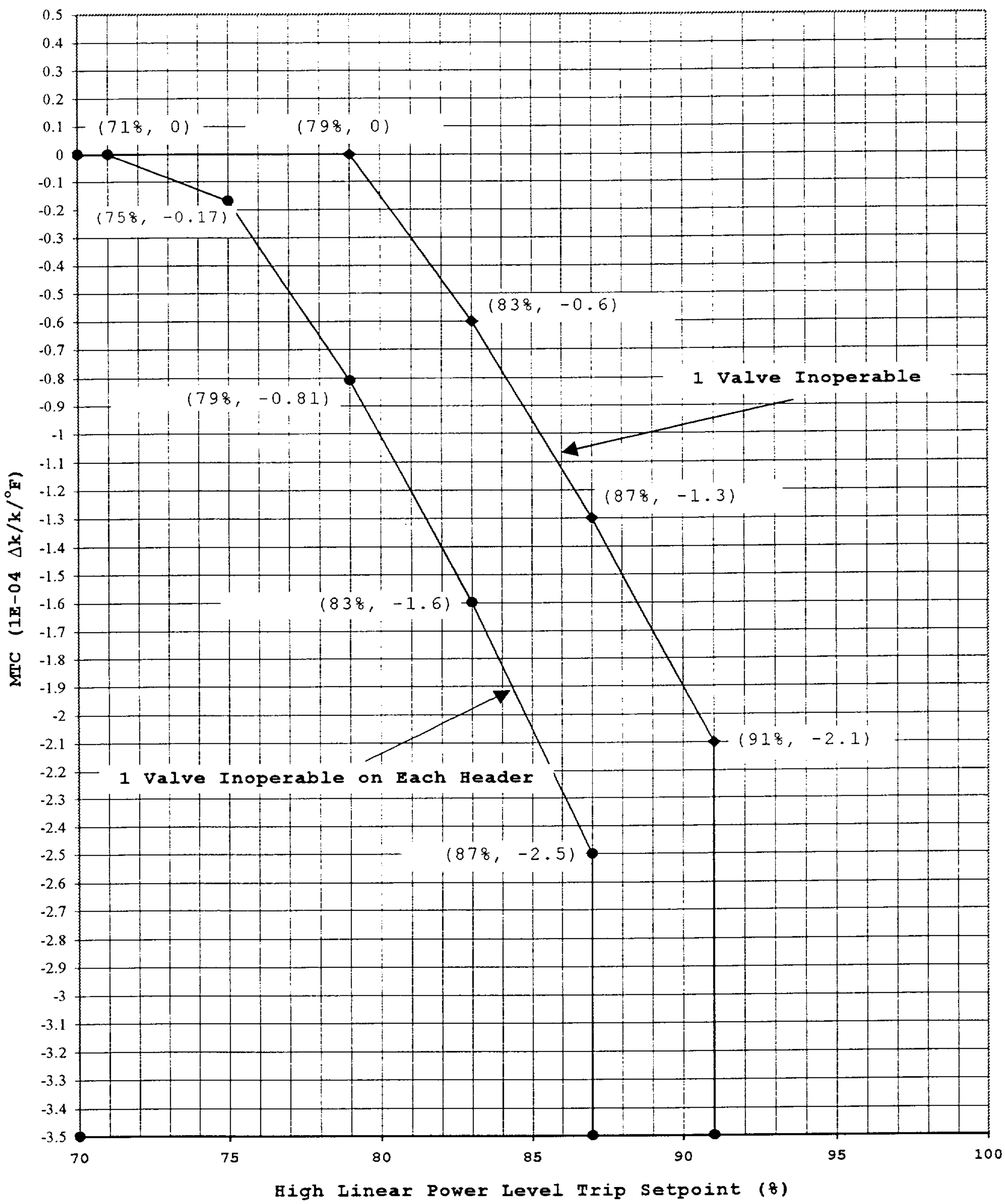
TABLE 3.7-1

MAXIMUM ALLOWABLE LINEAR POWER LEVEL AND HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION WITH BOTH STEAM GENERATORS

Number of Inoperable Safety Valves	Maximum Allowable Linear Power Level And High Trip Setpoint (Percent of RATED THERMAL POWER)
1 Valve Inoperable	79% (except as allowed by Figure 3.7-1)
1 Valve Inoperable on Each Header	71% (except as allowed by Figure 3.7-1)
Maximum of 2 Valves Inoperable on Each Header	43.0
Maximum of 3 Valves Inoperable on Each Header	25.0

FIGURE 3.7-1

Maximum High Linear Power Level And Trip Setpoint Versus MTC



CORE OPERATING LIMITS REPORT

10) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137, Supplement 2-P-A, dated April, 1998 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

11) "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for CEA Position, 3.1.3.6 for Regulating CEA and Group P Insertion Limits, and 3.2.4.b for DNBR Margin).

12) "Technical Manual for the CENTS Code," CENPD 282-P-A, February 1991 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for CEA Position, 3.1.3.6 for Regulating and Group P Insertion Limits, and 3.2.4.b for DNBR Margin.

13) Letter: O.D. Parr (NRC) to F.M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for 6.9.5.1.4, 6.9.5.1.5, and 6.9.5.1.8 methodologies.

14) Letter: O.D. Parr (NRC) to A.E. Scherer (CE), dated December 9, 1975 (NRC Staff Review of the Proposed Combustion Engineering ECCS Evaluation Model changes). NRC approval for 6.9.5.1.6 methodology.

15) Letter: K. Kniel (NRC) to A.E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.5.1.9 methodology.

16) Letter: 2CNA038403, dated March 20, 1984, J.R. Miller (NRC) to J.M. Griffin (AP&L), "CESEC Code Verification." NRC approval for 6.9.5.1.11 methodology.

17) "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," CENPD-132-P, Supplement 4-P-A, Revision 1 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

6.9.5.2 The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.5.3 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

The boron capability required below 200°F is based upon providing a sufficient SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either borated water from the refueling water tank or boric acid solution from the boric acid makeup tank(s) in accordance with the requirements of Specification 3.1.2.7.

The contained water volume limits includes allowance for water not available because of discharge line location and other physical Characteristics. The 61,370 gallon limit for the refueling water tank is based upon having an indicated level in the tank of at least 7.5%.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limits on contained water volume and boron concentration of the boric acid sources, when mixed with the trisodium phosphate, ensures a long term pH value of ≥ 7.0 for the solution recirculated within containment after a LOCA. This pH limit minimizes the evolution of iodine and helps to inhibit stress corrosion cracking of austenitic stainless steel components in containment during the recirculation phase following an accident.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA or a misalignment of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN. CEAs that are confirmed to be inoperable due to problems other than addressed by ACTION (a) of Specification 3.1.3.1 will not impact SHUTDOWN MARGIN as long as their relative positions satisfy the applicable alignment requirements.

For a single CEA misalignment, there is 1) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, 2) a small effect on the available SHUTDOWN MARGIN, and 3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with a trippable but misaligned CEA permits a two hour time interval during which attempts may be made to restore the CEA to within the alignment requirements. The time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution. Problems may also cause more than one control rod to be immovable where the control rods continue to be trippable. With trippable but multiple inoperable rods, the alignment limits and the restriction on THERMAL POWER in accordance with the provisions of Specification 3.1.3.6 for insertion limits assures fuel rod integrity during continued operation. These provisions are sufficient to allow 72 hours to restore the inoperable rods to operable status when it is confirmed that the cause of the immovable rods is an electrical problem in the rod control system or an electrical or mechanical problem with the rod stepping mechanism exclusive of the rod holding coil that must function for a reactor trip. In such cases, the control rods will continue to be capable of fulfilling their primary safety function.

The CPCs provide protection to the core in the event of a large outward misalignment of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA.

The ACTION statements applicable to a trippable but misaligned or inoperable CEA include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements brings the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available SHUTDOWN MARGIN which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F. This limitation also ensures fuel pin pressures will not exceed design limits.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate limit includes appropriate uncertainty and penalty factors necessary to provide a 95/95 confidence level that the maximum linear heat rate calculated by COLSS is greater than or equal to that existing in the core. To ensure that the design margin to safety is maintained, the COLSS computer program includes uncertainties associated with planar radial peaking measurement, engineering design factors, state parameter measurement, software algorithm modeling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the operating limit power level based on linear heat rate, margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits specified in the CORE OPERATING LIMITS REPORT can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs.

The available water volume limits represent the analytically assumed maximum and minimum volume of water that can be transferred from the refueling water tank to containment via the emergency core cooling system and containment spray before pump suction is switched to the sump.

The limits on water volume and boron concentration of the boric acid sources, when mixed with the trisodium phosphate, ensures a long term pH value of ≥ 7.0 for the solution recirculated within containment after a LOCA. This pH limit minimizes the evolution of iodine and helps to inhibit stress corrosion cracking of austenitic stainless steel components in containment during the recirculation phase following an accident.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 102% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The "as-found" requirements are consistent with Section XI of the ASME Boiler and Pressure Vessel Code, 1986 Edition, and Addenda through 1987. The MSSV capacity exceeds that required to maintain steam generator pressure less than 110% of secondary system design pressure following a turbine trip with a loss of condenser vacuum from 102% RATED THERMAL POWER (100% + 2% for instrument error). A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived by an analysis of a loss of condenser vacuum event initiated at the reduced power levels listed in Table 3.7-1 that shows peak steam generator pressures are maintained below 110% of design pressure.

To provide power level limits more amenable to MSSV testing, the LOCV analysis also determines the combination of allowable initial power levels and moderator temperature coefficients (MTC) that yield acceptable results for the single most limiting valve and one bank of valves inoperable. These power level/MTC combinations are the basis of Figure 3.7-1.

The 4-hour completion time for required Action (a) is a reasonable time period to reduce power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs. An additional 8 hours is allowed in Action (a) to reduce the setpoints in recognition of the difficulty of resetting all channels of this trip function within a period of 4 hours. The completion time of 12 hours for Action (a) is based on operating experience in resetting all channels of a protective function and on the low probability of the occurrence of a transient that would result in steam generator overpressure during this period.

ENCLOSURE 2

TO

2CAN120001

<u>MARKUPS TO OPERATING LICENSE AND TECHNICAL SPECIFICATION/BASES PAGES</u>

(For Information Only)

(4) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) EOI, pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

c. This amended license shall be deemed to contain and is subject to conditions specified in the following Commission regulations in 10 CFR Chapter I; Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

EOI is authorized to operate the facility at steady state reactor core power levels not in excess of ~~2815~~3026 megawatts thermal. Prior to attaining this power level EOI shall comply with the conditions in Paragraph 2.C.(3).

(2) Technical Specifications

Amendment #226 11/13/00

The Technical Specifications contained in Appendix A, as revised through Amendment No. 226 are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications.

Exemptive 2nd paragraph of 2.C.2 deleted per Amendment 20, 3/3/81.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following issuance of the license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission.

2.C.(3)(a) Deleted per Amendment 24, 6/19/81.

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable, throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~2815~~ 3026 ~~mwt~~MWt.

OPERATIONAL MODE - MODE

1.4 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Section 50.73 to 10CFR Part 50.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
a. Four Reactor Coolant Pumps Operating	≤ 110% of RATED THERMAL POWER	≤ 110.712% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.75%	≤ 0.819%
4. Pressurizer Pressure - High	≤ 2362 psia	≤ 2370.887 psia
5. Pressurizer Pressure - Low	≥ ~~1675~~ 1650 psia (2)	≥ ~~1643.9~~1618.9 psia (2)
6. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
7. Steam Generator Pressure - Low	≥ 751 psia (3)	≥ 738.6 psia (3)
8. Steam Generator Level - Low	≥ 22.2% (4)	≥ 21.5% (4)

3/4.1.3 CONTROL ELEMENT ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All CEAs shall be OPERABLE with each CEA of a given group positioned within 7 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more CEA(s) inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within the next 6 hours.

~~b. With one CEA trippable but inoperable due to causes other than addressed by ACTION (a), above, and inserted beyond the Long Term Steady State Insertion Limits but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.~~

~~c~~b. With one CEA trippable but inoperable due to causes other than addressed by ACTION (a)~~,~~ above, but within its above specified alignment requirements~~ and either fully withdrawn or within the Long Term Steady State Insertion Limits if in CEA group 6 or group P~~, operation in MODES 1 and 2 may continue pursuant to the requirements of Specifications 3.1.3.5 and 3.1.3.6.

~~d~~c. With more than one CEA trippable but inoperable due to causes other than addressed by ACTION (a)~~,~~ above, but within the above specified alignment requirements, restore the inoperable CEA(s) to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours.

~~e~~d. With one ~~or more~~ CEA~~(s)~~ trippable but misaligned from any other CEA in its group by more than 7 inches~~ but less than or equal to 19 inches~~, operation in MODES 1 and 2 may continue, provided that, for inward deviations, core power is reduced in accordance with the limits specified in the CORE OPERATING LIMITS REPORT and, for all deviations, within ~~1~~2 hours ~~the misaligned CEA(s) is~~ either:

1. Restore~~d~~ the misaligned CEA to ~~OPERABLE status~~ within its above specified alignment requirements, or

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2. ~~Declared inoperable and~~Verify the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. ~~After declaring the CEA inoperable,~~ ~~o~~Operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.5 and 3.1.3.6 provided:

   a) Within ~~one~~ two hours following the misalignment the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT~~;~~, and~~the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.~~

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b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours~~.~~;

Otherwise, be in at least HOT STANDBY within the next 6 hours.

*See Special Test Exceptions 3.10.2 and 3.10.4.

ACTION: (Continued)

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~~2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:~~

~~a) Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.~~

~~b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.~~

~~Otherwise, be in at least HOT STANDBY within the next 6 hours.~~

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~~g~~e. With more than one CEA ~~trippable but~~ misaligned from any other CEA in its group by more than ~~19~~ 7 inches (indicated position), be in at least HOT STANDBY within 6 hours.

~~f. With one CEA trippable but misaligned from any other CEA in its group by more than 19 inches, operation in MODES 1 and 2 may continue, provided that core power is reduced in accordance with the limits specified in the CORE OPERATING LIMITS REPORT and within one hour, the misaligned CEA is either:~~

~~1. Restored to OPERABLE status within its above specified alignment requirements, or~~

~~2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:~~

~~a) Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.~~

~~b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.~~

~~Otherwise, be in at least HOT STANDBY within the next 6 hours.~~

SURVEILLANCE REQUIREMENTS

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4.1.3.1.1 The position of each CEA shall be determined to be within 7 inches (indicated position) of all other CEAs in its group at least once per 12 hours.

4.1.3.1.2 Each CEA not fully inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 92 days.

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~~REACTIVITY CONTROL SYSTEMS~~

~~ACTION: (Continued)~~

~~g. With more than one CEA trippable but misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.1.3.1.1 The position of each CEA shall be determined to be within 7 inches (indicated position) of all other CEAs in its group at least once per 12 hours.~~

~~4.1.3.1.2 Each CEA not fully inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 92 days.~~

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REGULATING AND GROUP P CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups and Group P CEAs shall be maintained within the following limits:

a. One or more CEACs operable:

1. The regulating CEA groups and Group P CEAs shall be limited to the withdrawal sequence and to the insertion limits specified in the CORE OPERATING LIMITS REPORT. ~~with:~~

~~a.~~ CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limit is restricted to:

a) ~~1.~~ ≤ 5 Effective Full Power Days per 30 Effective Full Power Day interval~~s~~, and

b) ~~2.~~ ≤ 14 Effective Full Power Days per calendar year.

2. ~~b.~~ CEA insertion between the Short Term Steady State Insertion Limit and the Transient Insertion Limit shall be restricted to ≤ 4 hours per 24 hour interval.

b. Both CEACs inoperable:

Regulating CEA Group 6 may be inserted no further than 127.5 inches withdrawn which is the Transient Insertion Limit when both CEACs are inoperable. All other CEAs must be maintained fully withdrawn.

APPLICABILITY: MODES 1* and 2*#

ACTION:

a. With the regulating CEA groups or Group P CEAs inserted beyond the Transient Insertion Limit, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours of exceeding the Transient Insertion Limit either:

1. Restore the regulating CEA groups or Group P CEAs to within the limits, or

2. Reduce THERMAL POWER as follows:

a) One or more CEACs Operable:

1) Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position ~~using~~ specified in the CORE OPERATING LIMITS REPORT, or

* See Special Test Exceptions 3.10.2 and 3.10.4

With K_{eff} ≥1.0.

2) Be in at least HOT STANDBY within 8 hours of exceeding the Transient Insertion Limit.

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~~b. With the regulating CEA groups or Group P CEAs inserted between the Long Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, either:~~

~~1. Restore the regulating groups or Group P CEAs to within the Long Term Steady State Insertion Limit within two hours, or~~

~~2. Be in at least HOT STANDBY within the next 6 hours.~~

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

b) Both CEACs Inoperable:

Be in at least HOT STANDBY within 8 hours of exceeding the Transient Insertion Limit.

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b. With the regulating CEA groups or Group P CEAs inserted between the Long Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, either:

1. Restore the regulating groups or Group P CEAs to within the Long Term Steady State Insertion Limit within two hours, or

2. Be in at least HOT STANDBY within the next 6 hours.

c. With the regulating CEA groups or Group P CEAs inserted between the Short Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 4 hours per 24 hour interval, operation may proceed provided any subsequent increase in thermal power is restricted to ≤ 5% of rated thermal power per hour.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group and Group P CEAs shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Alarm is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups or Group P CEAs are inserted beyond the Long Term Steady State Insertion Limit or the Short Term Steady State Insertion Limit but within the Transient Insertion Limit shall be determined at least once per 24 hours.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, place the reactor trip breakers of the inoperable channel in the tripped condition within 1 hour or be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1.

ACTION 6 -

a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) ~~or~~ of all other CEAs in its group. After 7 days, operation may continue provided that ACTION 6.b is met.

b. With both CEACs inoperable, operation may continue provided that:

1. Within 1 hour the margin required by Specification 3.2.4.b (COLSS in service) or Specification 3.2.4.d (COLSS out of service) is satisfied.

2. Within 4 hours:

a) All CEA groups are withdrawn within the limits of Specifications 3.1.3.5 and 3.1.3.6.b, ~~to and subsequently maintained at the "Full Out" position,~~ except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2~~ or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn~~.

b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to both CEACs inoperable.

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "OFF" mode except during CEA motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

3. At least once per 4 hours, all CEAs are verified fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, or as permitted by ~~2 a) above~~Specification 3.1.3.6.b, then verify at least once per 4 hours that the inserted CEAs are aligned within 7 inches (indicated position) of all other CEAs in their group.

ACTION 7 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or open the affected reactor trip breakers within the next hour. The trip breakers associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT			TRIP SETPOINT	ALLOWABLE VALUES
1.	SAFETY INJECTION (SIAS)			
	a.	Manual (Trip Buttons)	Not Applicable	Not Applicable
	b.	Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
	c.	Pressurizer Pressure - Low	≥ ~~1675~~ 1650 psia	≥ ~~1643.9~~1618.9 psia
2.	CONTAINMENT SPRAY (CSAS)			
	a.	Manual (Trip Buttons	Not Applicable	Not Applicable
	b.	Containment Pressure - High-High	≤ 23.3 psia	≤ 23.490 psia
3.	CONTAINMENT ISOLATION (CIAS)			
	a.	Manual (Trip Buttons	Not Applicable	Not Applicable
	b.	Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. MAIN STEAM AND FEEDWATER ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	≥ 751 psia (2)	≥ 738.6 psia (2)
5. CONTAINMENT COOLING (CCAS)		
a. Manual (Trip Buttons	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
c. Pressurizer Pressure - Low	≥ ~~1675~~ 1650 psia	≥ ~~1643.9~~1618.9 psia
6. RECIRCULATION (RAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Tank - Low	~~54,400 ± 2,370 gallons (equivalent to~~ 6.0 ± 0.5% indicated level~~)~~	~~between 51,050 and 58,600 gallons (equivalent to between~~ 5.111% and 6.889% indicated level~~)~~
7. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage	(4)	2300 ± 699 volts with a 0.64 ± 0.34 second time delay
b. 460 volt Emergency Bus Undervoltage	423 ± 2.0 volts with an 8.0 ± 0.5 second time delay	423 ± 4.0 volts with an 8.0 ± 0.8 second time delay

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. An available~~contained~~ borated water volume of between ~~464,900~~384,000 and ~~500,500~~503,300 gallons~~ (equivalent to an indicated level between 91.7% and 100%, respectively)~~,

b. Between 2500 and 3000 ppm of boron,

c. A minimum solution temperature of 40°F, and

d. A maximum solution temperature of 110°F

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water tank inoperable, restore tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature.

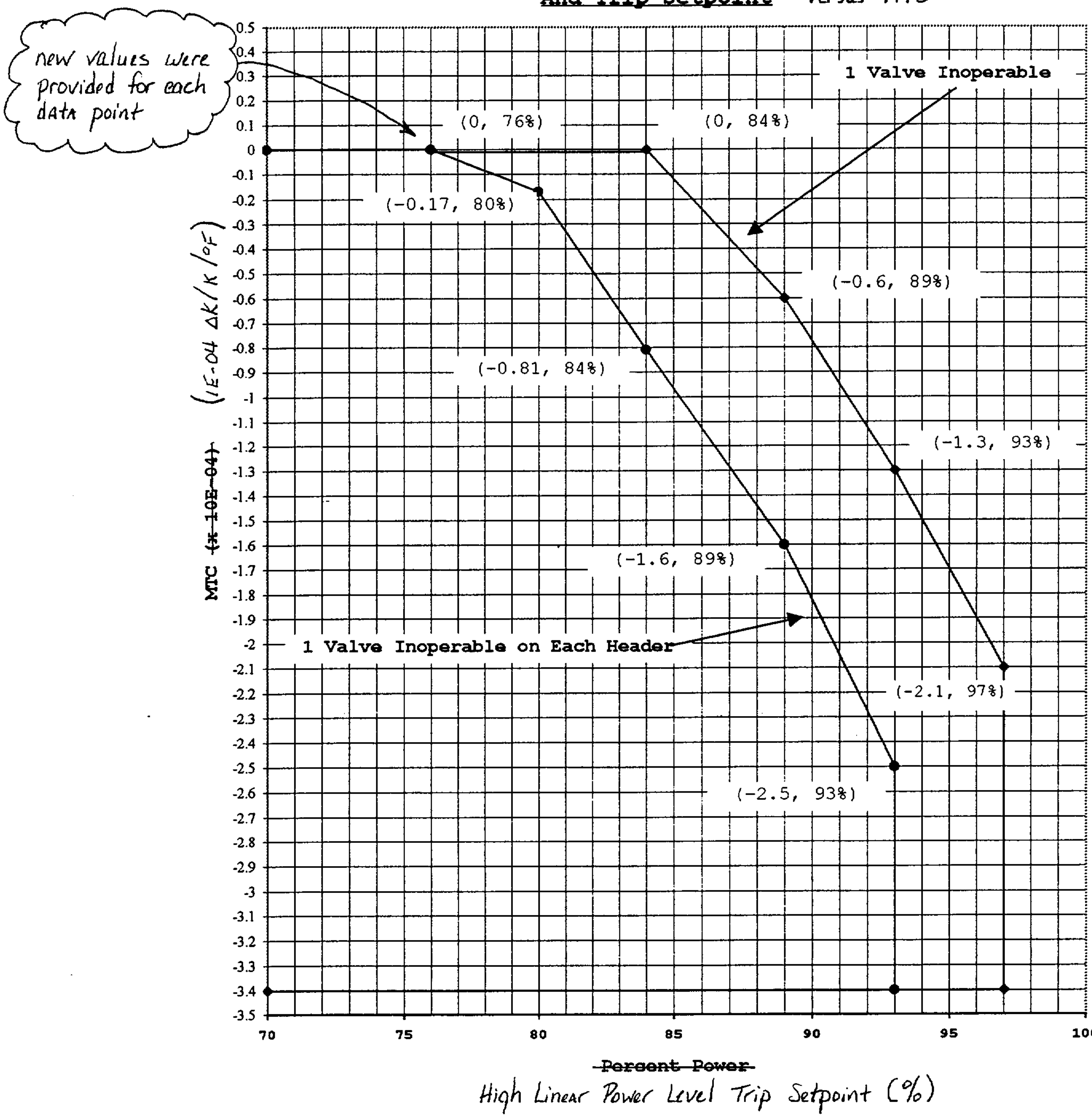
TABLE 3.7-1

MAXIMUM ALLOWABLE LINEAR POWER LEVEL AND HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION WITH BOTH STEAM GENERATORS

Number of Inoperable Safety Valves	Maximum Allowable Linear Power Level And High Trip Setpoint (Percent of RATED THERMAL POWER)
1 Valve Inoperable	~~84~~79% (except as allowed by Figure 3.7-1)
1 Valve Inoperable on Each Header	~~76~~71% (except as allowed by Figure 3.7-1)
Maximum of 2 Valves Inoperable on Each Header	43.0
Maximum of 3 Valves Inoperable on Each Header	25.0

FIGURE 3.7-1

~~MTC Versus~~ Maximum High Linear Power Level And Trip Setpoint Versus MTC



CORE OPERATING LIMITS REPORT

10) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137, Supplement 2-P-A, dated April, 1998 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

11) "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for CEA Position, 3.1.3.6 for Regulating CEA and Group P Insertion Limits, and 3.2.4.b for DNBR Margin).

12) "Technical Manual for the CENTS Code," CENPD 282-P-A, February 1991 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for CEA Position, 3.1.3.6 for Regulating and Group P Insertion Limits, and 3.2.4.b for DNBR Margin.

13) Letter: O.D. Parr (NRC) to F.M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for 6.9.5.1.4, 6.9.5.1.5, and 6.9.5.1.8 methodologies.

14) Letter: O.D. Parr (NRC) to A.E. Scherer (CE), dated December 9, 1975 (NRC Staff Review of the Proposed Combustion Engineering ECCS Evaluation Model changes). NRC approval for 6.9.5.1.6 methodology.

15) Letter: K. Kniel (NRC) to A.E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.5.1.9 methodology.

16) Letter: 2CNA038403, dated March 20, 1984, J.R. Miller (NRC) to J.M. Griffin (AP&L), "CESEC Code Verification." NRC approval for 6.9.5.1.11 methodology.

17) "Calculative Methods for the ABB CE Nuclear Power Large Break LOCA Evaluation Model," CENPD-132-P, Supplement 4-P-A, Revision 1 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

6.9.5.2 The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.5.3 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

BASES

The boron capability required below 200°F is based upon providing a sufficient SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either borated water from the refueling water tank or boric acid solution from the boric acid makeup tank(s) in accordance with the requirements of Specification 3.1.2.7.

The contained water volume limits includes allowance for water not available because of discharge line location and other physical characteristics. The 61,370 gallon limit for the refueling water tank is based upon having an indicated level in the tank of at least 7.5%.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limits on contained water volume and boron concentration of the boric acid sources, when mixed with the trisodium phosphate, ensures a long term pH value of ≥ 7.0 for the solution recirculated within containment after a LOCA. This pH limit minimizes the evolution of iodine and helps to inhibit stress corrosion cracking of austenitic stainless steel components in containment during the recirculation phase following an accident.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA or a ~~large~~ misalignment ~~(≥ 19 inches)~~ of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN. CEAs that are confirmed to be inoperable due to problems other than addressed by ACTION (a) of Specification 3.1.3.1 will not impact SHUTDOWN MARGIN as long as their relative positions satisfy the applicable alignment requirements.

For ~~small~~ a single CEA misalignment~~s (< 19 inches) of the CEAs~~, there is 1) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, 2) a small effect on the available SHUTDOWN MARGIN, and 3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with a trippable but ~~small~~ misaligned~~ments of~~ CEA~~s~~ permits a ~~one~~ two hour time interval during which attempts may be made to restore the CEA to within the~~its~~ alignment requirements. The ~~one hour~~ time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution. Problems may also cause more than one control rod to be immovable where the control rods continue to be trippable. With trippable but multiple inoperable rods,~~;~~ the alignment limits and the restriction on THERMAL POWER in accordance with the provisions of Specification 3.1.3.6 for insertion limits~~,~~ assures fuel rod integrity during continued operation. These provisions are sufficient to allow 72 hours to restore the inoperable rods to operable status when it is confirmed that the cause of the immovable rods is an electrical problem in the rod control system or an electrical or mechanical problem with the rod stepping mechanism exclusive of the rod holding coil that must function for a reactor trip. In such cases, the control rods will continue to be capable of fulfilling their primary safety function.

The CPCs provide protection to the core in the event of a large outward misalignment ~~(≥ 19 inches)~~ of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. ~~However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on 1) the available SHUTDOWN MARGIN, 2) the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, and 3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.~~

The ACTION statements applicable to a trippable but misaligned or inoperable CEA~~s~~ include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements brings the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available SHUTDOWN MARGIN which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed ~~on operation with inoperable CEAs~~ to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F. This limitation also ensures fuel pin pressures will not exceed design limits.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate limit includes appropriate uncertainty and penalty factors necessary to provide a 95/95 confidence level that the maximum linear heat rate calculated by COLSS is greater than or equal to that existing in the core. To ensure that the design margin to safety is maintained, the COLSS computer program includes uncertainties associated with planar radial peaking measurement, engineering design factors, state parameter measurement, software algorithm modeling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the operating limit power level based on linear heat rate, margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits specified in the CORE OPERATING LIMITS REPORT can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs.

BASES

The available~~contained~~ water volume limits represent the analytically assumed maximum and minimum volume of water that can be transferred from the refueling water tank to containment via the emergency core cooling system and containment spray before pump suction is switched to the sump~~includes an allowance for water not usable because of tank discharge line location or other physical characteristics~~.

The limits on ~~contained~~ water volume and boron concentration of the boric acid sources, when mixed with the trisodium phosphate, ensures a long term pH value of ≥ 7.0 for the solution recirculated within containment after a LOCA. This pH limit minimizes the evolution of iodine and helps to inhibit stress corrosion cracking of austenitic stainless steel components in containment during the recirculation phase following an accident.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from ~~100~~102% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The "as-found" requirements are consistent with Section XI of the ASME Boiler and Pressure Vessel Code, 1986 Edition, and Addenda through 1987. The MSSV capacity exceeds that required to maintain steam generator pressure less than 110% of secondary system design pressure following a turbine trip with a loss of condenser vacuum from~~the~~ 102% RATED THERMAL POWER (100% + 2% for instrument error)~~ steam flow with steam pressure at 110% of the secondary system design pressure~~. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived by an analysis of a loss of condenser vacuum event initiated at the reduced power levels listed in Table 3.7-1 that shows peak steam generator pressures are maintained below 110% of design pressure.

To provide power level limits more amenable to MSSV testing, the LOCV analysis also determines the combination of allowable initial power levels and moderator temperature coefficients (MTC) that yield acceptable results for the single most limiting valve and one bank of valves inoperable. These power level/MTC combinations are the basis of Figure 3.7-1.

The 4-hour completion time for required Action (a) is a reasonable time period to reduce power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs. An additional 8 hours is allowed in Action (a) to reduce the setpoints in recognition of the difficulty of resetting all channels of this trip function within a period of 4 hours. The completion time of 12 hours for Action (a) is based on operating experience in resetting all channels of a protective function and on the low probability of the occurrence of a transient that would result in steam generator overpressure during this period.

ENCLOSURE 3

TO

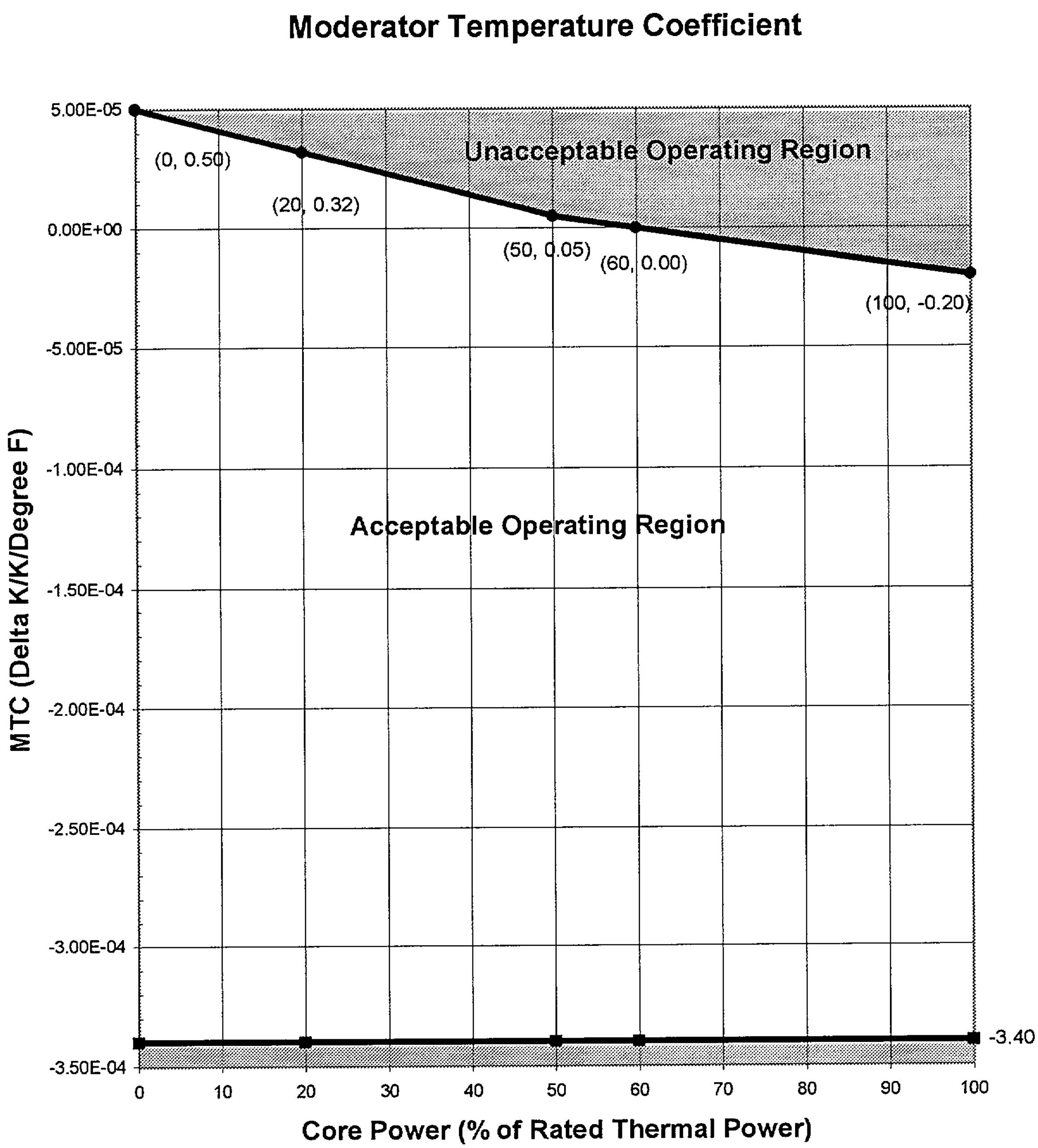
2CAN120001

<u>REVISIONS TO CORE OPERATING LIMITS REPORT (COLR)</u>

(For Information Only)

Figure 1 will be revised IAW 10CFR50.59 changing the most negative limit from -3.4 to -3.8 x 10^{-4} Δk/k/°F

FIGURE 1



Note:

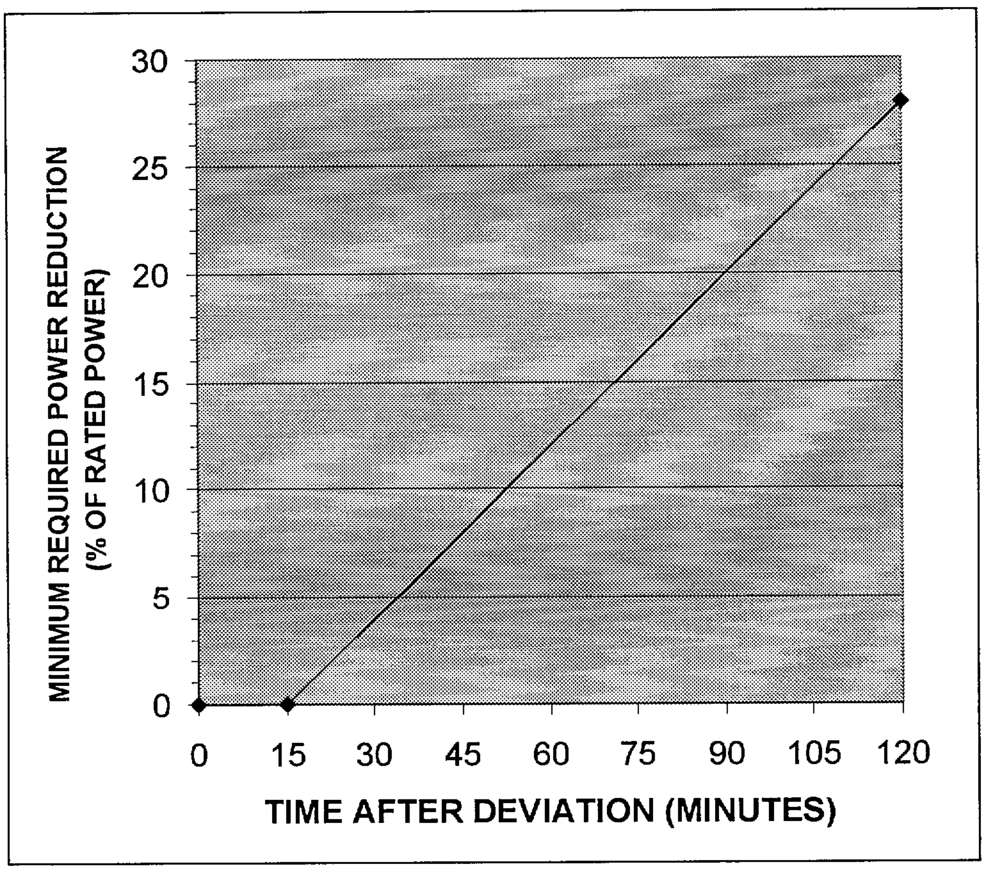
Per Technical Specification 3.1.1.4.a. and b., the Moderator Temperature Coefficient (MTC) maximum upper design limit shall be less positive than +0.5 x 10^{-4} Δk/k/°F whenever THERMAL POWER is ≤ 70% of RATED THERMAL POWER and less positive than 0.0 x 10^{-4} Δk/k/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER. Therefore, the actual MTC must be less than the COLR upper limit at zero power. At all other powers, the actual MTC may be equal to the COLR upper limit.

Existing Figure 2 of the Cycle 15 COLR will be replaced with the following figure:

FIGURE 2

REQUIRED POWER REDUCTION AFTER INWARD CEA DEVIATION WITH BOTH CEACs INOPERABLE *

***When core power is reduced to 60% of rated power per this limit curve, further reduction is not required**



Note to reviewer - the linear heat rate limit will be revised as specified below in accordance with (IAW) 10CFR50.59

6) 3/4.2.1 - LINEAR HEAT RATE

With COLSS out of service, the linear heat rate shall be maintained ≤ 13.7 kW/ft for cycle burnups up to 200 EFPD; and ≤ 13.0 kW/ft for burnups exceeding 200 EFPD.

7) 3.2.3 - AZIMUTHAL POWER TILT- T_q

The measured AZIMUTHAL POWER TILT shall be maintained ≤ 0.03.

8) 3/4.2.4 - DNBR MARGIN

The DNBR limit shall be maintained by one of the following methods:

a) With COLSS in service and neither CEAC operable - Maintain COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by 13%.

b) With COLSS out of service and at least one CEAC operable - Operate within the Region of Acceptable Operation shown on Figure 4, using any operable CPC channel.

c) With COLSS out of service and neither CEAC operable - Operate within the Region of Acceptable Operation shown on Figure 5, using any operable CPC channel.

9) 3.2.7 - AXIAL SHAPE INDEX

The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a) COLSS IN SERVICE
$-0.27 \leq ASI \leq +0.27$

b) COLSS OUT OF SERVICE (CPC)
$-0.20 \leq ASI \leq +0.20$

ENCLOSURE 4

TO

2CAN120001

REVISED TECHNICAL REQUIREMENTS MANUAL (TRM) PAGES

(For Information Only)

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. One boric acid makeup tank with:

1. A minimum indicated tank level of 36%,

2. A boric acid concentration between 3.0 WT% and 3.5 WT%, and

3. A minimum solution temperature of 55°F.

b. The refueling water tank with:

1. A minimum indicated tank level of 7.5%,

2. A minimum boron concentration of 2500 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

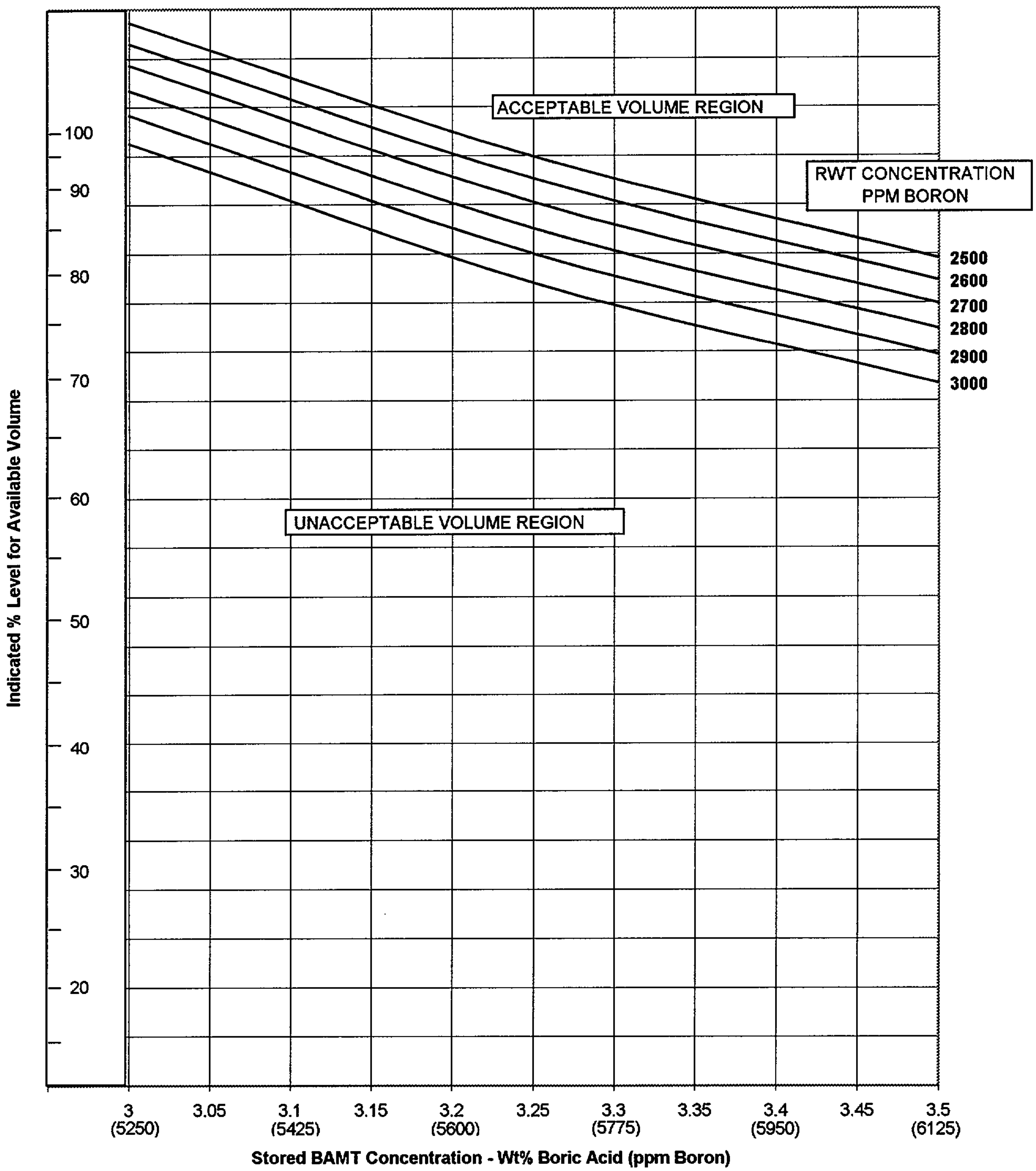
2. Verifying the contained borated water volume of the tank, and

3. Verifying the boric acid makeup tank solution temperature is greater than 55°F.

b. At least once per 24 hours by verifying the RWT temperature when it is the source of borated water and the outside air temperature is < 40°F.

Figure 3.1-1

Minimum Boric Acid Makeup Tank Volume as a Function of Stored BAMT Concentration and Refueling Water Storage Tank Concentration



REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. At least one of the following sources with a minimum solution temperature of 55°F.

1. One boric acid makeup tank, with the tank contents in accordance with Figure 3.1-1, or

2. Two boric makeup tanks, with the combined contents of the tanks in accordance with Figure 3.1-1, and

b. The refueling water tank with:

1. An indicated tank level of between 91.7% and 100%,

2. Between 2500 and 3000 ppm of boron,

3. A minimum solution temperature of 40°F, and

4. A maximum solution temperature of 110°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank(s) inoperable, restore the make up tank(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least that specified in the CORE OPERATING LIMITS REPORT at 200°F; restore the above required boric acid makeup tank(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operations. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid makeup pumps, 5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of that specified in the CORE OPERATING LIMITS REPORT after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 and a small fraction of the borated water from the refueling water tank required in Specification 3.1.2.8.

The requirement in TRM Specification 3.1.2.8 for a minimum available volume of borated water in the refueling water tank ensures the capability for borating the RCS to the desired concentration. The value listed is consistent with the plant ECCS requirements.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a sufficient SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either borated water from the refueling water tank or boric acid solution from the boric acid makeup tank(s) in accordance with the requirements of Specification 3.1.2.7. The contained water volume limits includes allowance for water not available because of discharge line location and other physical characteristics.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limits on water volume and boron concentration of the boric acid sources, when mixed with the trisodium phosphate, ensures a long term pH value of ≥ 7.0 for the solution recirculated within containment after a LOCA. This pH limit minimizes the evolution of iodine and helps to inhibit stress corrosion cracking of austenitic stainless steel components in containment during the recirculation phase following an accident.

ENCLOSURE 5

TO

2CAN120001

<u>POWER UPRATE LICENSING REPORT</u>

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List of Acronyms

AAC	Alternate AC
ACW	Auxiliary Cooling Water
ANO	Arkansas Nuclear One
AOO	Anticipated Operational Occurrence
AOR	Analysis of Record
ASI	Axial Shape Index
BAM	Boric Acid Mix
BLPB	Branch Line Pipe Break
BOC	Beginning of Cycle
BOP	Balance of Plant
CBC	Critical Boron Concentration
CCS	Containment Cooling System
CCWCC	Containment Chilled Water Cooling Coils
CEA	Control Element Assembly
CEAC	Core Element Assembly Calculator
CEAW	Control Element Assembly Withdrawal
CEDM	Control Element Drive Mechanism
CEDMCS	Control Element Drive Mechanism Control System
CENP	CE Nuclear Power, LLC
CENTS	Combustion Engineering Nuclear Transient Simulator
CFWS	Condensate and Feedwater System
COLR	Core Operating Limits Report
COLSS	Core Operating Limit Supervisory System
CPC	Core Protection Calculator
CPCS	Core Protection Calculator System
CRH	Control Room Habitability
CSB	Core Support Barrel (Reactor Vessel)
CST	Condensate Storage Tank
CTM	Centerline-to-Melt

List of Acronyms (cont.)

CVCS	Chemical and Volume Control System
DBE	Design Basis Earthquake
DEG/PD	Double-Ended Guillotine breaks in the reactor coolant Pump Discharge leg
DF	Decontamination Factor
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DW	Deadweight
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
ECP	Emergency Cooling Pond
EDG	Emergency Diesel Generators
EFAS	Emergency Feedwater Actuation Signal
EFPD	Effective Full Power Days
EFPH	Effective Full Power Hours
EFPY	Effective Full Power Years
EFW	Emergency Feedwater
EM	Evaluation Model
EOC	End of Cycle
EOI	Entergy Operations, Inc.
EPA	Environmental Protection Agency
EQ	Environmental Qualification
ESFAS	Engineered Safety Features Actuation System
FAC	Flow Accelerated Corrosion
FHA	Fuel Handling Accident
Fr	Radial Distortion Factor
FWCS	Feedwater Control System
FWLB	Feedwater Line Break
GDC	General Design Criteria
GIS	(Event-)Generated Iodine Spiking
GWD/MTU	Gigawatt Days per Metric Ton Uranium

List of Acronyms (cont.)

GEIS	Generic Environmental Impact Statement
HELB	High Energy Line Break
HFP	Hot Full Power
HLPT	High Linear Power Trip
HPP	High Pressurizer Pressure
HPSI	High Pressure Safety Injection
HVAC	Heating, Ventilation and Air Conditioning
HZP	Hot Zero Power
I&C	Instrumentation and Controls
IBW	Inverse Boron Worth
IPE	Infrequently Performed Evolution
LBB	Leak before Break
LCO	Limiting Condition for Operation
LHR	Linear Heat Rate
LOAC	Loss of AC
LOCA	Loss of Coolant Accident
LOCV	Loss of Condenser Vacuum
LOF	Loss of (Reactor Coolant) Flow
LPD	Local Power Density
LPSI	Low Pressure Safety Injection
LPZ	Low Population Zone
LSGL	Low Steam Generator Level
LSGP	Low Steam Generator Pressure
LSS	Lower Support Structure (Reactor Vessel)
LTC	Long Term Cooling
LTOP	Low Temperature Overpressure Protection
MCL	Main Coolant Loop (Reactor Coolant System)
MHA	Maximum Hypothetical Accident
MSIS	Main Steam Isolation Signal
MSIV	Main Steam Isolation Valve

List of Acronyms (cont.)

MS	Main Steam
MSLB	Main Steam Line Break
MSR	Moisture Separator-Reheaters
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
MVAR	Mega-Volt-Amp Reactance
MWe	Megawatts Electrical
MWt	Megawatts Thermal
NEMA	National Electrical Manufacturers Association
NOp	Normal Operating
NPDES	National Pollution Discharge Elimination System
NPSH	Net Positive Suction Head
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OSG	Original Steam Generator
PD	Pump Discharge
pf	Power factor
PIS	Pre-existing Iodine Spiking
PLCS	Pressurizer Level Control System
PLHGR	Peak Linear Heat Generation Rate
PPCS	Pressurizer Pressure Control System
PSA	Probabilistic Safety Assessment
PSV	Pressurizer Safety Valve
PT, P/T	Pressure and Temperature
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RIR	Reactivity Insertion Rate

List of Acronyms (cont.)

RCPB	Reactor Coolant Pressure Boundary
ROPM	Required Over-Power Margin
RPS	Reactor Protection System
RSG	Replacement Steam Generator
RTD	Resistance Temperature Detector
RTO	Reactor Trip Override
RV	Reactor Vessel
RVI	Reactor Vessel Internals
SAFDL	Specified Acceptable Fuel Design Limit
SAR	Safety Analysis Report (the Updated Final Safety Analysis Report)
SDBCS	Steam Dump and Bypass Control System
SDBS	Steam Dump and Bypass System
SDCS	Shutdown Cooling System
SDM	Shutdown Margin
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIAS	Safety Injection Actuation Signal
SIS	Safety Injection System
SIT	Safety Injection Tank
SPDS	Safety Parameters Display System
SRSS	Square Root Sum of the Squares
SSC	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
SW	Service Water
TAM	Thermal Anchor Movement
TH	Thermal Expansion
UBDI	Uncontrolled Boron Dilution Incident
UGS	Upper Guide Structure (Reactor Vessel)
UHS	Ultimate Heat Sink
VOPT	Variable Overpower Trip
VWO	Valves-Wide-Open

EXECUTIVE SUMMARY

The purpose of the Arkansas Nuclear One, Unit 2 (ANO-2) Power Uprate is to increase the electrical output of ANO-2. The proposed power uprate will permit power operation up to 3026 megawatts thermal (MWt). This power level is 7.5% above the current maximum rated thermal power of 2815 MWt. Enclosure 2, the Power Uprate Licensing Report, documents the acceptability for operating at the increased power level. Westinghouse Topical Report WCAP-10623, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," (January 1983) was used as a guideline in preparation of the ANO-2 Power Uprate Report.

To ensure protection of the health and safety of plant personnel and the public following the power uprate, the ANO-2 design basis and accident analyses were reviewed with respect to associated changes to the plant operating conditions and characteristics. The analyses and evaluations supporting the uprate were performed by CE Nuclear Power, LLC (formerly ABB-CE), a division of Westinghouse Electric Company, LLC, and Entergy Operations, Inc. These reviews are described in the report. In particular, Section 7.1 presents a summary of the emergency core cooling system (ECCS) performance analysis that demonstrates conformance to the ECCS acceptance criteria for light water nuclear power reactors, 10CFR50.46, for ANO-2 at uprate conditions. Section 7.3 summarizes the ANO-2 Safety Analysis Report Chapter 15 accident analyses that demonstrate conformance with the applicable acceptance criteria at power uprate conditions. A listing of these scenarios may be found in Table 7.3-1. The table identifies how each accident event was evaluated in relation to power uprate. The safety analyses and design reviews demonstrate that the acceptance criteria are met.

In response to our application dated November 29, 1999 (2CAN119901), and various supplemental letters, the NRC granted License Amendment No. 222 dated September 29, 2000 (2CNA090002), to revise the license and technical specifications to maintain consistency with the transient and accident analyses which evaluated the impact of the replacement steam generators. Many of these analyses were performed assuming uprated conditions. This report will not reproduce the information contained in those submittals but will make reference to them as appropriate. The uprate analyses/evaluations were performed in accordance with the current ANO-2 licensing bases except where noted.

Additionally, the design pressure of the ANO-2 containment building has been increased to 59 psig. This was necessary due to the increased peak accident pressure that resulted primarily from the larger water volume in the replacement steam generators. The NRC granted the design pressure increase in License Amendment No. 225 dated November 13, 2000 (2CNA110002), based on our request of November 3, 1999 (2CAN119903), as supplemented. Again, this report will not reproduce the information contained in these submittals but will make reference to them as appropriate. Containment response is reviewed in Section 7.2.

While extensive reviews of balance of plant (BOP) systems have been completed, this submittal focuses only on those BOP systems that interface with nuclear steam supply systems. The remaining BOP analyses and evaluations are on file at ANO.

To accommodate operation at the uprated power level, several plant modifications were necessary. These modifications are discussed in the report.

1 INTRODUCTION

1.1 PURPOSE AND SCOPE

In order to increase the rated thermal power of a nuclear power plant, detailed analyses and evaluations are required to ensure the plant can operate acceptably and in compliance with applicable licensing requirements at the increased rated thermal power conditions. The Arkansas Nuclear One, Unit 2 (ANO-2) effort to evaluate and analyze the plant for operation at the increased rated thermal power conditions is commonly referred to as ANO-2 Power Uprate.

The ANO-2 Power Uprate Licensing Report (Power Uprate Report) provides the results of the evaluations and analyses conducted to determine that ANO-2 can in fact operate acceptably and in compliance with applicable licensing requirements at the power uprated conditions. When approved by the NRC, ANO-2 will be authorized to operate at a Nuclear Steam Supply System (NSSS) power level up to 3026 megawatts thermal (MWt), which represents a 7.5% increase above the currently licensed core power rating of 2815 MWt. A 7.5% uprate would increase the unit's design gross electrical output to approximately 1048 megawatts electrical (MWe).

A 7.5% uprate was selected based on several factors. Since the unit's steam generators required replacement due to various corrosion related phenomena that had occurred over the years of operation, an economic decision was made to design the replacement steam generators to accommodate an increase in rated thermal power. It was determined, based on economics, that a power uprate of at least 6.5% would be required in order to recover the capital investment of larger replacement steam generators and other modifications that would be necessitated by the installation of larger steam generators. Scoping efforts were then initiated to explore whether an uprate greater than 6.5% was possible based on five criteria:

- satisfactory safety analysis results,
- satisfactory margins on all safety-related systems, structures and components,
- satisfactory margins for reactor vessel head Alloy 600 nozzles,
- acceptable additional cost above the cost to achieve a 6.5% uprate, and
- the ability of the replacement steam generators to support a higher uprate.

Based on the above criteria and the physical limitations of the replacement steam generators (i.e., height and interface requirements), a 7.5% uprate was determined to be the optimum level. The replacement steam generators accommodate a 7.5% uprate while anything less would limit the possibility of maximizing MWe production.

The ANO-2 Power Uprate Project scope is divided among Entergy Operations, Inc. (EOI), and CE Nuclear Power, LLC (formerly ABB-CE), a division of Westinghouse Electric Company, LLC. The CE Nuclear Power (CENP) scope includes the NSSS performance parameters, design transients, systems, components, accidents and nuclear fuel. The EOI scope included analyses such as the maximum hypothetical accident dose calculation and fuel handling dose calculations.

Some analyses have already been reviewed and approved by the NRC staff as part of the replacement steam generator effort. Where applicable, references will be made to the following three license amendments:

1) Containment Uprate License Amendment dated November 3, 1999 (2CAN119903), as supplemented by letters dated April 4, 2000 (2CAN040004), May 31, 2000 (2CAN050009), June 9, 2000 (2CAN060007), June 29, 2000 (2CAN060014), August 8, 2000 (2CAN080005), and August 16, 2000 (2CAN080010), and approved by the NRC in a safety evaluation dated November 13, 2000 (2CNA110002);

2) Containment Cooling License Amendment dated June 29, 2000(2CAN060003), as supplemented by letter dated October 4, 2000 (2CAN100004) and approved by the NRC in a safety evaluation dated November 13, 2000 (2CNA110003); and

3) Applicable Limits and Setpoints License Amendment dated November 29, 1999 (2CAN119901), as supplemented by letters dated January 26 (2CAN010008), May 17 (2CAN050005 and 2CAN050006), May 31 (2CAN050009) and August 4, 2000 (2CAN080004) and approved by the NRC in a safety evaluation dated September 29, 2000 (2CNA090002).

1.2 METHODOLOGY AND ACCEPTANCE CRITERIA

Westinghouse Topical Report WCAP-10623, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," (January 1983) was used as a guideline in preparation of the ANO-2 Power Uprate Report. Since its submittal to the NRC, the methodology has been used successfully as a basis for power uprate projects on over twenty pressurized water reactor (PWR) units, including Vogtle Units 1 and 2, Turkey Point Units 3 and 4, and Farley Units 1 and 2.

Additional guidance regarding the scope and content of an acceptable power uprate submittal was obtained from Licensing Topical Report NEDC-31897P-A, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," (May 1992). Although this topical report was written specifically for boiling water reactors, it contains useful information for PWR power uprate reports as well. SECY-97-042 (February 18, 1997), Section 3, "Power Uprate Review Process," cited both of these topical reports as documents the NRC should use to establish the basis for NRC review of power uprate submittals. Additional insights were derived from the NRC letter dated February 8, 1996, "Staff Position Concerning General Electric Boiling-Water Reactor Extended Power Uprate Program."

The limiting aspects of the nuclear steam supply systems (NSSS) and components were evaluated to determine the impacts of the increased power level on ANO-2. The analyses and evaluations were performed in accordance with the following acceptance criteria:

- The safety aspects of the plant that are affected by an increase in the thermal power level have been evaluated both for NSSS and balance of plant (BOP) systems.

- Evaluations and reviews were based on licensing criteria, codes and standards applicable to the plant at the time of the power uprate submittal (i.e., no change to the previously established licensing basis for the plant).

- Evaluations/analyses were performed using standard Westinghouse/CENP analysis methods for the Safety Analysis Report accidents and transients affected by power uprate.

- Plant modifications required to support power uprate were designed to applicable safety requirements, code and industry standards and station modification procedures and reviewed in accordance with 10CFR50.59.

- Plant systems and components impacted by the power uprate have been reviewed to ensure no significant increase in challenges to the safety systems.

- A review was performed to assure that an increased thermal power level continues to comply with the existing plant environmental regulations.

- A review was performed pursuant to 10CFR50.92(c) establishing that no significant hazards exist as a result of operation at the increased power level.

- Revisions to the ANO-2 Safety Analysis Report (SAR) will be made in accordance with 10CFR50.71(e).

1.3 ARRANGEMENT OF POWER UPRATE LICENSING REPORT

The objective of the power uprate effort is the increase in electrical output of ANO-2. Therefore, Section 2.0 of this report provides an overview of the BOP system and equipment changes needed to effect this output increase. However, the focus of the BOP discussion is on those structures, systems and components (SSCs) that interface with NSSS SSCs. The report also presents information to demonstrate that the replacement steam generators were designed and will operate acceptably at power uprate conditions. Most of the remaining sections discuss the various aspects of the NSSS to demonstrate that ANO-2 remains in compliance with licensing and design bases, criteria and requirements at power uprate conditions. Section 9 discusses various other topics such as high energy line break, electrical equipment qualifications, and impacts on the Operations Department. Section 10 of the report contains the environmental impact assessment. The report provides references to applicable SAR sections and indicates sections that have been identified for revision. Additional revisions may be identified as further reviews are performed.

1.4 TECHNICAL BASIS FOR SIGNIFICANT HAZARDS EVALUATION

This report provides the technical basis for the determination of no significant hazards evaluation supporting the changes to the operating license and technical specifications. The evaluation is provided in the attachment to the cover letter of this submittal.

1.5 CONCLUSION

The results of the NSSS analyses and evaluations demonstrate that the ANO-2 NSSS will operate acceptably and in compliance with applicable licensing requirements at the power uprate conditions.

END OF SECTION

2 BALANCE OF PLANT / NSSS INTERFACES

2.1 BOP OVERVIEW

This section provides information regarding balance of plant (BOP) structures, systems and components (SSCs) with an emphasis on those aspects of the design and operation of BOP SSCs that could affect the reactor and its safety features or other safety-related SSCs. The SSCs have been verified to be capable of functioning under power uprate conditions without compromising the nuclear safety of the plant under both normal operating and transient conditions.

The BOP SSCs have been evaluated for the impact of the 107.5% power uprate and either found acceptable, appropriately modified, or scheduled for modification. A list of modifications with power uprate considerations is provided in Table 2-2, which is ordered by the outage date of the modification. Refueling outage 2R14 has recently been completed; outage 2R15 will precede the first cycle at the uprated power level.

A brief history of megawatt electrical (MWe) changes is presented in Table 2-1 below. The historical information (i.e. prior to 2000) is not intended to be comprehensive but to list significant MWe changes.

Table 2-1

Cycle	Date	Change	MWt	MWe
1	1978	None	2815	958
10	1992	OSG first tube plugging during 2FO92*	2815	958
12	1997	OSG continuing tube plugging 2R12**	2815	928
15	2000	RSGs Installed; HP and LP Turbine modified 2R14	2815	978
16	2002	Power Uprate 2R15	3026	1048

* ANO Unit 2 – Forced Outage in 1992; ** ANO-2 – Twelfth Refueling Outage

Successive original steam generator(s) (OSG) tube plugging from 2FO92 up to 2R14 increasingly caused degraded OSG steam pressure and reduced MWe generation.

Changes in MWe output can vary not only with licensed reactor power but also with steam generator condition, turbine and Rankine cycle performance. The replacement steam generators (RSGs), turbine modifications and power uprate all contribute to regaining and improving the MWe output of ANO-2.

The RSG design had to include increased tubing surface area to accommodate power uprate. Additional steam generator tubing surface area was accomplished by increasing the diameter of the lower shell, and therefore the tubesheet, by four inches. The tubesheet diameter increase was a major factor that allowed for a greater number of tubes, and the tubing surface area in each

RSG is approximately 109,000 ft^2 compared to approximately 87,000 ft^2 in the OSGs, an increase of about 25% in tubing surface area. This surface area permits a 107.5% power uprate while maintaining prudent design for RSG thermal hydraulic parameters and adequate tubesheet structural strength. The RSGs were installed prior to Cycle 15. Power uprate is planned for Cycles 16 and beyond. Cycle 16 is scheduled to begin in April 2002.

In general, the BOP systems most affected by power uprate are those listed below. Not all sections in the SAR chapters are discussed in this report:

Report Section 2.2: SAR Chapter 8 - Electric Power

1. Grid stability
2. Main generator
3. Transformers
4. Emergency diesel generators
5. Alternate AC power source

Report Section 2.3: SAR Chapter 9 - Auxiliary Systems

1. Fuel pool system
2. Service water system
3. Ultimate heat sink
4. Containment cooling
5. Shutdown cooling

Report Section 2.4: SAR Chapter 10 - Steam and Power Conversion

1. Turbine
2. Main steam supply system
3. Water chemistry
4. Steam dump and bypass system
5. Condensate and feedwater
6. Emergency feedwater

Other BOP evaluations for power uprate are listed in Section 2.5 of this report. Since these BOP SSCs and topical areas do not directly impact the reactor or its safety features, they are listed only and not discussed further. However, each SSC and topical area has been evaluated for power uprate and is acceptable.

Table 2-2

ANO-2 Modifications with Power Uprate Considerations

Item	Change	Installation Outage
Moisture separator-reheater	Improved chevrons; improved heat exchanger bundles	2R12 2R13
Main condensers	Increased surface area	2R13
Hydrogen coolers	Replace with larger cooler	2R14
Stator water cooler piping	Direct supply of cooling water	2R14
Feedwater control system	Improved reactor trip override; system response improvements	2R14
Steam generators	Replacement has larger tubing surface area	2R14
Main generator	Rewound for 1,133 MVA	2R14
High pressure turbine	Replace complete high pressure assembly	2R14
Low pressure turbine	Replace four stages with more efficient blade and bucket design	2R14
Heater drain pumps	Three new pump stages; new motors; increase recirculation line size	2R15
Iso-phase buses	Increase cooling	2R15
Main transformers	Increase cooling	2R15
Stator water coolers	Replace with larger cooler	2R15

2.2 SAR CHAPTER 8 – ELECTRIC POWER

2.2.1 Grid Stability

Grid stability is discussed in SAR Section 8.2, in safety evaluations for the FSAR, and in the safety evaluation for Amendment No. 215 dated April 28, 2000. Grid stability has been evaluated for power uprate and the licensing basis document conclusions in these documents are unaffected.

2.2.1.1 Evaluation

The transmission system has been evaluated to ensure that the grid system remains stable for the upgraded generation level at ANO. Increasing the generation level results in a slight reduction in stability margins for the ANO units. However, this slight reduction in margins does not result in instability of the ANO units for the disturbances (faults or other single contingency events) expected on the transmission system. Offsite power systems will return to equilibrium without cascading trips of additional transmission lines, generators or other transmission equipment after these disturbances. Additionally, during such disturbances the offsite system will continue to supply the safety related buses with acceptable voltage levels (per NEMA standards) so motors can start and perform their required safety function.

Transmission studies are performed every two years to reconfirm the acceptability of the offsite power sources for ANO per our letter dated March 27, 1992 (1CAN039206). These studies ensure that the load on the offsite transmission system does not increase to levels which result in offsite voltage levels that are inadequate to power safety equipment during various transmission equipment outages (single contingency outages). The minimum offsite voltage levels obtained from these studies are evaluated to determine if they are at least as great as the voltage level required in the latest ANO plant degraded voltage analyses and appropriate subsequent calculations. The next transmission studies are scheduled for first quarter of 2001.

A recent transient stability study confirmed the stability of the offsite power system during summer peak conditions following the simultaneous loss of both ANO units. A generated output for the station was assumed that bounds the ANO-2 power uprate. This study indicated that a three phase fault near the ANO switchyard on any of the three 500 kV lines leaving ANO (cleared under normal breaker clearing time of 5 cycles) results in the ANO units and the system remaining stable for both existing and power uprate generation levels. The study indicates that, for both current and uprate generation levels, when either the Mablevale or the Pleasant Hill 500 kV line is already out of service, the ANO units are unstable following a three phase fault on the other of these two lines. To avoid this situation, procedures require that if one of these two 500 kV lines is out of service, the ANO unit generation is reduced to 1300 MW to prevent instability should a fault occur on the other of these two 500 kV lines. Additionally, to avoid unit instability during periods of minimum load conditions on the transmission system, ANO procedures limit the ANO-2 absorption levels to 200 MVAR.

2.2.1.2 Conclusion for Grid Stability

Grid stability studies have demonstrated that for power uprate the transmission grid remains stable. Safety related buses will be acceptably supplied by the offsite power sources following postulated transmission system disturbances.

2.2.2 Main Generator

The main generator is discussed in SAR Section 8.3.1.

During 2R14, the generator was reconditioned and upgraded for power uprate conditions. Inspections and testing during 2R13 of the main generator stator windings had confirmed the presence of stator bar water leaks and abrasions. Therefore, the stator was rewound for power uprate conditions.

The SAR section has been evaluated for power uprate and no changes are expected except for the following:

Starting in Cycle 16, the generator rating is increased:

Table 2-3

Main Generator Ratings

Item	Present and Cycle 15	New rating, Cycle 16 forward
Output (kVA)	1,047,300	1,133,334
Power factor (pf) (Rated)	0.90	0.95
Output (MW) at 0.90 pf	942.6	--
Output (MW) at 0.95 pf	994.9	1076.7

2.2.2.1 Evaluation

The generator is currently rated 1047.3 MVA at a 0.90 pf (lagging), which equates to 942 MW and 457 MVAR. Although the generator stator was rewound during 2R14, the generator ratings were not increased due to limitations in the stator water system. During Cycle 15, generation is a maximum of 980 MWe. From capability curves prior to rewind, the generator is limited during Cycle 15 to approximately 370 MVAR (lagging), which equates to 0.936 power factor. Based on discussion and correspondence with the Entergy Transmission and Distribution group, it is acceptable for ANO to restrict its power factor to 0.95 or higher at full generator capability.

Larger stator water coolers will be installed during 2R15, allowing the generator rating to be increased to 1133.3 MVA at 0.95 pf that equates to 1077 MWe and 353 MVAR. According to revised General Electric capability curves for the rewound generator, the generator is limited to approximately 370 MVAR (lagging) at 1052 MWe. This equates to a 0.94 power factor, which provides slightly more reactive capability than the 0.95 power factor restriction at full rated power.

2.2.2.2 Conclusion for Main Generator

The main generator was reconditioned and the stator rewound in 2R14. After the stator water coolers are replaced in 2R15, the main generator will be acceptable for power uprate.

2.2.3 Transformers

2.2.3.1 House Load Changes for Power Uprate Evaluation

House loads to operate ANO-2 are about 41 MWe. The loads are evaluated in summer peak conditions when cooling water systems, ventilation and air conditioning loads are largest. The evaluation projects that house loads for normal operation will increase slightly for power uprate. Accident loads are bounded by the analysis of record (AOR).

Normal loads are brought into ANO-2 via the unit auxiliary and startup transformers. These loads are classified by voltage levels: 6900, 4160 and 480 volts. Since no new facilities are planned, the evaluation assumed that any load increases would be due to motors.

The evaluation determined that there are no increases in the 6900-volt motor loads caused by the reactor coolant pumps and circulating water pumps. These are the only motors at this voltage level so there are no accident load increases on 6900-volt windings. Since these motors are not restarted after a loss of offsite power, there are no emergency diesel generator loads at the 6900-volt level.

The 4160-volt motor loads will increase by approximately 660 horsepower (hp) or about 0.5 MWe. All of this load increase is for normal operation only. As described further in this section, condensate and feedwater flow rates will increase for power uprate. The evaluation projected an increase of about 160 hp total for three condensate pumps. Similarly, the two heater drain pump loads will increase by a total of about 300 hp. Finally, the service water pump loads will increase by a maximum of 200 hp during summer operation of normal loads. The motors for the condensate, heater drain, and service water pumps are supplied by the 4160-volt stage of the transformers. The feedwater pumps are turbine driven so there are no electrical load changes due to them.

There is no increase in accident loads on the 4160-volt windings above the AOR.

There is no increase in 480-volt motor loads, normal or accident above the AOR.

2.2.3.2 Main Transformer Evaluation

The main transformers are discussed in SAR Section 8.3.1.1.1. The main transformers have been evaluated for power uprate.

The ANO-2 main transformer bank consists of three single-phase units, each rated at 333.5 MVA for a total three-phase capacity of 1000.5 MVA. However, the entire power output of the electrical generator does not flow through the main transformers. At power uprate conditions, approximately 4% ($\approx$ 41.5 MWe) of the power will flow into the unit auxiliary transformer to serve ANO-2 house loads.

An engineering evaluation indicates that the main transformers' carrying capacity improves by up to 13% with additional cooling. This evaluation is conservatively based on hot spot limitations and not average winding rise temperature increase. Additional main transformer cooling will be installed during 2R15.

The SAR section is unaffected except for the following:

The generator MVAR output spans from a 0.95 pf lagging (353 MVAR) to 200 MVAR leading.

SAR statements regarding capacity will be revised for the additional cooling to be installed in 2R15.

2.2.3.3 Unit Auxiliary Transformer Evaluation

The unit auxiliary transformer is discussed in SAR Section 8.3.1.1.2. The equipment has been evaluated for power uprate. The unit auxiliary transformer is a three-winding transformer having an overall capacity of 58.5 MVA with a capacity of 32.8 MVA on the 6900-volt winding and a capacity of 25.7 MVA on the 4160-volt winding.

The present load on the 6900-volt winding is approximately 32 MVA based on readings made during the summer. This is the most heavily loaded winding but, as stated above in Section 2.2.3.1, power uprate will not increase the load on this winding.

The present load on the 4160-volt winding is about 19.9 MVA or approximately 77% of its rating. As stated in Section 2.2.3.1 of this report, this is the winding affected by power uprate but it is also the winding with reserve capacity.

The SAR section is unaffected except for the following:

The expected increase due to power uprate is about 0.56 MVA at 0.90 pf and brings the 4160-volt winding loading to about 80% of capacity.

2.2.3.4 Startup Transformer #3 Evaluation

Startup Transformer #3 is discussed in SAR Section 8.3.1.1.2. The equipment has been evaluated for power uprate. The SAR section is unaffected.

Startup Transformer #3 is identical to the ANO-2 unit auxiliary transformer in size and rating. This transformer is used for plant start-up and shutdown. It is also one of the two required offsite power sources needed to provide plant power for ANO-2 accident mitigation. Upon a plant trip (generator lockout), one of the condensate pumps is shed. If there is a safety injection actuation system signal with the trip, the two heater drain pumps are also tripped along with the main chiller. Therefore, for accident mitigation, Startup Transformer #3 has significantly less duty than the unit auxiliary transformer. Consequently, the minor load added by power uprate will not be significant or cause any transformer winding overload.

2.2.3.5 Startup Transformer #2 Evaluation

Startup Transformer #2 is discussed in SAR Section 8.3.1.1.2. The equipment has been evaluated for power uprate. The SAR section is unaffected.

Startup Transformer #2 is a three-winding transformer having an overall capacity of 45 MVA with a capacity of 25 MVA on the 6900-volt winding and a capacity of 21 MVA on the 4160-volt winding. It is the second (delayed) of the two required offsite power sources needed to provide plant power for ANO-2 accident mitigation. It is shared by ANO-1 and ANO-2. Except for specially analyzed conditions, Startup Transformer #2 is kept in pull-to-lock and buses are manually loaded with restrictions on the loadings (safety buses only). For those specially analyzed conditions, an automatic load shedding scheme limits the total loads which can be auto-transferred to Startup Transformer #2. Procedures also restrict the loads to be auto-transferred to Startup Transformer #2 so that the rated capacity of the transformer is not exceeded.

2.2.3.6 Conclusion for Transformers

There are no accident load increases at any voltage level above the AOR. The accident load AOR is bounding.

There are no 6900- and 480-volt motor load increases. The motor load increases at 4160 volts are about 0.5 MWe on a winding level that has adequate margin.

When the additional cooling is installed, the main transformers will be adequate for power uprate conditions.

No changes are required for the unit auxiliary transformers for power uprate.

No changes are necessary for Startup Transformer #3 for power uprate.

No changes are necessary for Startup Transformer #2 for power uprate.

2.2.4 Emergency Diesel Generators

The emergency diesel generators (EDGs) are discussed in SAR Section 8.3.1.1.7. The EDGs have been evaluated for power uprate. The SAR section is unaffected.

2.2.4.1 Evaluation

Loads for accident mitigation are bounded by the AOR. Therefore, no changes are planned for the EDGs.

Power uprate will cause an increase in decay heat and therefore an increase in EFW pump flow rate. However, engineering evaluations for power uprate determined that no change to the EFW pump motor loads is required since the AOR that provides design input in the EDG analysis used bounding pump pressures and flow rates. Therefore, the EDG AOR encompasses the power uprate load.

Diesel generator loads during accident conditions continue to be within the design and licensed ratings of this machine. Also, the response time for diesel generator starting and loading is not impacted by power uprate. In fact, the current loading calculation is conservative since the changes identified are all load reductions. Diesel generator load reductions are the result of a modification to the containment cooler fans and the removal of the electrical loads for the sodium hydroxide pumps.

The blade pitch was reduced during 2R14 on the containment cooler fans in order to lower the horsepower loads imposed on motors. Without the modification, the motors would have operated over their nameplate horsepower rating for a short time during accident conditions with the replacement steam generators. Containment fan operation for accident conditions is described in the containment uprate licensing amendment submittal dated November 3, 1999 (2CAN119903), and its supplements. The current AOR retains the previous value for the containment cooling fan motors since this value is conservative.

The loads for the sodium hydroxide pumps have been removed from the AOR as a result of a modification implemented during refueling outage 2R13 that replaced the sodium hydroxide system with trisodium phosphate-filled baskets, a passive system.

2.2.4.2 Conclusion for Emergency Diesel Generators

The EDGs are adequate for power uprate. The modifications discussed above have resulted in margin enhancement.

2.2.5 Alternate AC Power Source

The alternate AC (AAC) power source is discussed in SAR Section 8.3.3. The AAC has been evaluated for power uprate. The SAR section is unaffected.

2.2.5.1 Evaluation

The AAC diesel generator (i.e., station blackout diesel generator) loads are minimally affected by power uprate and are bounded by the AOR. The AAC diesel generator loads during accident conditions will continue to be within the design and licensed ratings of this machine, which has a continuous duty rating of 4400 kW at 4160 volts, 60 hertz, 0.80 pf. The AAC generator is sized to carry the largest load from the either of two ANO-1 safety buses or either of two ANO-2 safety buses.

As discussed in the preceding section, the current accident loading calculation is conservative since the modification to the containment cooler fans actually results in a load reduction and margin enhancement.

2.2.5.2 Conclusion for Alternate AC Power Source

The AAC power source is adequate for power uprate.

2.3 SAR CHAPTER 9 – AUXILIARY SYSTEMS

2.3.1 Fuel Pool System

The fuel pool system is discussed in SAR Section 9.1.3. The fuel pool system has been evaluated for power uprate. The SAR section is unaffected.

2.3.1.1 Evaluation

With the increase in decay heat due to power uprate considered, the AOR is bounding and fuel pool temperature is still maintained ≤ 150°F for a full core discharge and a service water system inlet temperature ≤ 85 °F. The AOR is bounding for power uprate for make-up to the fuel pool from service water, the assured make-up source. The AOR is bounding for power uprate for fuel bundle thermal hydraulic analysis, as is the SAR dose rate figure.

2.3.1.2 Conclusion for the Fuel Pool System

The fuel pool system is acceptable for power uprate.

2.3.2 Service Water System

The service water (SW) system is discussed in SAR Section 9.2.1. The system and the auxiliary cooling water system have been evaluated for power uprate. The SAR section has been evaluated for power uprate and was revised as a result of a 2R14 modification as discussed below.

2.3.2.1 Evaluation

As previously designed, the stator water coolers received auxiliary cooling water (ACW) after the water passed through and cooled the main chillers. In doing so, the water was heated prior to reaching the stator water coolers. This design had not been a problem due to the degraded OSGs that caused the output of the electric generators to be reduced. ACW is non-essential cooling that is isolated for accident conditions.

For power uprate, the stator water coolers need their own direct source of cooling water. This was accomplished in 2R14 by changing the main chiller condenser cooling supply from ACW to non-essential service water. The stator water coolers are still supplied by ACW but without the water first passing through the main chillers. The SAR has been revised to reflect the 2R14 modification. As stated in the introduction to Section 2.2.2.1, the stator water coolers will be replaced with larger capacity coolers in 2R15.

2.3.2.2 Conclusion for the Service Water System

With the revised ACW and SW flows, adequate cooling for power uprate conditions is provided for SSCs cooled by the ACW and SW systems.

No changes are required for essential SW, evaluated at power uprate conditions, which provides cooling during accident conditions.

2.3.3 Ultimate Heat Sink Evaluation

The ultimate heat sink (UHS) is discussed in SAR Section 9.2.5. The UHS has been evaluated for power uprate and the SAR section is unaffected.

2.3.3.1 Evaluation

Power uprate will not adversely impact the emergency cooling pond (ECP) temperature and inventory during design basis events based on analyses that demonstrate that the heat load rejected to the ECP will not increase above what is already assumed in the AOR. Why this is so is discussed in our letter dated August 16, 2000 (2CAN080010), which provided supplemental information regarding the containment uprate licensing amendment request.

2.3.3.2 Conclusion for the Ultimate Heat Sink

UHS capabilities are adequate for power uprate.

2.3.4 Containment Cooling Evaluation

The containment cooling system (CCS) is discussed in SAR Section 9.4.5. The CCS has been evaluated for power uprate. The SAR section is unaffected.

2.3.4.1 Evaluation

Normal cooling is provided by the main chillers that supply chilled water to the non-safety-related containment chilled water cooling coils (CCWCCs). The CCWCCs were replaced during 2R14 with larger coil banks that improve heat transfer capability. The pitch of the CCS fans was reduced during 2R14 but the improved heat transfer capability of the CCWCCs more than offset the reduction in CCS air flow. Although the RSGs operate at a higher temperature than the OSGs, the key is not the temperature but heat loss from the steam generators and related piping. Since the insulation system for the RSGs is improved over that provided for the OSGs, the amount of steam generator heat loss is decreased. The insulation on related piping is unchanged and adequate. This reduced loss, combined with improved CCWCC performance, should result in a lower containment bulk average air temperature than Cycle 14 or earlier.

Containment cooling during accident conditions is provided by the safety-related containment service water cooling coils (CSWCCs). The CCS fans and the CSWCCs are addressed in our letter dated June 29, 2000, "Technical Specification Change Request for ANO-2 Containment Cooling" (2CAN060003), and its supplement dated October 4, 2000 (2CAN100004). Service water flow to the CSWCCs is unchanged.

2.3.4.2 Conclusion for Containment Cooling

With replacement CCWCCs and the modifications to the CCS fans, the CCS is now adequate for power uprate for both normal and accident conditions. Adequate accident condition cooling is addressed in the containment uprate submittals cited.

2.3.5 <u>Shutdown Cooling</u>

This NSSS fluid system is addressed in Section 4.1 of this report.

2.4 SAR CHAPTER 10 - STEAM AND POWER CONVERSION

To accomplish power uprate, several parameters in the condensate, feedwater and steam supply system will change as shown below in Table 2-4. Values are best estimate, used to illustrate magnitude of change.

Table 2-4
Cycle Comparisons

Item	Cycle 14 (degraded OSG)	Cycle 15 (RSGs installed)	Cycle 16 forward (power uprate)	Δ Cycle 16 minus Cycle 15
Licensed MW thermal (MWt)	2815	2815	3026	211
MW electrical (MWe)	928	978	1048	70
Main steam flow rate-100%, lbm/hr	12,720,000	12,514,541	13,660,596	1,146,055
Main steam pressure at throttle valves, psia	769	907	900	-7
Condensate flow rate, lbm/hr	9,103,000	8,954,963	9,765,977	811,014
Feedwater pump speed, rpm	3145 to 3175	3575	3775	200
Feedwater pump suction pressure, psig	605-612	622	596	-26
Heater drain pump flow rate, lbm/hr	3,582,000	3,559,578	3,894,619	335,041

2.4.1 Turbine

The turbine is described in SAR Section 10.2.

2.4.1.1 Design Bases

The design bases of the turbine are described in SAR Section 10.2.1. This equipment and its subsystems have been evaluated for power uprate. The SAR section is unaffected for power uprate except for the following:

With the normal configuration of the steam dump and bypass system (SDBS), the turbine generator can accept a load rejection of 49% power without a reactor trip. See Section 2.4.4 below.

2.4.1.2 Evaluation

The turbine generator system is described in SAR Section 10.2.

Concurrent with the RSG installation in 2R14, the high pressure turbine steam path components and four stages of the low pressure turbine were replaced with newer designs that support plant operation during Cycle 15 and optimize plant performance for power uprate. All replacements are by the original equipment manufacturer, General Electric Co.

Turbine-generator auxiliary systems have been evaluated and are adequate for power uprate.

Trip functions and devices have been evaluated and adequately protect the high and low pressure turbines at power uprate conditions.

The SAR section is unaffected for power uprate except for the following:

Design flow at valves-wide-open (VWO) conditions for power uprate is 14,214,115 lb/hr at 1.18"/2.05" Hga backpressure (dual-pressure condenser) at initial steam conditions of 896.0 psia at 1194.9H btu/lb and zero makeup. The VWO design ensures that the turbine maximum guaranteed rating (MGR) is met. For rated conditions, the flow is 13,660,596 lb/hr at 900 psia. See Table 2-4 for changes from Cycles 14 and 15.

2.4.1.3 Turbine Missiles

Turbine missiles are discussed in SAR Section 10.2.3. Turbine missiles have been evaluated for the turbine modifications. The SAR section is unaffected by the turbine modifications needed to achieve power uprate.

2.4.1.4 Conclusion for the Turbines

The high and low pressure turbines have been redesigned explicitly for power uprate.

Turbine-generator auxiliary systems are adequate for power uprate.

The turbine modifications needed to achieve power uprate do not change the probability of turbine missiles, which is acceptably low.

Trip functions and devices are adequate for power uprate.

2.4.2 Main Steam Supply System

The main steam supply (MS) system is described in SAR Section 10.3.

Components in the MS system have either been replaced with new equipment designed for power uprate, modified previously with power uprate considered, or evaluated as satisfactory for power uprate.

2.4.2.1 Evaluation

The objectives of the MS system are described in SAR Section 10.3. Power uprate does not change any of these objectives and the SAR section is unaffected.

The MS system description and system operation are discussed in SAR Section 10.3.2. System operation has been evaluated for power uprate.

The SAR description and system operation are unaffected by power uprate except for the following:

Blowdown following a postulated pipe rupture in the main steam line is now initially limited by integral RSG steam nozzle flow restrictors. The integral RSG flow limiting nozzles were not present in the OSGs. Their presence necessitates a change to the main steam line break (MSLB) analysis. See Section 7 of this report.

The safety evaluation of the MS system is described in SAR Section 10.3.3. The safety evaluation has been evaluated and is unaffected for power uprate.

The turbine stop valves have been evaluated for the dynamic effects of higher flow rates. The evaluation demonstrated that the AOR is bounding.

2.4.2.2 Moisture Separators-Reheaters

The moisture separator-reheaters (MSRs) are discussed in SAR Section 10.2 and 10.3. Power uprate does not affect this discussion. Heating and moisture removal at power uprate conditions were included in the design for the 2R12 and 2R13 modifications made to the MSR chevrons and heater bundles. The SAR section is unaffected; MSR safety valves orifices may be resized during 2R15 but this change is below the SAR level of detail.

2.4.2.3 Conclusion for Main Steam Supply System

Safety-related and non-safety-related components and sub-systems in the MS system as described in the SAR are adequately sized and designed to perform their intended function at power uprate for normal, transient and accident conditions.

2.4.3 Water Chemistry Evaluation

Water chemistry is discussed in SAR Section 10.3.5.

The ANO-2 primary and secondary chemistry programs, including sampling, have been evaluated for power uprate and found to be suitable for protection of the RSGs and for power uprate conditions of flow, pressure and temperature. The effects of the increase in primary and secondary water volumes were included in the evaluation.

The primary sampling system is suitable for the expected small increase in primary temperatures ($\approx$ 2–5° F) due to power uprate.

The original steam generators were affected by various components in the feedwater system that contained copper. ANO management has been aggressive in removing all major copper components from the unit. All feedwater heaters were replaced several cycles before and are free of copper-bearing parts. The replacement components are suitable for power uprate. Removing copper from all fluid systems in the condensate, feedwater, steam supply, extraction and drain systems will help ensure that for the remaining life of ANO-2, the RSGs can supply the steam pressure and flow required for uprate conditions.

2.4.3.1 Primary-side Chemistry

Primary side chemistry uses lithium hydroxide for pH and corrosion control and hydrogen (H_2) for oxygen scavenging.

The SAR section is unaffected except for the following:

CE Nuclear Power Co., LLC (CENP, formerly ABB-CE) has evaluated a concern with RSG and core designs regarding deposition of nickel on the core. Based on CENP recommendations, a new lithium strategy is being incorporated starting in Cycle 15.

2.4.3.2 Secondary-side Chemistry

Power uprate will result in higher secondary side flow rates but the current system is adequate to meet the needed injection rates. The ANO chemistry department has software obtained from EPRI that models the secondary side and serves as a simulator for predicting chemistry changes based on specific inputs such as temperature and flow. These parameter changes are programmed into the simulator software. The secondary sampling system is suitable for the expected small increase in secondary temperatures due to power uprate.

The SAR section discussion of secondary-side chemistry is unaffected for power uprate.

2.4.3.3 Conclusion for Water Chemistry

Water chemistry is adequate for power uprate.

2.4.4 <u>Steam Dump and Bypass System Evaluation</u>

The steam dump and bypass system (SDBS) is discussed in SAR Section 10.4.4.

The SAR section is unaffected by power uprate except for the following:

During normal operation with three turbine bypass valves and two downstream atmospheric dump valves under automatic control, the SDBS load rejection capacity is 49% instead of 51%.

2.4.4.1 Conclusion for the Steam Dump and Bypass System

No modifications to the steam dump and bypass system are required for power uprate; however, the control system will be adjusted as necessary during 2R15 to accommodate the higher steam flow and higher reactor heat load. The steam dump and bypass control system (SDBCS) is discussed in Section 4.2 of this report.

After implementing changes to the steam dump and bypass control system, the SDBS will adequately respond to transient conditions under power uprate conditions. The 51% capability was determined by SDBS component capacities, not a specific event causing a load rejection. The decrease in load rejection capability to 49% does not significantly change the SDBS operational feature of avoiding some reactor trips and reducing secondary side safety valve openings.

2.4.5 Condensate and Feedwater Systems

The condensate and feedwater system (CFWS) is discussed in SAR Section 10.4.7. The effect of power uprate on the SAR discussion is discussed below.

The hydraulics and thermodynamics associated with changes in the CFWS have been thoroughly analyzed utilizing modeling techniques that are improved over those available during original design. The original condensate, feedwater, steam supply, extraction and drain systems are generously sized and the new velocities and pressure drops will be well within the acceptable ranges stated in technical manuals and equipment manufacturers' recommendations. General Electric supplied the final heat balance(s), which is an integrated mass and energy model but not a hydraulic analysis. Hydraulic modeling is performed separately.

The new hydraulic modeling provides good insights into the operation of the condensate and feedwater system. Recommendations for procedure changes such as when, during power ascension and descension, the initial and second heater drain pumps should be placed into service are drawn from the modeling analysis. The hydraulic model also provides improved insights and parameters for NSSS transient analyses. The NSSS transient analysis provides parameters required to tune the steam dump and bypass system (SDBS), used to mitigate the transients and reduce the incidence of reactor trips that would otherwise be caused by the transient.

2.4.5.1 Design Bases

The condensate and feedwater system design bases are discussed in SAR Section 10.4.7.1. The condensate and feedwater system was originally designed for a single condensate train to be capable of maintaining generation at an 80% level. Therefore, during normal operation with two trains, each train is not heavily loaded. Although power uprate will reduce the single train capability from 80 to 65%, each train maintains excellent capability for normal two train operation. Power uprate will not affect the SAR section except for the following:

The five stages of low pressure and two stages of high pressure feedwater heaters are capable of 130% of normal flow rather than 160%.

2.4.5.2 Evaluation

Each analysis at power uprate conditions for condensate, feedwater and heater drain pumps includes evaluations of the following pump parameters: required total head and flow, horsepower, and net positive suction head (NPSH) available versus NPSH required. Analyses demonstrate that the following equipment is adequate for uprate conditions without modification:

(1) condensate pumps and motors; (2) feedwater pumps and feedwater pump turbines; and (3) feedwater heaters, heater extraction lines and valves and heater drains. The heater drain pumps and motors will be modified as described below in Section 2.4.5.4. After the heater drain pump modifications, these parameters will be adequate for each pump type.

At uprated conditions, the feedwater control system will satisfy higher flow demand by increasing the speed of the turbine-driven feedwater pumps (Table 2-3) to force higher flow to the RSG. (The feedwater control system is discussed in Section 4.2 of this report) The higher feedwater pump speed will cause a reduction in feedwater pump suction pressure. This causes a reduction in the condensate system differential pressure head measured from the condenser hot well to the feedwater pump suction pressure point. This reduced differential pressure head in turn causes the system resistance head curve (a function of the square of the volumetric flow rate, gpm^2) to extend to a new equilibrium point where the system resistance curve crosses the performance curve for condensate pumps in parallel operation. This is a point of higher flow rate through the existing condensate piping and low pressure feedwater heaters.

While the suction pressure at the feedwater pumps is lower in Cycle 16 forward than in Cycle 15, conversely the discharge pressure of the feedwater pumps will increase. The increased pressure is required to force the higher flow rate through existing feedwater piping, high pressure feedwater heaters, feedwater control and isolation valves and into each steam generator.

Although the feedwater suction pressure will decrease due to the power uprate, the pressure is well above the suction pressure alarm and trip points and no changes to the suction header alarm and trip setpoints are required.

SAR tables will be revised for these parameters to reflect power uprate conditions.

Equipment and component trips in the CFWS for main steam line break and feedwater line break are described in our letter dated November 3, 1999 (2CAN991103), "Containment Building Design Pressure Increase to 59 Psig," and its supplements.

2.4.5.3 Feedwater Heaters

The feedwater heaters are discussed in SAR Sections 10.4.7.1 through 10.4.7.3.

The feedwater heaters have been evaluated for power uprate conditions for extractions, design pressures, pressure drops, and drain, tube and nozzle velocities. In addition, feedwater heater vibration characteristics and shell-side relief valve capacities have been evaluated.

The feedwater heaters' manufacturer performed evaluations using its own computer programs. The manufacturer states that the programs were developed utilizing its own experience over many years and incorporating industry standards such as the following:

1. Standards for Closed Feedwater Heaters by the Heat Exchange Institute (HEI)

2. Standards of the Tubular Exchange Manufacturers Association, Inc. (TEMA) and
3. Proprietary data of Heat Transfer Research, Inc. (HTRI).

The SAR sections are unaffected except for the following:

Plant operation with one train of low pressure heaters isolated is a very infrequently performed evolution (IPE) that would be caused by a feedwater heater tube rupture or severe tube leakage. However, the evaluation showed that a 70% power level could be achieved without challenging the tube side flow capacity of the low pressure heaters. The power level is limited by shell-side relief valve capacity to 65%. The 65% value is a reduction from the previous power level of 80% for this IPE.

2.4.5.4 Feedwater Heater Drain System and Extractions to Feedwater Heaters

Feedwater heaters drains are described in SAR Section 10.4.7.2.

<u>Extractions to feedwater heaters</u>

The extractions have been evaluated for power uprate using the Bechtel TE605 Flash program. The program evaluates single- and two-phase flow in the extraction and drain lines and whether the heater level control valves are adequately sized.

The SAR discussion is unaffected by power uprate.

<u>Low pressure heater drains</u>

The low pressure heater drains and heater control valves have been evaluated for flow rates and pressure drops and single- and two-phase flow for power uprate.

The SAR discussion is unaffected by power uprate.

<u>High pressure heater drains and heater drain pumps</u>

For each train, drains from a moisture separator-reheater and two high pressure feedwater heaters (and the highest pressure low pressure feedwater heater) are collected in the heater drain tank that supplies suction to a heater drain pump. The heater drain pump injects the recovered drain fluid into the condensate piping just upstream (suction) of the feedwater pump.

A recirculation control configuration, consisting of piping and a control valve, branches from the heater drain pump discharge back to the heater drain tank to maintain a steady level in each tank. The recirculation line size for each train will be increased and new control valves installed.

As described above, power uprate will result in lower feedwater pump suction pressure. Feedwater pump suction pressure is the single parameter which strongly affects the interaction between the condensate, heater drain and feedwater pumps. This in turn causes a decrease in the total head across the heater drain pump and an increase in flow rate since its suction pressure (at the heater drain tank) remains essentially the same as the Cycle 15 pressure whereas its discharge pressure is reduced. Analysis has demonstrated that the existing heater drain pumps and motors are inadequate for power uprate. Three of ten stages of the heater drain pumps will be replaced and larger horsepower heater drain pump motors installed in 2R15.

The SAR figure showing the recirculation line size will be revised as will be the SAR table for the replacement heater drain pumps data.

2.4.5.5 Condensate Pumps

The SAR section is unaffected regarding the condensate pumps except for the following:

The condensate pumps are not 50% capacity pumps. The percent capacity description will be removed from the SAR since it is not meaningful information. Prior to RSG and power uprate, three of four pumps were normally placed in service during the power ascension to 100% power. This will not change for power uprate.

2.4.5.6 Conclusion for the Condensate and Feedwater Systems

The condensate pumps and motors are adequate for power uprate for total head, flow, horsepower and NPSH.

The feedwater pumps and feedwater turbines are adequate for power uprate for total head, flow, NPSH, turbine capacity and steam flow to the turbines.

The feedwater heaters were generously sized for the original plant configuration and are acceptable for power uprate, including vibration characteristics and shell-side relief valve capacities.

The extraction lines are adequately sized for power uprate.

The low pressure feedwater heater drains and level control valves are adequately sized for power uprate.

The high pressure feedwater heaters drains and level control valves are adequately sized for power uprate.

With the changes described above, the heater drain pumps and motors will be adequate for power uprate for total head, flow, horsepower and NPSH.

2.4.6 Emergency Feedwater System

The emergency feedwater (EFW) system is described in SAR Section 10.4.9. The system has been evaluated and the SAR section is unaffected by power uprate.

2.4.6.1 Evaluation

Although power uprate will cause an increase in decay heat, engineering evaluations for power uprate determined that no change to the EFW pump flow rate is required since the AOR is bounding. Calculations demonstrate that the EFW pumps can provide the minimum flow rate necessary to support the safety analysis flow rate assumptions. There is sufficient net positive suction head (NPSH) available at 1000 gpm, which is well above the evaluated flow rate. NPSH margins are even greater if the EFW flow assumed in safety analyses is considered.

The design parameters and criteria for the EFW have been evaluated for power uprate. The condensate transfer system provides the preferred source of water to the EFW pump suctions, although the service water system (SWS) is the assured safety-related source. The increase in RCS volume due to the RSGs does not adversely impact the EFW system. The increased demand for condensate requirements for cooldown can be met with the current system configuration and operation. No physical changes are necessary to the system. The assured source of feedwater, service water, has been evaluated and has adequate capacity for power uprate conditions.

The current technical specifications require a minimum volume in the condensate storage tanks (CSTs). A technical specification change request dated January 27, 2000 (0CAN010004), allows alignment to either the CSTs or the qualified condensate storage tank (QCST), which is already in use for Unit 1. The change request clarifies that the volume in the storage tank aligned to EFW is required to provide enough water for 30 minutes of operation. This allows the operator sufficient time to switch to the service water system, if necessary. The minimum required level for the QCST in the proposed technical specification is determined based on the requirements for both ANO-1 and ANO-2 and the ANO-2 7.5 % power uprate. No additional changes to the CST requirements are needed for power uprate. The proposed TS also allows alignment to the CSTs, for which the minimum required volume is conservatively left unchanged. This volume is adequate for power uprate conditions with no change to the technical specifications.

2.4.6.2 Conclusion for the Emergency Feedwater System

The EFW system is adequate for power uprate.

2.5 OTHER BOP EVALUATIONS FOR POWER UPRATE

The following BOP SSCs and topics have been evaluated for power uprate. Since these BOP SSCs and topical areas do not directly impact the reactor or its safety features, they are only listed and not discussed further. However, the following SSCs and topical areas have been evaluated for power uprate and are acceptable:

1. Pressure and Temperature (PT) calculations for impacted SSCs
2. Civil reconciliation of PT calculations
3. Component engineering reconciliation of PT calculations
4. Flow accelerated corrosion reconciliation of PT calculations
5. Fire barrier reconciliation of PT calculations
6. I&C reconciliation of PT calculations
7. Mechanical reconciliation of PT calculations
8. Nuclear safety analysis reconciliation of PT calculations
9. Structural reconciliation of PT calculations
10. Condenser, Cooling Tower and Circulating Water
11. Main steam pressure drop calculation
12. Heater drain tank sizing
13. Solenoid operated valves
14. Air operated valves
15. Auxiliary feedwater performance
16. MSR relief valve capacity
17. Steam generator blowdown capacity
18. Blowdown heat exchanger replacement
19. CO_2, N_2, and H_2 systems
20. Domestic water
21. Instrument air system
22. Radwaste and primary sampling systems
23. Fire detection and halon systems
24. Turbine building sump, oily waste, auxiliary building drains and auxiliary building sump
25. Auxiliary building HVAC
26. Turbine building HVAC
27. Isophase bus
28. Short circuit and load flow evaluations
29. Protective relaying and metering
30. Main Generator Exciter

END OF SECTION

3 NUCLEAR STEAM SUPPLY SYSTEM PARAMETERS

This section describes the set of key plant parameters established as a basis for the uprating evaluations. These values are contained in the Groundrules document for Cycle 16 and are used as inputs for various analyses and evaluations. The Groundrules document serves as an interface between ANO-2 and the fuel vendor for defining plant equipment parameters which reflect current and/or expected plant characteristics that are used in the reload analyses. Only certain key parameters from the Groundrules are discussed here.

3.1 INTRODUCTION

Key parameters for ANO-2 include the following:

- Reactor power
- Axial shape index
- Reactor vessel inlet temperature
- Pressurizer pressure
- Reactor coolant flow rate
- Feedwater temperature
- Steam/feedwater flow rate
- Steam pressure

Defining these parameters provides the framework for determining more detailed plant parameters (such as heat rejection rates, mass and energy release rates, radiation source terms, and emergency core cooling system parameters) and for evaluating the design limiting accidents and transients. The uprated parameters are also used to determine the impact on functional design requirements and structural integrity of NSSS systems and components and of those systems which interface with the NSSS.

During refueling outage 2R14, ANO-2 replaced the original steam generators. The power uprate parameters are based on operation with the replacement steam generators (RSGs) and on the various analyses which were performed to support the RSGs as well as the analyses performed specifically for power uprate.

3.2 INPUT PARAMETERS AND ASSUMPTIONS

Table 3-1 is a summary of the key operating conditions for the primary side and the corresponding analytical assumptions, where applicable. Cycle 16 will be the first cycle at the uprated power conditions. Cycle 15, which is the first cycle with the RSGs, and Cycle 13 data are provided for comparison with the power uprate conditions.

3.3 DISCUSSION OF PARAMETERS

The power level for the uprating was set at 3026 MWt. This is approximately 7.5% higher than the current NSSS power rating of 2815 MWt. As presented in this report, various analyses have been performed to demonstrate that the reactor may be operated safely at this uprated power.

The limits on axial shape index remain unchanged from Cycle 15.

RCS flow rate was calculated using RCS pressure drops and equations for the RCP performance curves. The expected volumetric flow rate will decrease insignificantly from Cycle 15 to Cycle 16 due to the effect of higher temperatures on density and flow velocities. The Cycle 16 flow rate is well within the analytical values for maximum and minimum RCS flow.

The anticipated full power core inlet temperature for Cycle 16 is 551° F. This is a small increase from the expected value for Cycle 15 of 549° F. The analytical minimum and maximum values for T_{cold} will remain the same as for Cycle 15. The expected RCS hot leg temperatures for Cycles 15 and 16 (604° F and 609° F, respectively) are below the original design temperature for T_{hot}, which was 612.5° F. (Because the actual RCS flow for Cycle 1 was higher than anticipated, the actual Cycle 1 T_{hot} was less than design.)

Pressurizer pressure, the analytical minimum and maximum as well as nominal, will remain the same for Cycle 16 as for Cycle 15. Neither the RSGs nor the power uprate affect the operating pressurizer pressure.

Feedwater temperature will increase from Cycle 15. Heat balance calculations predict a feedwater temperature of 453° F for Cycle 16 at 100% power. This is up from 444° F at 100% power for Cycle 15. Cycle 15 at 100% power essentially corresponds to Cycle 16 at 93% power.

For Cycle 16 the required steam/feedwater flow at 100% power is an increase of approximately 9.3% over Cycle 15 flows.

The design steam pressure for Cycle 15 and beyond is 900 psia at the turbine. Cycle 16 steam pressures were projected using best estimate fouling. Best estimate fouling is based on the Westinghouse fleet experience with present-day water chemistry control. (Westinghouse is the RSG supplier.)

Similar to feedwater temperature, Cycle 15 steam pressure is closely related to the 93% power uprate steam pressure. Since steam pressure varies inversely with power level, the Cycle 16 steam pressure would be expected to be lower than Cycle 15. However, to obtain best point operating pressure (900 psia at the turbine), for Cycle 16 forward T_{cold} will be increased to 551° F from the 549° F value for Cycle 15. Therefore, the steam pressure at the control valve for the two cycles is essentially the same. Steam generator dome pressure will increase for Cycle 16 to 944 psia from 938 psia for Cycle 15.

3.4 ACCEPTANCE CRITERIA

The values for these parameters are deemed acceptable if the various safety analyses meet their acceptance criteria. For example, many analyses demonstrate that adequate margin exists to departure from nucleate boiling (DNB) and linear heat generation limits. The initial conditions, i.e., reactor power, pressurizer pressure, reactor coolant temperature at the core inlet, core power distribution, etc., are selected for the various analyses based on the core operating limits allowed by the limiting conditions for operation. The core operating limits are generally bounded by the values assumed in the analyses. The core operating limits are defined as a set of initial conditions for which the specified acceptable fuel design limits are not violated as a result of the most rapid decrease in thermal margin caused by an anticipated operational occurrence.

Additionally, the plant structural analyses and NSSS performance analyses ensure that the operating conditions are properly analyzed.

Table 3-1

Summary of Key Operating Conditions and Corresponding Analytical Assumptions[1]

Parameter	Cycle 13 (Reference)	Cycle 15 (RSG Cycle)	Cycle 16 (Pwr Uprate)
Core Rated Power, MWt			
Actual	2815	2815	3026
Analytical Assumption	2900	2900/3087	3087
Axial Shape Index			
> 20% power	+/- 0.3	+/- 0.3	+/- 0.3
Primary Coolant Flow Rate			
Operating (gpm)	334,047	352,000	351,800
Operating (lb/hr)		$132*10^6$	$132*10^6$
Minimum/T.S. 3.2.5, lb/hr [2]	$108.4*10^6$	$120.4*10^6$	$120.4*10^6$
Minimum Analytical (gpm)	289,800	315,560	315,560
Maximum Analytical (gpm)	355,200	386,400	386,400
Core Inlet Temperature, °F			
Operating	545	549	551
Minimum Technical Specification / 3.2.6	542	542	542
Minimum Analytical (Includes Uncertainty)	540	540	540
Maximum Technical Specification / 3.2.6	554.7	554.7	554.7
Maximum Analytical (Incl. Uncertainty)	556.7	556.7	556.7
Pressurizer Pressure, psia			
Operating	2200	2200	2200
Minimum Technical Specification / 3.2.8	2025	2025	2025
Minimum Analytical (Includes Uncertainty)	2000	2000	2000
Maximum Technical Specification / 3.2.8	2275	2275	2275
Maximum Analytical (Incl. Uncertainty)	2300	2300	2300
Core Outlet Temperature, °F			
Operating	603	604	609
Feedwater Temperature, °F			
Operating	452	444	453
Feedwater Flow/Exit Steam Flow, lbm/hr			
Operating	$12.7*10^6$	$12.5*10^6$	$13.66*10^6$
Steam Pressure at Control Valve, psia			
Operating	767	907.5	908.0
Design		900	900

Notes:

(1) The actual NSSS operating point for ANO-2 will probably vary slightly from this estimated operating point due to differences in actual RCS flowrate versus the estimated flowrate, actual steam generator performance versus calculated performance, etc. The actual operating point will be chosen to optimize the plant efficiency and NSSS operating conditions while staying within the range of operating conditions analyzed.

(2) An RCS flow of 120.4×10^6 lbm/hr is approximately equivalent to an RCS flow of 322,000 gpm at reactor coolant pump inlet conditions of 2200 psia and 553°F.

END OF SECTION

4 NUCLEAR STEAM SUPPLY SYSTEMS

This section discusses the impact of the power uprate on the functional design requirements and structural integrity of the NSSS. The nuclear steam supply systems and components were verified to be capable of performing their intended functions at uprated conditions. The control systems associated with the NSSS (such as the pressurizer level and pressure control systems) were also evaluated. Aside from the replacement steam generators and associated changes, no NSSS modifications are necessary for power uprate. Minor control system adjustments will be made to accommodate the changes in operating conditions.

4.1 NSSS FLUID SYSTEMS

The following NSSS fluid systems were reviewed:

- the reactor coolant system (RCS),
- the chemical and volume control system (CVCS),
- the safety injection system (SIS), and
- the shutdown cooling system (SDCS).

The power uprate evaluations considered (1) the functional and design objectives of these systems, (2) the impact (if any) of power uprate on the objectives, and (3) the ability of each system to carry out its design functions under power uprate conditions. Each system was found to be capable of fulfilling its intended functions under power uprate conditions with no modifications.

The structural integrity of these systems has been evaluated for the impact of power uprate and found to be acceptable. The structural integrity of the reactor coolant system is discussed in greater detail in Section 5 of this report. Also, Section 8.4 discusses the effect of neutron fluence on the reactor vessel.

4.1.1 Reactor Coolant System

The reactor coolant system is described in SAR Section 5.1. The RCS is a two-loop pressurized-water design with U-tube steam generators and four reactor coolant pumps (RCPs). Overpressure protection is provided by two code safety valves on the pressurizer. There is no power-operated relief valve (PORV).

For the purposes of this evaluation, the reactor coolant system was considered to include the following volumes and components:

- Reactor vessel including upper head and vent connection;
- Reactor vessel internals;
- Pressurizer including surge line, ASME code safety valves, spray piping, spray valves and vent connections;

- RCS piping (hot legs, RCP suction piping and RCP discharge piping);
- Safety injection nozzles, charging and letdown nozzles, surge line and spray line nozzles;
- Primary side of the steam generators (inlet nozzles, inlet plenum, inactive and active regions of the U-tubes, outlet plenum and outlet nozzles); and
- Reactor coolant pumps.

The key functions of the RCS are as follows:

1. Act in conjunction with other systems to provide sufficient cooling of the core during all normal plant evolutions and anticipated operational occurrences to preclude significant core damage.

2. Transfer heat generated in the core to the main steam system via the steam generators.

3. Transfer heat to the shutdown cooling system during shutdown conditions.

4. Provide a barrier against fission product release to the environment, including during low temperature conditions.

5. Enhance the fission process by functioning as a neutron moderator.

6. Serve as a medium for boric acid, which functions as a neutron poison for reactivity control.

7. Allow for natural circulation flow following a loss of RCP flow, including adequate RCP coastdown characteristics.

8. Maintain subcooled conditions by controlling pressurizer pressure via heaters and spray flow.

RCS operating conditions at uprated power are discussed in Section 3 of this report. RCS pressure will remain the same as in Cycle 15 and flow rate will decrease insignificantly. However, the power uprate will result in higher neutron flux and higher RCS temperatures. The limiting component for the effects of higher flux is the reactor vessel. Section 8.4 of this report addresses the effects of neutron fluence on the vessel pressure-temperature limits. The higher temperatures associated with the power uprate are still within the bounds of the original design temperatures of 650° F for the RCS and 700° F for the pressurizer and an original design T_{hot} of 612° F. For Cycle 16, T_{hot} will be approximately 609° F; Cycle 15 T_{hot} is approximately 604° F.

Sufficient core cooling under power uprate conditions is verified by various plant transient and safety analyses discussed in other sections of this report, as is the general acceptability of power uprate changes for the RCS. The acceptability of the power uprate changes for normal operating transients is verified by the review of the NSSS control systems described in Section 4.2 of this report.

The natural circulation capacity of the RCS is expected to be acceptable for power uprate conditions. The ability to perform a natural circulation cooldown without drawing a steam bubble in the reactor vessel head has been evaluated for power uprate conditions. The calculation demonstrated that such a cooldown can be accomplished. The emergency feedwater (EFW) system provides the heat sink during natural circulation conditions. The effect of power uprate on EFW is discussed in Section 2.4 of this report.

The effect of power uprate and the replacement steam generators on the low temperature overpressure protection (LTOP) analysis was discussed in correspondence dated December 21, 1999 (2CAN129907), which requested a change to the technical specification bases discussion. The limiting event for LTOP was revised to be the energy addition transient (RCP start) instead of the mass addition transient. The power uprate will also increase the neutron fluence in the reactor vessel, which will affect the pressure-temperature limits. See Section 8.4 for a discussion of fluence for Cycle 16.

The other key functions are not affected by the power uprate. No system modifications (other than the replacement steam generators) were required to accommodate the power uprate.

The quench tank, described in SAR Section 5.5.11, is designed to receive and condense the discharges from the pressurizer safety valves during anticipated operational occurrences and prevent the discharge from being released to the containment. The sizing of the quench tank and its operational limits were verified to be acceptable for power uprate conditions. The SAR description is revised for Amendment 16 to incorporate results of analyses which were done for the replacement steam generators and the power uprate.

4.1.1.1 Alloy 600 Material Evaluation

A current issue facing all PWR owners is the primary water stress corrosion cracking (PWSCC) of Alloy 600 material exposed to primary coolant, especially at elevated temperatures. The RSGs are installed with Alloy 690 tubes, which are more resistant to most types of corrosion, but there are other applications of Alloy 600 in the RCS that remain. These include the control element drive mechanism (CEDM) nozzles, in-core instrumentation nozzles and vent lines in the reactor vessel head; RTD nozzles in the RCS piping; and instrumentation nozzles and heater sleeves in the pressurizer.

The effect of power uprate temperatures on the RCS Alloy 600 material relative to the potential for cracking has been evaluated. The pressurizer will continue to operate at the same temperatures and thus the potential for PWSCC for the pressurizer will not increase with power uprate. The cold leg nozzles are only slightly affected because T_{cold} will increase from 549° F to 551° F. At this temperature PWSCC has not been an industry concern.

The increase in susceptibility for the Alloy 600 components in the hot legs and the reactor vessel has been evaluated. Any cracking in Alloy 600 would be axially oriented, and short axial cracking does not threaten the structural integrity of the affected components. The failure mechanism has not changed, and extensive leakage would have to occur over a period of years prior to break. Therefore, PWSCC is not considered to be a significant safety concern.

Increasing T_{hot} only increases the susceptibility of the nozzles by decreasing the potential time to cracking; it has no impact on the safety significance of the cracking. Also, the temperature for the ANO-2 CEDM nozzles is generally less than T_{hot} due to the amount of bypass cooling flow.

In the EPRI Histogram of Reactor Vessel Head Nozzle Assessments, the power uprate increased the susceptibility ranking for ANO-2 CEDM nozzles. However, ANO-2 still remains in the lowest assessment group in the EPRI Histogram. In addition, enhanced visual Generic Letter 88-05 inspections are performed during every plant outage to detect any leakage in the Alloy 600 nozzles in the reactor vessel head.

4.1.2 Chemical and Volume Control System

The chemical and volume control system is described in SAR Section 9.3.4. The CVCS consists of four basic subsystems as follows:

- letdown,
- charging,
- boron addition, and
- reactor makeup water.

Each of these subsystems consists of various valves, tanks, pumps, and instrumentation which perform the functions of the CVCS. Its primary design function is to maintain reactor coolant system inventory and control RCS chemistry. Other RCS support functions include serving as a part of the RCS pressure boundary, aiding in containment isolation, providing auxiliary pressurizer spray, and providing for RCP seal bleedoff flow. The CVCS functions were evaluated for performance under power uprate conditions and were verified to be capable of performing those functions without system modifications. The requirements for the boric acid makeup tank volume and concentration will change, as discussed below.

The key functions of the CVCS are as follows:

1. Maintain the chemistry and purity of the reactor coolant during normal operation and during shutdown, including crud burst cleanups.

2. Maintain RCS inventory by compensating for coolant contraction or expansion resulting from changes in the reactor coolant temperature and for other normal coolant losses or additions.

3. Control boron concentration in the RCS to maintain acceptable control element assembly (CEA) configuration, compensate for reactivity changes associated with burnup and major changes in coolant temperature and xenon concentrations, and provide the required shutdown margin when the reactor is subcritical.

4. Provide auxiliary pressurizer spray for operator control of pressure during the final stages of shutdown and to allow pressurizer cooling.

5. Provide a means of measuring RCS leakage during steady-state operation.

6. Provide capability for functional testing of the safety injection system check valves.

7. Include instrumentation for measurement of fission product activity.

8. Monitor and collect RCP seal controlled bleedoff.

9. Provide capability of testing the RCS pressure boundary.

10. Provide one of several alternate makeup sources to the RCS during a loss of decay heat removal event.

The chemistry and purity of the RCS will be maintained under power uprate conditions. Although the power uprate will cause an increase in the generation of activity in the RCS, this change is considered to be negligible. The higher power level may necessitate changes to the RCS chemistry requirements, but any changes will be well within the capabilities of the CVCS. Primary chemistry is discussed in Section 2.4 of this report.

Because of the higher RCS inventory due to the replacement steam generators, the wider range of RCS temperature changes will result in a greater volume change than before power uprate. This has been evaluated and found to be within the capability of the CVCS. The design requirement to provide for letdown or makeup for a 75° F/hr heatup or cooldown was reviewed for power uprate and concluded that, except for a momentary deficit (that is accommodated by an acceptable drop in pressurizer level and later excess capacity), this requirement is met. No CVCS modifications are required to successfully meet the increased volume requirements.

The projected RCS letdown temperature for power uprate is very close to the original operating assumption for the CVCS; the original assumed letdown temperature was 550° F while the power uprate value for T_{cold} is expected to be 551° F. Therefore, the increased temperature due to power uprate has essentially no effect on the operation of the system.

The ability of the CVCS to provide the required shutdown margin has been evaluated and found to be adequate for power uprate, with a minor exception to the boric acid makeup (BAM) tank. SAR Section 9.3.4 explains that the letdown loop of the CVCS is not required to achieve cold shutdown. In this situation (a cooldown without letdown), the total makeup that can be added to the RCS is limited by the total RCS shrink during cooldown. Because of the core design for power uprate, the minimum required boron concentration in the BAM tank will need to be increased to accommodate this situation. This has necessitated changes to Technical Specification 3.1.2.7 and 3.1.2.8, which are discussed in the attachment to the cover letter which submitted this report (2CAN120001). The new cooldown without letdown analysis uses the same methodology as the analysis that produced the existing technical specification figure (that analysis was reported in correspondence supporting Technical Specification Amendment No. 82). Although not specifically required for the power uprate, enhancements were made in the new analysis which resulted in increased conservatism. For example, pressurizer level was conservatively increased to the maximum allowed by technical specifications.

The other key functions of the CVCS have been reviewed and are not impacted by the power uprate. No system modifications are required to accommodate the power uprate.

4.1.3 Safety Injection System

The safety injection system is described in SAR Section 6.3, "Emergency Core Cooling System." The SIS consists of two active subsystems and one passive subsystem:

- the high pressure safety injection (HPSI) system (active),
- the low pressure safety injection (LPSI) system (active), and
- the safety injection tanks (SIT) (passive).

The components of the high pressure and low pressure subsystems are arranged in two separate and redundant trains, each of which is capable of performing 100 percent of the required system design functions. There are four safety injection tanks, one tank connected to each reactor coolant system cold leg. In the event of a large break loss-of-coolant accident (LOCA), the safety injection tanks function to reflood the core following blowdown and to provide cooling until the active subsystems begin to inject cooling water.

The key functions of the SIS are as follows:

1. Provide core cooling during a LOCA. (Limit peak fuel rod cladding temperature in order to maintain a coolable core geometry and minimize release of fission products to the containment.)

2. Maintain core subcritical during a LOCA.

3. Limit hydrogen generation caused by zircaloy/steam reactions.

4. Provide sufficient long term cooling to ensure core temperatures are maintained at an acceptably low level.

5. Compensate for reactivity addition following a main steam line break.

The adequacy of the safety injection systems is verified by the various safety analyses performed in support of the power uprate and Cycle 16. The ability of the HPSI system to perform its design function during sump recirculation has been verified for power uprate conditions. The formal review concluded that the objectives are either not affected or are acceptable based upon specific analysis or by qualitative evaluation. A more detailed review of the ECCS analyses is presented in Section 7.1 of this report.

The minimum system performance requirements are not changing from Cycle 15 to Cycle 16. No system modifications are required for power uprate.

4.1.4 Shutdown Cooling System

The shutdown cooling system is described in SAR Section 9.3.6. The SDCS consists of two heat exchangers, two pumps and associated valves and piping. The system functions to remove RCS heat (both decay heat and sensible heat) at temperatures less than 300° F and to achieve and maintain refueling temperatures. During normal power operation, the two pumps in the shutdown cooling system serve as the low pressure safety injection pumps of the SIS and the two heat exchangers are aligned to the containment spray system.

The key functions of the SDCS are as follows:

1. Removes decay heat from the core and sensible heat from the RCS when the RCS is below 300° F and 300 psia. The ANO-2 Technical Specifications require achieving cold shutdown (less than 200° F) from normal operating temperatures within 36 hours under certain conditions.

2. Maintains refueling temperatures of 135° F or less in the RCS while providing flow adequate to ensure uniform boron concentration.

The ability of the SDCS to achieve cold shutdown (less than 200° F) in 36 hours has been verified. This evaluation is comparable to that described in SAR Section 9.3.6.6 (Amendment 15) with considerations for the effects of the replacement steam generators, the power uprate, and system changes since the original evaluation. The assumed service water temperature was changed to 121° F, a conservative assumption which is consistent with the ultimate heat sink analysis.

The SDCS remains adequate to maintain refueling temperatures and a uniform boron concentration in the RCS.

The SDCS functions have been reviewed and are either not impacted by the power uprate or are acceptable based upon specific analysis or by qualitative evaluation. No system modifications are required to accommodate the power uprate.

4.2 NSSS CONTROL SYSTEMS

SAR Section 7.7 discusses various control and instrumentation systems that are not essential for the safety of the plant. Changes for power uprate required for the pressurizer level control system (PLCS), feedwater control system (FWCS), and steam dump and bypass control system (SDBCS) are discussed in the sections below. Changes to the core operating limit supervisory system (COLSS) are discussed in Section 7.4 of this report.

Power uprate does not affect the design of the anticipated transient without scram (ATWS) systems (the diverse scram system/diverse turbine trip (DSS/DTT) and the diverse emergency feedwater actuation system (DEFAS)). Although not expected, adjustments to the setpoints may be necessary based on actual operating conditions for Cycle 16.

Some minor scaling adjustments will be needed for the excore and incore detector circuits so that they read correctly at the higher power levels. Similarly, the plant computer/SPDS software will need minor adjustments for the new operating conditions (higher power levels, steam flows, etc.). The current plant process for modifications is adequate to make such adjustments.

The remaining control systems discussed in SAR Section 7.7 are not impacted by power uprate and are not discussed further:

- pressurizer pressure control system (PPCS) (RCS pressure is not changing for power uprate);
- control element drive mechanism control system (CEDMCS); and
- shutdown cooling instrumentation and controls.

The performance of the basic NSSS operational control systems (PPCS, PLCS, FWCS, and SDBCS) was evaluated in steady state operation and design basis control system maneuvering transients for the new operating conditions associated with power uprate. This evaluation verified that certain changes to the NSSS control systems would ensure proper NSSS dynamic performance such that the number of reactor trips is minimized for the load maneuvering transients and certain equipment malfunctions, and post-trip responses are acceptable.

Table 4-1 lists the operational objectives of these NSSS control systems. Although the objectives are identified specifically for individual control systems, the systems work together to meet all of the objectives.

The maneuvering transients are listed in Table 4-2. These were simulated using a best estimate simulation code with ANO-2 power uprate data, including nominal plant operating conditions, and considering beginning and end of cycle core conditions.

The revised NSSS control systems setpoints will provide proper control system performance for the power uprate operating conditions. With respect to plant safety, the NSSS response to load maneuvering transients and equipment malfunctions is bounded by the plant safety analyses.

4.2.1 Pressurizer Level Control System

The pressurizer level control system is only affected by an adjustment which will be made to an input from the RCS T_{avg} vs. level program. This is required since T_{avg} is increasing in Cycle 16. The remaining setpoints associated with level control functions (letdown valve control, charging pump auto start and stop, heater cutoff, etc.) were used in this evaluation and found to be acceptable.

4.2.2 Feedwater Control System

The increased power level in Cycle 16 requires an adjustment to the main feedwater pump speed demand curve. (The feedwater system is discussed in Section 2.4 of this report.) To maintain the desired steam pressure at the turbine control valve at 100% power, the main feedwater flow will be increased. The increased demand is still well within the design capacity of the main feedwater pumps. The feedwater pump speed requirements for Cycle 16 may be adjusted slightly based on actual Cycle 16 operating experience.

The anticipated Cycle 16 pump speed requirements were incorporated into the NSSS control system evaluation and found to be acceptable. Also, beginning of cycle (BOC) and end of cycle (EOC) core conditions were considered. Transients were simulated using a best estimate simulation code after the computer code basedeck was updated with Cycle 16 and RSG parameters.

Starting in Cycle 15, reactor trip and turbine trip transients were evaluated for a modified reactor trip override (RTO) logic. The objective of the RTO mode modification is to maximize refill of the steam generators post-trip while limiting overcooling of the reactor coolant system (RCS). The FWCS modified RTO mode responds by ramping the main feedwater regulating valve closed at a predetermined rate, setting the main feedwater pump speed to the minimum speed setpoint, and setting the bypass feedwater valve to a preset position based upon the selected RCS average temperature.

4.2.3 Steam Dump and Bypass Control System

The steam dump and bypass control system (SDBCS) is provided to improve plant availability by fully utilizing the dump/bypass valve capacity to remove NSSS thermal energy. This is achieved by the selective use of turbine bypass and atmospheric dump valves to avoid unnecessary reactor trips and to prevent the opening of secondary side safety valves whenever this can be averted by the controlled release of steam. The valves are positioned based on an error signal determined by a comparison between actual steam pressure and a steam pressure setpoint generated as a function of the NSSS load (steam flow). (The steam dump and bypass system is discussed in Section 2.4 of this report.)

The SDBCS will be recalibrated to accommodate the increased 100% steam flow and the change to the header pressure vs. power relationship based on analyses of maneuvering transients. The maneuvering transients were evaluated using a best estimate simulation code with ANO-2 plant specific data. Projected nominal plant operating conditions were assumed for these transient evaluations. All of the transients were evaluated at the uprated power level with the RSGs (Cycle 16 conditions). BOC and EOC core conditions were considered.

The evaluation demonstrated that SDBCS pressure control was adequate to maintain RCS T_{avg} and to control RCS shrinkage. For the reactor trip and turbine trip cases, the pressurizer heaters remained covered and the pressurizer pressure remained well above the safety injection system actuation setpoint. Therefore, under uprated power conditions the SDBCS will continue to perform its design functions.

Table 4-1

NSSS Control Systems Operational Objectives

Pressurizer Pressure Control System

- Maintain RCS operating pressure.
- Yield acceptable pressure response to normal system volume changes during load changes.
- Ensure that the minimum pressure observed during normal operations is above the setpoint of the safety injection actuation system and that the maximum pressure is below the high pressurizer pressure trip.

Pressurizer Level Control System

- Compensate for changes in coolant volume during load changes.
- Prevent the draining of the pressurizer as a result of a reactor trip.
- Limit the water volume to minimize the energy release during a LOCA.
- Prevent the uncovering of the heaters by the out-surge of water following load decreases (10 percent step decrease and 5 percent per minute ramp decrease).
- Prevent the water level from reaching the safety valve nozzles following a loss of load.

Feedwater Control System

- Automatically control the feedwater flow rate from 0% power to 100% power to each steam generator to ensure satisfactory steam generator downcomer water level during design basis maneuvering transients as well as during steady state operation.
- Provide automatic control of the feedwater flow rate to the steam generators following a reactor trip without overcooling the NSSS.

Steam Dump and Bypass Control System

- Provide automatic removal of the RCS energy in a controlled manner following a unit trip.
- Permit small turbine load rejections of up to 74 percent original power rating without opening the main steam safety valves or tripping the reactor (up to 49 percent at uprated power when the two upstream atmospheric dump valves are in the normal, closed position).
- Provide a means of manually controlling reactor coolant temperature during plant normal heatup and cooldown when the condenser is available.
- Automatically control main steam pressure during hot standby.

Table 4-2

NSSS Transients Evaluated

- Reactor trips from 100%, 80%, 60%, and 20% power
- Turbine trips from 100%, 80%, 60%, and 25% power
- Load rejection from 100% to 75% power
- Load rejection from 100% to 60% power

- Reactor trip with loss of main feedwater
- Loss of one of two running feed pumps at 100% power
- Heater drain pump trip at 100% power
- Condensate pump trip with auto-start of idle pump at 100% power
- Condensate pump trip without auto-start of idle pump at 100% power

- 15% to 50% power ramp at 5%/minute with two condensate pumps and one main feed pump running
- 50% to 100% power ramp at 5%/minute with two main feed pumps, three condensate pumps, and one heater drain pump
- 100% to 15% power ramp at 5%/minute
- 100% to 50% step load decrease
- 10% step load decrease from 100% and 25% power
- 10% step load increase from 90% and 15% power
- Loss of stator water cooling runback 100% to 27% load decrease in two minutes

- Main feed regulation valve fails open at 100% power
- Main feed pump recirculation valve fails open with and without auto-start of idle condensate pump at 100% power
- Steam dump and bypass system valve fails open at 70% and at 40% power
- Steam dump and bypass system valve fails shut at 80% and at 50% power

END OF SECTION

5 NSSS COMPONENTS

The structural integrity of the reactor coolant system (RCS) has been verified for power uprate conditions. This section describes the analyses and assessments of all RCS components that determined the effects caused by design condition changes.

5.1 INTRODUCTION

The structural integrity analyses assessed the impact of the design condition changes by making comparisons to quantifiable limits. For example, RCS major component nozzle and support structural analyses produced loads on the system due to normal operation, earthquake and pipe break conditions. These loads were combined according to regulatory guidelines and were used to calculate stresses. The stresses were then compared to clearly defined ASME stress limits to demonstrate that the structural integrity of the components was maintained. Therefore, the results were system loads and the quantifiable limits were allowable stresses. The new design basis analyses employed some basic methodology changes. These changes are summarized below.

Sections 5.2 and 5.3 discuss reactor vessel internals, fuel design and control element assembly qualifications. Sections 5.4 through 5.7 discuss the main coolant loop, the pressurizer and the surge line. Included in the main coolant loop is the main coolant piping itself, the replacement steam generators (RSGs), the reactor coolant pumps and motors, and supports for the reactor vessel, the steam generators, and the reactor coolant pumps. Section 5.8 discusses the tributary lines.

5.1.1 ANO-2 Reactor Description

ANO-2 incorporates a pressurized water reactor (PWR) with two reactor coolant loops. The reactor core is composed of 177 fuel assemblies and 81 control element assemblies (CEAs). Each fuel assembly, which provides for 236 fuel rod positions, consists of five guide tubes welded to spacer grids and is secured at the top and bottom to end fittings. The guide tubes each displace four fuel rod positions and provide channels that guide the CEAs over their entire length of travel. At selected core locations, the central guide tubes of the fuel assembly house incore instrumentation. The fuel is low enrichment UO_2 in the form of ceramic pellets encapsulated in Zircaloy tubes that form a hermetic enclosure.

The reactor coolant enters the reactor vessel through four cold leg nozzles (two nozzles per reactor coolant loop), flows downward between the reactor vessel wall and the core support barrel, passes through the flow skirt where flow distribution is equalized and then into the lower plenum. Reactor coolant flows upward through the lower support structure, through holes in the core support plate, and enters the bottom of each fuel assembly through holes in the lower end fitting. Within the fuel assembly, coolant flows upward, parallel to the fuel rods, passing through a series of spacer grids (one Inconel grid and eleven Zircaloy grids), experiencing heatup due to heat transfer from the fuel rods. Reactor coolant exits the fuel assembly through the upper end fitting, passes through the fuel alignment plate and the upper guide structure, then exits the reactor vessel through two hot leg nozzles (one nozzle per reactor coolant loop).

Long term reactivity control is accomplished by adjustment of soluble boron within the reactor coolant and by the inclusion of fixed burnable neutron absorber at selected fuel rod locations. Short term reactivity control is accomplished by repositioning the CEAs or by tripping the CEAs. The ANO-2 CEAs consist of five Inconel tubes (called "CEA fingers"), each loaded with a stack of cylindrical poison pellets. The CEA fingers are secured at the top end in a fixed array by a structure that is called the "CEA spider." The CEA spider holds four of the CEA fingers in a square array that matches the four outer guide tubes of the fuel assembly, while the fifth finger is secured at the center of the array matching the center guide of the fuel assembly. The spider of each CEA is attached to an extension shaft. Controlled motion of each CEA is accomplished by a CEA drive mechanism (CEDM), a magnetic drive mechanism that moves the extension shaft (with CEA) up or down. The CEAs are tripped upon interruption of electrical power to the CEDM magnet coils.

5.2 REACTOR VESSEL INTERNALS STRUCTURAL ANALYSIS

The reactor vessel internals addressed in this section comprise both core support and internal structures. These structures are illustrated in Figure 5-1. Core support structures include the lower support structure (LSS), core support barrel (CSB) and upper guide structure (UGS) components. Internal structures include the core shroud, CEA shroud and instrument support components. The reactor vessel internals support and orient the fuel assemblies, CEAs, and incore instrumentation and guide the flow of reactor coolant through the reactor vessel. The reactor vessel internals also absorb static and dynamic loads and transmit these and other loads to the reactor vessel flange. The internals are designed to safely withstand the forces due to deadweight, handling, pressure differentials, vibration and seismic acceleration and LOCA loads. The fuel and CEAs are addressed in Section 5.3 of this report.

5.2.1 Introduction

This section evaluates the impact of revised thermal, hydraulic, seismic and pipe break input data on the Level A+B (normal operating plus upset condition) and Level D (faulted condition) structural evaluations of reactor vessel internals (RVI) components documented in the analyses of record (AOR). The impact of the revised hydraulic input data on the ability of the holddown ring to provide adequate RVI hold down force was also evaluated.

The revised thermal input reflects changes in temperature distribution associated with the 7.5% power uprate. The revised hydraulic input, in the form of hydraulic loads and pressure differentials acting on RVI components, envelopes the possible range of measured flow rates associated with both the replacement steam generator and the current fuel assembly hardware. The revised seismic input, comprising seismic loads and moments on RVI components, reflects installation of the replacement steam generators and current fuel assembly mass as well as modeling refinements to the seismic analysis. The revised pipe break input, also comprising loads and moments on RVI components, reflects the design basis replacement of main loop pipe breaks with branch line pipe breaks, and also reflects revised hydraulic input as well as modeling refinements to the pipe break analysis.

The Level A+B evaluation was performed in accordance with criteria defined in the 1973 draft of Subsection NG for Section III of the ASME Boiler and Pressure Vessel Code, per Section 4.2.2 of the ANO-2 SAR. Level D design criteria are not included in the 1973 draft of Subsection NG, but were obtained from Appendix F of the ASME Code as directed in SAR Section 4.2.2.

5.2.2 Methodology Used for Evaluation

All RVI components were evaluated as core support structures in accordance with criteria defined in SAR Section 4.2.2, as revised to adopt leak-before-break methodology, which eliminates main coolant pipe breaks from the design basis.

Critical Level A+B stress intensities in the RVI components were obtained from the AOR. The revised thermal, hydraulic and seismic [operating basis earthquake (OBE)] input data was compared with that used in the AOR. Fuel assembly weights were also compared. For those components where the revised input data is encompassed by the AOR input data, the AOR stresses were left as is. For those components where the revised input data is more limiting than the AOR input data, stresses were recalculated using the AOR methodology in combination with the revised input data. In some cases, AOR methodology was enhanced to address input data that was previously unavailable (e.g., for the calculation of thermal stresses as discussed below).

At the time of the AOR, detailed thermal information was unavailable, and thermal stresses were generally not calculated. For the replacement steam generators and power uprate, detailed thermal loadings on the RVI components were calculated. To facilitate incorporation of the new information using AOR methodology, these detailed thermal loadings were generated in the form of linearized and local temperature gradients, from which secondary and peak stress values are calculated. This approach to generating thermal stresses is appropriate if the product of the modulus of elasticity and the coefficient of thermal expansion is reasonably constant over the range of temperatures being considered. As a conservative measure, these secondary thermal stresses were added directly to the primary or primary plus secondary stress intensities calculated in the AOR to obtain revised primary plus secondary stress intensities. Local thermal stresses were considered in the fatigue evaluation described below.

All Level A+B stress intensities were evaluated against criteria defined in the 1973 draft of Subsection NG for Section III of the ASME Code, per Section 4.2.2 of the SAR. These criteria include limitations on primary membrane, primary membrane plus bending, and primary plus secondary stress intensities of 1 x S_m, 1.5 x S_m, and 3 x S_m, respectively.

A general fatigue evaluation of the RVI components was performed. This evaluation considered both the fatigue curve provided in the 1971 edition of the ASME Code (circa AOR), which was limited to 10^6 cycles and the extended fatigue curve adopted by later editions of the ASME Code. A low-cycle fatigue evaluation was also performed for those components in which local thermal stresses exceed 3 times S_m.

Critical Level D stress intensities in the RVI components were obtained from the AOR. The revised seismic [safe shutdown earthquake (SSE)] and pipe break loads were combined and compared with the AOR SSE and pipe break loads. For those components where the revised input data is encompassed by the AOR input data, the AOR stresses were left as-is. For those components where the revised input data is more limiting than the AOR input data, stresses were recalculated using the AOR methodology in combination with the revised input data.

All Level D stress intensities were evaluated against criteria defined in Appendix F of the ASME Code, per Section 4.2.2 of the SAR. These criteria include limitations on primary membrane and primary membrane plus bending stress intensities of 2.4 x S_m and 3.6 x S_m, respectively.

The holddown ring exerts a downward force on the CSB and UGS upper flanges; maintaining them in a clamped configuration to prevent rocking and sliding of the CSB and UGS assemblies relative to one another and to the reactor vessel. Sliding margin is defined as the ratio of the lateral (frictional) component of the net hold down load over the lateral hydraulic input load. The net hold down load includes the holddown ring load, dead weight and fuel spring loads, and the vertical hydraulic input load. Rocking margin is defined as the ratio of the moment generated by the net hold down load over the hydraulic input moment. A rocking or sliding margin of 2.0 is considered to be adequate for 4-pump operation, while 1.5 is adequate for the other operating conditions. For this evaluation, rocking and sliding margins were calculated using the revised hydraulic input loads and moments, in combination with holddown ring loads derived using the most recently obtained field measurement data for holddown ring deflection.

5.2.3 <u>Results of Evaluation</u>

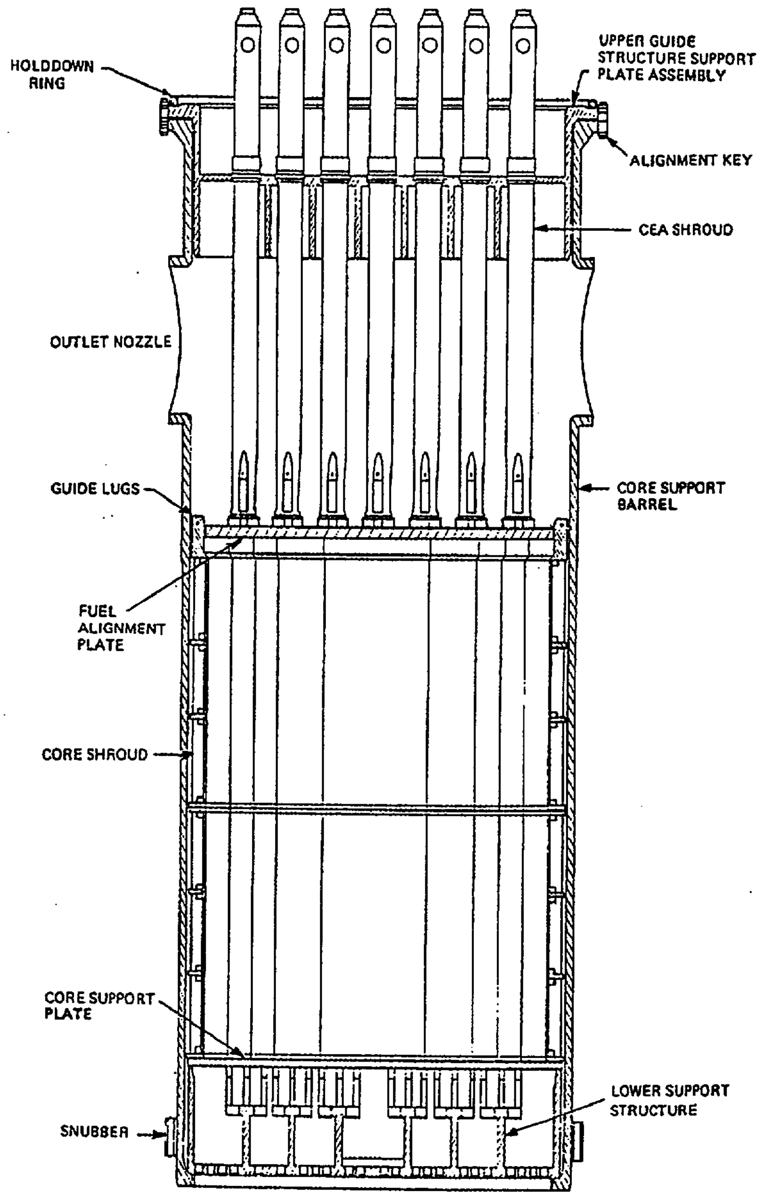
The results of this evaluation are presented in Table 5-1 below. Stress intensities in the RVI components resulting from the incorporation of revised input loads associated with RSG and power uprate were shown to satisfy design criteria for both normal operating-plus-upset and faulted design conditions. These results are summarized in the form of structural margins in Table 5-1; the positive nature of the margins demonstrates satisfaction of design criteria:

Table 5-1

Minimum Structural Margins for RVI Components

	Component	Minimum Structural Margin (%) Level A + B	Level D
CSB Components	Upper Flange	18.99	78.36
	Cylinders	8.01	35.27
	Lower Flange	57.26	80.60
	Snubbers	49.92	79.47
	Snubbers at Shell	53.99	80.83
	CSB-to-LSS Flexure	83.06	1.02
LSS Components	Cylinder	59.81	67.74
	Columns	46.12	1.11
	Beams	55.86	20.45
	Core Support Plate	27.22	19.07
UGS Components	Upper Flange	48.03	17.41
	Cylinder	94.53	86.07
	Beams	90.68	84.40
	CEA Shroud	64.3	23.80
	CEA Shroud Bolts	21.25	58.61
Internal Structures	Core Shroud Panels	5.72	-
	Core Shroud Guide Lugs	75.12	9.17
	Core Shroud Guide Lug Plate	3.52	-
	Core Shroud/LSS Flexure	79.58	-
	Alignment Keys	16.92	61.33
	Instrument Tube	77.13	-
	Instrument Tube Support	22.68	-

Figure 5-1: Reactor Vessel Internals Assembly



5.3 FUEL AND CONTROL ELEMENT ASSEMBLY STRUCTURAL INTEGRITY EVALUATION

This section provides a summary of the structural integrity calculations performed to evaluate the impact of the replacement steam generators (RSGs) and power uprate conditions for Cycles 15 and 16 and beyond on the fuel and control element assemblies (CEAs). These calculations covered the applicable limiting conditions by examining (1) the structural integrity of the fuel designs under limiting seismic and LOCA analysis conditions, (2) CEA 90% insertion time for normal and upset conditions, and (3) the CEA structural integrity under normal, upset, emergency and faulted conditions.

5.3.1 Structural Performance of the Fuel Designs under Seismic and LOCA Conditions

The overall objective of this analysis is to summarize fuel assembly seismic and LOCA condition structural evaluations that support the power uprate licensing beginning in Cycle 16. The analysis covered both the current power level and the power uprate beginning in Cycle 16. This evaluation is based on current fuel assembly and CEA designs. For example, the Batch J through S fuel assembly structural performance during seismic and LOCA conditions was evaluated. Batches J through S were selected since they represent the debris-resistant fuel designs that are representative of what will be used in Cycle 16 forward.

The criteria that determine acceptable fuel assembly performance for licensing are established in Section 4.2.1.1 of the ANO-2 SAR. Those criteria are shown below in Table 5-2:

Table 5-2

Condition	Component	Acceptance Criteria
Normal Operation and Upset (Including OBE)	All Components Except Fuel Rods	$P_m \leq S_m$ $P_m+P_b \leq F_s S_m$ Deflections limited such that allowable scram time of the CEAs is not exceeded Cumulative fatigue damage fraction shall not exceed 1.0
Normal Operation and Upset (Including OBE)	Fuel Rod Cladding	Deflections limited such that refueling difficulties do not result Tensile Stress ≤ 2/3 Minimum Unirradiated Yield Stress Unrecoverable Circumferential Strain (due to creep and fuel-clad interaction) ≤ 1% Gross Deformation (Collapse) must not occur under the combined effects of external pressure and long–term creep Cumulative Strain Cycling Usage (due to the same effects noted above for circumferential strain) ≤ 0.8
Emergency (DBE)	All Components	$P_m \leq 1.5\ S_m$ $P_m+P_b \leq 1.5\ F_s S_m$ Deflections limited such that CEAs are allowed to scram, but not necessarily within prescribed time Local yielding can occur (but adequate core cooling must be provided)
Faulted (DBE + LOCA)	Fuel Assembly End Fittings Except End Fitting Springs	$P_m \leq 2.4\ S_m$ or $0.7\ S_u$ $P_m+ P_b \leq 2.4\ F_s S_m$ or $0.7\ F_s S_u$
	All Components Except End Fittings	Deformations due to pipe breaks must be limited such that CEAs are allowed to scram, but not necessarily within the prescribed time

Notes (Table 5-2):

P_m = General Primary Membrane Stress Intensity
P_b = Primary Bending Stress Intensity
S_m = Material Allowable Stress Intensity at Highest Metal Temperature
S_u = Minimum Material Tensile Strength at Highest Metal Temperature
F_s = Shape Factor relating Ultimate and Plastic Bending Moments

The application of these criteria to this seismic/LOCA evaluation is as follows:

1. Criteria associated with component stresses of deflections

   For components other than spacer grids, detailed stress calculations are performed to confirm compliance with applicable stress criteria. For these components, adherence to the stated stress criteria is used to confirm compliance with deflection limitations. Compliance with the stress criteria is sufficient to demonstrate the capability to insert fuel rods and cool fuel assemblies. Due to its complex geometry, stress calculations are not done for a spacer grid. Rather, testing is performed to determine the grid's capability (termed "grid strength") to absorb impact loads and still stay within its manufacturing tolerances for rod pitch. In this way, it is concluded that the spacer grid satisfies its stress and deflection criteria provided the predicted impact loads are below its grid strength value.

2. Criteria associated with cumulative fatigue damage fractions

   For components other than fuel rods and spacer grids, standard calculations are performed to verify compliance with cumulative fatigue damage fractions. The evaluation of fuel rod cladding for cumulative fatigue damage fraction is discussed in the following section. For spacer grids, normal operation and upset conditions other than OBE do not impart any cyclic loads on the spacer grid structural members. For the OBE, the cumulative fatigue damage fraction is inherently accounted for in the determination of the grid's strength capability.

3. Criteria for fuel rod cladding strain, collapse and cyclic strain usage

   The criteria wording and/or method descriptions provided in Section 4.2.1.1 of the SAR show that these performance issues are not affected by an OBE event, since it is a short-term event that does not affect RCS pressure or reactor power. Without affecting the RCS pressure or reactor power, there is no effect on the circumferential strain or fatigue or on cladding collapse.

Based on the above information that highlights that RSG at power uprate conditions only impact the fuel due to updated seismic and LOCA conditions, all licensing requirements associated with fuel assembly seismic and LOCA conditions are satisfied if the following three conditions are met:

1. The spacer grid strengths exceed predicted impact loads.

2. For the remaining fuel assembly components, the allowable stresses exceed the predicted stresses.

3. The cumulative fatigue damage fraction following an OBE is less than 1.0 for components other than fuel rods.

5.3.1.1 CEA Structural Integrity

With the introduction of RSGs and power uprate, several reactor components were reanalyzed. The objective was to assess the impact arising from possible changes in system dynamic response, due to differences in the mass distribution of the overall reactor-steam generator system and different flow rates and temperatures.

CEA structural integrity depends on the characteristics of the CEAs as well as those components that comprise adjacent support structures and guide paths. The focus of this analysis is on structural integrity of the CEA rods themselves; more particularly, rod clad stresses under normal, emergency, and faulted conditions. Acceptance criteria for this evaluation are shown below:

Table 5-3
Max CEA Rod Clad Stress Criteria for Full Length CEAs

Operating Conditions	Max. Stress Intensity (psi)	Design Allowable (psi)
Normal Operating & Upset conditions(1)	P_m $P_m + P_b$	S_m $F_s * S_m$
Emergency conditions	P_m $P_m + P_b$	$1.5 * S_m$ $1.5 * F_s * S_m$
Faulted conditions DBE + LOCA	P_m $P_m + P_b$	$1.5 * S_m$ $1.5 * F_s * S_m$

(1) Upset conditions include Operating Basis Earthquake (OBE)

Evaluations of adjacent support structures or other components that comprise the guide paths for the CEAs were addressed in the sections detailing the support structure.

5.3.2 Methodology Used for Evaluation

5.3.2.1 Structural Performance of the Fuel Designs under Seismic and LOCA Conditions

Except for the spacer grids, this analysis utilizes comparisons of overall fuel assembly response to the seismic and LOCA events in order to evaluate whether earlier analytical results remain applicable for all assembly components. The evaluation method used for the earlier work is described in C-E Topical Report CENPD-178-P, Rev. 1-P (and CE NPSD-201-P, the NRC SER on CENPD-178-P). NRC approval was given in the listed SER.

Section 9.6 of C-E Topical Report CENPD-178-P, Rev. 1-P also discusses the basis for determining whether predicted spacer grid loads are acceptable.

C-E continues to have NRC approval for the above methods, as documented in CEN-386-P-A (the latest NRC-approved summary of methods for evaluating the performance of high burnup fuel, NRC letter dated June 22, 1992).

Existing analyses were utilized to show compliance of components other than spacer grids. For the spacer grids, loads from the new analyses were compared to grid load capability from 16x16 generic analyses. As discussed above, the methodologies for these evaluations were unchanged.

Design load inputs for this evaluation are obtained from the reactor vessel internals seismic and reactor vessel internals pipe break analyses. For components other than spacer grids, comparisons are provided for assembly response to the original and the new seismic and LOCA conditions. The spacer grid impact loads for direct comparison to the grid capability are described in Section 5.3.3.1 of this report.

5.3.2.2 CEA Structural Integrity

The regulatory requirements that govern this evaluation are summarized in the design criteria listed in Section 5.3.1.1 above. The method of confirming that calculated clad stresses are bounded by the allowable stress criteria for ANO-2 is demonstrated in two steps, involving comparison with the results of similar calculations performed on the San Onofre Unit 2 CEAs. These comparisons are permissible since the guide tubes and the CEA rods are essentially identical for ANO-2 and San Onofre.

5.3.3 Results of the Evaluation

5.3.3.1 Structural Performance of the Fuel Designs under Seismic and LOCA Conditions

The evaluation of the Batch J through S fuel assembly mechanical performance that is contained in this analysis supports the RSGs and power uprate, since the applicable licensing criteria are satisfied by the fuel assembly designs. Specifically, the three objectives listed in Section 5.3.1 have been achieved as follows:

1) The spacer grid strengths exceed predicted impact loads.

   The maximum spacer grid impact forces for the various events were compared to the load capability of the spacer grids from existing analyses for the following conditions: OBE, DBE, LOCA and DBE+LOCA. Section 4.3.1.1 provides the applicable references for the NRC-approved load comparison method.

   For each condition, the load capability exceeds the predicted maximum impact loads; therefore, spacer grid performance is acceptable.

2) For the remaining fuel assembly components, (i.e., fuel rods, guide tubes, upper and lower end fitting assemblies) the allowable stresses exceed the predicted stresses.

3) Results of detailed calculations verify that the cumulative fatigue damage fraction following an OBE is less than 1.0 for components other than spacer grids and fuel rods.

5.3.3.2 CEA Structural Integrity

Maximum stress intensities occur in the CEA rods when the fuel guide tubes forcibly deflect them during postulated OBE, SSE and LOCA conditions. These deflections were analytically shown to be about twice as great for the San Onofre CEAs as they are for ANO-2 operating with RSGs. Since Normal Operating loads (Condition I) between San Onofre and ANO-2 are comparable, calculated CEA clad stresses under Upset (OBE), Emergency and Faulted conditions are greater for the San Onofre CEAs.

For the purpose of this evaluation, combined SSE and LOCA stresses were compared with allowable stresses associated with emergency conditions. For all operating conditions, calculated San Onofre CEA clad stresses are shown to be less than the corresponding allowable stresses. Allowable stresses are identical for both the ANO-2 and the San Onofre CEAs since they are manufactured using the same materials.

Since the ANO-2 CEA clad stresses were lower than those for the San Onofre CEAs, the results demonstrate that the corresponding ANO-2 CEA clad stresses were conservatively bounded by the allowable stresses. These results confirm that the ANO-2 CEAs are in compliance with the design basis criteria for the plant discussed in Section 5.3.1.

5.4 RCS DEADWEIGHT AND THERMAL EXPANSION LOADS ANALYSES

This section describes the mechanical and thermal analyses performed to determine the response of the reactor cooling system (RCS) when subjected to the effects of deadweight (DW) and normal operating (NOp) loads. The effects of the RSGs and power uprate on the existing design basis DW and thermal expansion (TH) structural loads for the RCS were assessed, and where required, new design loads consistent with the RSGs and power uprate were produced.

This section addresses RCS structural components needed for maintaining the reactor coolant pressure boundary (RCPB) with the exception of tributary line piping up to the second valve. Tributary line piping were evaluated separately and the design is adequate for power uprate conditions as discussed in section 5.8. The surge line is not considered to be tributary piping. Where required, the surge line and remaining RCS components are discussed separately in this section. This is because the surge line is structurally decoupled from the rest of the RCS and has generally been analyzed separately, sometimes using different methodologies and/or computer codes. Additional thermal loadings (e.g., thermal stratification) are also considered in surge line analysis. The surge line can be structurally decoupled because its relatively lower mass and stiffness prevent it from affecting the rest of the RCS, i.e., the main coolant loop (MCL).

The MCL is defined as:

1. The reactor vessel (RV),
2. Reactor coolant pumps (RCPs),
3. Steam generators (RSGs),
4. Hot and cold leg piping,
5. RV, RCP and SG supports.

Therefore, the scope of the equipment addressed in this section is:

1. the MCL,
2. the pressurizer (PZR),
3. the surge line.

Although the RCP motors are outside of the RCPB, their mass and stiffness effects are included in the MCL analysis, as are the effects of the RV internals. The PZR is unaffected by RSG at power uprate conditions. However, the PZR surge and spray nozzles are qualified for RSG and power uprate piping loads. RV internals are addressed in Section 5.2 of this report.

NOp loads are defined as the loads resulting from the signed addition of the DW and TH loads for the 100% power condition. The original DW and NOp analyses performed for the MCL and surge line used system operating parameters consistent with the original steam generators (OSGs) and the associated core power level. The gross effects caused by RSG and uprate are due to the following:

1) an increase in the SG weight, and
2) a change in the hot and cold loop operating temperatures (i.e., a change in the values of T_{hot} and T_{cold}).

The new sets of DW and NOp loads for the RCS and surge line and all thermal movements at interface locations are a combination of unmodified and modified original design values, where original values are defined as those generated for the RCS with OSG configuration.

5.4.1 Regulatory Basis

The MCL major components and their supports and the piping were evaluated for DW and TH effects to determine loads due to NOp loading conditions. The RCS (and associated auxiliary, control and protection systems) was required to be designed with sufficient margin to assure that the design conditions of the RCPB were not exceeded during any condition of normal operation including anticipated operational occurrences (AOOs). The process of assuring the integrity of the RCPB determines that NOp structural loads are within allowable limits and remain so for uprated conditions.

Normal operating limits were established for steady state and transient plant operations. The normal operating limits were selected so that adequate margin exists between them and the design (allowable) limits. The structural analyses described herein only considered the gross effects of the specified plant transients, e.g., the magnitude and direction of friction forces at the SG sliding base vertical pads during plant heat-up and cool-down.

MCL components have also been analyzed for earthquake and pipe break effects. The results of each of the individual analyses were used as inputs to the structural integrity analysis of the RCS, in which stress analyses were performed in accordance with the rules of Section III of the ASME Code.

The surge line was also evaluated for deadweight and thermal expansion effects due to the RSG at uprated condition, the results of which were also used as input to ASME Code, Section III, stress analyses. In addition to weight and thermal expansion effects, transients were considered in the surge line analysis.

Thermal stratification is the controlling transient event for the surge line. The stratification transients analyzed for the RCS with OSG condition were developed from actual plant operating data, and are a compilation of data from a number of plants. These transients occur during plant heat-up and cool-down. Since the temperatures during heat-up and cool-down did not change for the RSG at uprated conditions, there was no impact on the stratification transients and the previous design basis transient results remained valid.

Thermal anchor movement (TAM) data was part of the input to the thermal stratification analysis. Since there has been a negligible change in the MCL piping NOp hot and cold leg temperature differentials (measured from ambient conditions), the RSG at uprated conditions has no impact on the original design basis TAMs.

5.4.2 Methodologies Used For This Evaluation

The original design basis analysis for DW and NOp loads employed a mathematical model to represent the MCL components. The surge line was analyzed separately from the rest of the RCS.

Detailed DW and NOp analyses were not needed to reevaluate the MCL for RSG at uprated conditions. The surge line has been analyzed for the original design basis DW and NOp conditions. Since plant heat-up and cool-down temperatures for the surge line do not change for RSG at uprated conditions, and TAMs at the surge nozzle/surge line interface remain unchanged, the original design basis surge line results remain valid.

Not all original design basis results for the MCL portion of the RCS remained unchanged for the RSG at uprated conditions. Therefore, the following methodology was implemented to determine new sets of design loads and thermal movements.

The generation of new design DW and NOp loads was a three step process. Step 1 was to determine and access margins between the calculated and design basis loads for the RCS with OSG configuration were determined and assessed. The set of loads resulting from step 1 was for the most part equivalent to the original design basis loads; however, there were some exceptions. Step 2 was to determine the percentage increases in SG weight and RCS operating temperatures resulting from RSG at power uprate conditions. Step 3 was to selectively apply the step 2 percentage increases to the step 1 loads to produce the set of new design loads. The original TAMs did not change due to the RSG at power uprate conditions.

In the first step of the process, the existing margins between the as-calculated and design loads for the RCS with OSG conditions were determined and examined. With the exception of the SG primary nozzles and supports (i.e., support skirt and sliding base), this examination proved that (1) the as-calculated values were the values used to establish the original design loads, and (2) a margin had not been applied to the as-calculated values to create the original design loads. Therefore, with the exception of the SG nozzles and supports, RSG and uprate effects were applied to the original design loads to create the new design loads.

The determination of factors for applying RSG and uprate effects is explained in the following sections.

5.4.3 RSG Increased Weight Effects

The SG wetted weight comparison is shown below in Table 5-4 at both the 0% and 100% power conditions. Percentage increases due to RSG are shown.

Table 5-4

Condition	Total Weight of Component (lbs.)		% Increase in Weight
	RSG	OSG	
0% Power	1,510,611	1,335,622	13.1
100% Power	1,428,777	1,251,487	14.2

The procedure for incorporating the effects of increased SG weight was to calculate the ratio of RSG to OSG weight at 0% and 100% power conditions, determine a bounding weight increase value, and use this value for increasing DW loads at selected RCS locations.

The most significant DW effects occur under NOp conditions. A differential vertical thermal growth develops between the RV and SGs during plant heat-up, because the combined growth of the RV columns and integral support feet is significantly greater than the vertical growth occurring in the vicinity of the SG support pedestal. This differential thermal growth causes the RV and support columns to apply uplifting loads to the SGs through the interconnecting main coolant loop piping, resulting in the outward tilting of the SGs and liftoff of the sliding base front pad. The front pad liftoff will cause a redistribution of the SG weight loads. The three vertical pads remaining in contact will receive a larger proportion of the SG weight load than in the deadweight configuration. Additionally, more load will be distributed through the hot legs to the RV during NOp, because the hot leg will become a load path in the absence of sliding base front pad contact. Any SG weight increases will be redistributed in the same proportions.

A conservative approach was taken in this analysis. The SG weight increase was accounted for at the SG supports (i.e., at the vertical pads), and at the nozzles and piping connecting the RV and SGs by scaling all possible components of load up by a factor of 1.122. This was done for both the deadweight and the NOp (deadweight plus thermal) conditions.

Load increases in the hot and cold legs are in reality less that 12.2%, because (1) SG weight is only one contributor to the load at any piping location, and (2) SG weight loads are distributed through both the hot and cold legs. Also, the hot legs are much stiffer than the cold legs, and will therefore act as load paths for a higher proportion of loads coming from the SG weight increase. Therefore, the effect of increased SG weight on the main coolant loop piping was determined to be limited to the hot legs, and load increases on the cold loops were limited to the ends of the SG outlet nozzles. To be consistent with the general approach taken in the analysis, the entire DW increase was applied to the SG outlet nozzle locations, even though the actual percentage increase of any of the loads at these locations would have been significantly less than the SG weight percentage increase.

The design deadweight load for the RSGs is considerably greater than the NOp load. Therefore, downstream analyses performed using the existing RSG support design loads remain conservative and valid.

5.4.4 Power Uprate Effects

As the hot leg to cold leg temperature differential increases, the relative difference in hot and cold loop linear thermal expansion increases. This in turn increases the imbalance in piping loads transmitted to the SG centerline and results in increased piping and nozzle loads throughout the system. This effect is linearly proportional to the temperature differential.

The comparison of hot and cold loop temperature differentials at the nominal 100% power condition is shown below. The percentage increase due to power uprate is shown in the last column. A conservative and bounding T_{hot} is used for power uprate conditions (T_{hot} calculated by using the technical specification lower limit RCS flow rate and upper limit T_{cold}). T_{hot} is expected to be approximately 609 °F.

Table 5-5

RCS Configuration	Operating Temperatures (°F)		ΔT (°F)	% Increase due to power uprate
	T_{cold}	T_{hot}		
OSG	553	611	58	
RSG at uprated conditions	554.7	617.7	63	8.62

Based on the above results, a conservative percentage increase of 10% was selected as a guideline for increasing all thermal loads.

Comparison of original as-calculated thermal movements to design thermal movements showed that the original design values were equal to the as-calculated values. Consequently, no margin existed in the original design thermal movement values.

The thermal movement of a given point on the MCL is controlled by the temperature differential from ambient to operating conditions.

Resulting thermal factors are shown below:

Table 5-6

Location	T_{new}	T_{old}	Thermal Factor (%)
hot loop	617.7	611	1.24
cold loop	554.7	553	0.35

Based on the above results, which showed changes on the order of 1% for either loop, it was concluded that the original design basis TAMs remained valid for the RSG with power uprate conditions.

The design and operating temperatures of the pressurizer, and therefore its thermal movements, were not affected by RSG and power uprate. The TAMs on the hot leg piping did not change. Furthermore, and as discussed above, the hot leg surge line nozzle DW and NOp loads were only marginally affected by RSG and power uprate. Therefore, the surge line DW and NOp loads from the original design basis remained valid.

The following can be stated regarding conditions affecting the PZR. Minor changes in the MCL piping T_{hot} and T_{cold} values due to RSG at uprated conditions did not change the TAMs either at the hot leg/surge line interface, or along the surge line up to the surge line/PZR nozzle interface. Also, building growths at the base of the PZR were unchanged, and design temperature conditions (653 °F) affecting thermal expansion of the PZR enveloped the current conditions. Since the controlling transient conditions (including surge line thermal stratification effects) were unchanged from the original design basis, the original thermal analyses remained bounding. Therefore, the original design basis PZR loads and motions remained valid for RSG with power uprate.

5.4.4.1 Results for This Evaluation

The results of this evaluation consisted of (1) the MCL major component and component support design loads, (2) the MCL tributary nozzle design loads, (3) the surge line design loads, and (4) the MCL piping tributary nozzle TAMs.

Final design loads are found in the individual component and component support specifications. These loads were subsequently used as input to stress analyses and structural integrity assessments of the MCL and Surge line for the RSG and power uprate conditions.

The loads and thermal movement results documented herein pertain to external interface locations. These locations are at the component support interfaces and the tributary nozzle terminal ends.

5.5 REACTOR COOLANT SYSTEM SEISMIC ANALYSIS

The purpose of this analysis was to determine the dynamic response of the reactor coolant system (RCS) with RSGs to seismic excitations. This included producing an ANSYS reactor coolant system model that duplicated the dynamic characteristics of the original STRUDL model. The ANSYS RCS model was then modified to represent the RCS with RSGs. The seismic effects of the RSG on surge line response were also evaluated.

These calculations were performed for system operating parameters consistent with the uprate core power level, so that the results are applicable to power uprate.

5.5.1 Regulatory Basis

The regulatory bases for the seismic evaluation of the RCS are contained in General Design Criteria 1, 2, and 14. Statements of conformance to these GDCs are contained in FSAR Amendment No. 14, Section 3.1, for the original plant configuration, which continues to be the licensing basis for the RSG configuration with power uprate. Per Section 3.2.1 and Table 3.2-2 of the SAR, the design of these components conforms to Regulatory Guide 1.29 and sections of the ASME code pertaining to Seismic Category I components.

5.5.2 Methodology Used for This Evaluation

The original seismic analysis was performed using the STRUDL code to define the structural properties of the RCS. The current analysis used the ANSYS code to define the structural properties of the RCS. The ANSYS model was developed by converting the STRUDL commands to equivalent ANSYS commands. The equivalence of the two models was confirmed by comparison of their frequencies and mode shapes. The RCS model was updated to include the RSGs and a detailed RV internals representation. The seismic analysis of the RCS with the RSG was performed using the same methodology as the original analysis. The original OBE time history motions were used as input to the RCS supports. Motions were applied separately in the three orthogonal directions; X parallel to the hot leg, Y vertical, and Z perpendicular to the hot leg. Time history motions were applied at each RCS support location. The motions applied varied depending on the location in the building model that the RCS support was attached to. The response of the RCS for each direction was calculated using the method of modal superposition with a constant damping ratio of 1% for OBE. For DBE, twice the OBE time history input was used with a constant damping ratio of 2%. The response loads for the three directions were combined by finding for each result component the maximum of the result component due to the vertical excitation plus or minus the result component due to each horizontal excitation.

ANSYS calculated the maximum results for $Y \pm X$ and for $Y \pm Z$ on a time history basis. The two sets of results were transferred to a spreadsheet. The spreadsheet determined the absolute maximum result for each component.

The evaluation of the seismic effects of the RSG configuration on the surge line was performed by comparison to the seismic effects of the OSG configuration on the surge line. The original seismic analysis of the surge line was performed by response spectrum analysis for one-half percent (½ %) constant damping. The inputs to the surge line analysis were response spectra at the hot leg nozzle, which were generated from time history responses of the RCS with the OSG configuration to seismic excitation.

The seismic evaluation of the surge line for the RCS with RSG was performed by design review and comparison to the original seismic analysis. No new seismic analysis of the surge line was necessary.

5.5.3 Results for This Evaluation

This evaluation produced two types of results. The first type is results that were used directly in other analyses, to validate the RCS seismic loads. These include interface loads, main loop nozzle loads and component loads. The second type is time history motions and response spectra that were used as input to other analyses. These include time history motions at branch line nozzles, at RVI to RV interfaces and on the RSG, and branch line nozzle response spectra.

Output from the original seismic analysis of the RCS included seismic response spectra at the surge line hot leg nozzle. Since the evaluation determined that the response spectra at the hot leg nozzle enveloped the seismic response spectra at the other surge line supports, these response spectra were used as the input to the original spectral analysis of the surge line.

Seismic response spectra at the surge line hot leg nozzle were also generated as an output of the current RCS seismic analysis described above. A comparison of the original OBE ½ % damping response spectra at the surge line hot leg nozzle with the OBE ½ % damping response spectra for the RCS with RSG indicated that the original response spectra envelop the current response spectra over the applicable frequency range. Based on that comparison, both the OBE and DBE response loads at the surge line piping assemblies, nozzles and supports from the original analysis were demonstrated to remain bounding for the RSG configuration.

5.6 LEAK BEFORE BREAK EVALUATION

The evaluations documented in Combustion Engineering Topical Report CEN-367-A, "Leak Before Break Evaluation of Primary Coolant Piping in Combustion Engineering Designed Nuclear Steam Supply Systems" are applicable considering revised loadings due to steam generator replacement and power uprate. This allows the dynamic effects of postulated main loop pipe breaks to be excluded from the design basis of main loop components, internals, fuel, supports, attachments and appurtenances, and attached piping systems. Breaks in attached branch line piping are required to be considered.

Elimination of postulated primary loop pipe ruptures from the design basis is permitted by the revised General Design Criterion 4 (GDC 4) of Appendix A to 10 CFR Part 50. The revised rule is based on the development of advanced fracture mechanics technology using the LBB concept. The revised rule was published (52 FR 41288) on October 27, 1987. The criteria for evaluation of compliance with GDC 4 are defined in Chapter 5 of NUREG – 1061, Volume 3. The criteria address loading criteria, degradation of piping resulting from stress corrosion cracking, fatigue, or water hammer, materials properties, postulated through wall flaws and leak detection capability, margin in terms of applied loads, and margin between the leakage-size flaw and the critical size flaw.

The use of Topical Report CEN-367-A requires leakage detection systems to meet the guidelines of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems". ANO-2 leakage detection systems that meet RG 1.45 requirements are in place.

5.6.1 Methodology Used for this Evaluation

Each of the criteria for evaluation of compliance with the revised GDC 4 were reviewed for impacts due to steam generator replacement and power uprate.

5.6.2 Results of this Evaluation

Each of the six LBB compliance criteria in topical report CEN-367-A was evaluated. Compliance with all criteria was found to be unaffected by replacement of steam generators and power uprate. Criteria (1), (4), (5), and (6) had the largest potential for impact due to the revision of system normal operating and seismic loads caused by steam generator replacement and power uprate. All piping load revisions were found to be within the envelope loads employed in the CEN-367-A evaluation. Therefore, the evaluations of the topical report for Criteria (1), (5), and (6) are applicable for ANO-2 following power uprate and RSG. Criterion (2) is not impacted by generator replacement or power uprate since there is no impact on stress corrosion, water hammer and fatigue of the main loop piping.

Criterion (3) requires that the welding process employed to connect the replacement generators to the main loop piping are equivalent to the original shielded metal arc welding (SMAW) process. Entergy confirms that the welds between the RSG and the RCS piping meet or exceed the J-R properties used in the CEN 367-A evaluation. This satisfies Criteria 3 and hence, the evaluations of the topical report are applicable for ANO-2 following power uprate and RSG.

Criterion (4) is addressed as follows. The replacement steam generators and power uprate do not impact the leakage detection systems employed at ANO-2. Since the evaluation conservatively employed only pressure loading in the determination of initial flaw size, the evaluation remains enveloping. The flow rate correlation for leakage through a flaw is not impacted by power uprate or steam generator replacement. Therefore, the Criterion 4 evaluations remain applicable to ANO-2 after steam generator replacement and power uprate.

Postulated pipe breaks in the remaining high energy piping systems are considered. Specifically pipe breaks in the surge, shutdown cooling, safety injection, steam and feed water lines are considered in the design of main loop components, internals, fuel, supports, attachments and appurtenances, and attached piping.

5.7 REACTOR COOLANT SYSTEM PIPE BREAK ANALYSIS

The purpose of this analysis was to determine the dynamic response of the reactor coolant system (RCS) to postulated pipe breaks. Evaluations were performed for system operating parameters consistent with the uprate core power level, so that the results are applicable to power uprate.

The dynamic response of the RCS to pipe breaks was determined, and response loads, motions and spectra at pre-determined interface locations were provided as input to specifications and to downstream structural analyses. Response loads and motions from the RCS analysis were provided as input to the RSG design specifications, NSSS component specifications and balance of plant piping evaluations.

Hot leg response motions and spectra due to pipe breaks other than surge line breaks were provided as input to the surge line analysis. The response of the surge line to these excitations was also determined, and response loads at pre-determined surge line pipe, nozzle and support locations were provided as input to the NSSS component piping specification.

5.7.1 Regulatory Basis

Per General Design Criterion (GDC) 4, ANO-2 safety systems are required to be designed to withstand the consequences of postulated pipe breaks so as not to compromise safe shutdown. The RCS, its components, supports, appurtenances, reactor vessel internals and connected tributary piping, including the surge line, are designed to Faulted Loading Conditions, which include loads resulting from postulated pipe breaks.

Per the original version of GDC 4, the mechanical design basis for the original ANO-2 RCS configuration included postulated breaks in all high energy piping above one inch (1") in diameter. Through application of Leak-Before-Break (LBB) technology, which was subsequently allowed by revisions to GDC 4, the need to consider the mechanical (dynamic) effects of main coolant loop breaks (MCLBs) was eliminated. Following the application of LBB, the remaining pipe breaks in the mechanical design basis of the RCS are all primary and secondary side branch line pipe breaks (BLPBs) interfacing with the RCS. Of these, the limiting breaks with respect to RCS structural considerations are breaks in the largest tributary pipes:

- main steam line (MSL)
- feedwater line (FW)
- surge line (SL)
- safety injection line (SI)
- shutdown cooling line (SDC)

A review of pipe stresses and fatigue usage factors (where appropriate) for the as-built configurations of these piping systems eliminated the intermediate breaks in all but the surge line. The terminal end breaks in all five piping systems listed above plus intermediate breaks in the surge line remained as controlling pipe breaks. Further review of loadings on the RCS determined that the terminal end BLPBs at the RCS interface consistently and conclusively enveloped the BLPBs at the other terminal ends with respect to their effect on RCS response. Therefore, the final set of BLPBs postulated and analyzed for RCS response for the RSG configuration with power uprate consisted of fifteen (15) BLPBs.

The response of the surge line to non-surge line pipe breaks is also based on GDC 4 criteria. surge line response to RCS hot leg motions from selected breaks was determined by analysis. Response of the surge line to pressurizer motions is based on smaller pipe breaks at the top of the pressurizer. These breaks are not affected by LBB, RSG or power uprate, and their effects on the surge line are enveloped by the effects of the five major branch line pipe breaks.

5.7.2 Methodology Used for This Evaluation

There are significant differences between the pipe break analysis methodology for the RCS with OSG vs. the RCS with RSG. The original dynamic pipe break analysis of the reactor vessel and its supports was performed using non-linear time history methods only for RV response. The mathematical model defined using STRUDL representation of the RV, RV internals, RV columns and hot legs, and the non-linear time history analyses were performed using the CEDAGS computer code. The model was analyzed for RCS pipe breaks most significant to RV response, i.e., terminal end guillotines at the RV inlet and outlet nozzles and at the SG primary inlet nozzle. Mass-stiffness (α-β) damping of not more than 3% at significant modes of vibration was used in the time history analysis of the RV. The remainder of the RCS with OSG was analyzed statically for the set of MCL pipe breaks in the original mechanical design basis.

For the RCS with RSG, non-linear response time history analyses were performed to calculate the RCS response to the selected BLPBs. These BLPBs replaced the MCL breaks in the mechanical design basis, following elimination of the MCL breaks, using leak-before-break arguments. For the pipe break analysis of the RCS with RSG, two three-dimensional ANSYS models of the entire RCS were developed from the RCS seismic model, one for secondary side breaks and one for primary side breaks. The response of the entire RCS to pipe breaks was calculated using non-linear response time history analysis. The ANSYS computer code was used to perform the dynamic transient time history BLPB analyses, using the modal superposition method and constant 3% modal damping.

For both pipe break models, gapped and preloaded RCS supports were de-linearized. Directional spring supports gaps and preloads at full power were included for all pipe break analyses. The RVI snubbers between the core support barrel (CSB) and the inside of RV lower head were modeled as gapped springs for all pipe break analyses.

For the secondary side pipe break model, the representation of the RVI remained essentially the same as that for the seismic model, because secondary side breaks do not cause RV blowdown, just vibratory input.

A more detailed model of the RVI was included in the primary side pipe break model, because these pipe breaks cause RV blowdown loads. This RVI model included hydro-mass and coupling terms, as well as additional nodes for RV blowdown input loadings.

Input loadings applied to the RCS included thrust at the break locations, jet impingement loadings at and away from the break locations, RV blowdown loadings for the primary side BLPBs, and asymmetric pressurization loads on the RSG and RCPs for all pipe breaks except the MSLB (which does not cause SG subcompartment pressurization). Jet targets and jet impingement loadings were based on cone jets or fan jets, depending on the break type and break scenario. After elimination of main coolant loop breaks by application of LBB, none of the limiting BLPBs for ANO-2 cause asymmetric pressurization to occur between the RV cavity and the RV shell.

Linear response time history analysis was used to calculate surge line response to non-surge line BLPBs. Time history motions at the RCS hot leg interface due to BLPBs were applied to the surge line, and the surge line response was determined using the ANSYS computer code. Mass-stiffness (α-β) damping of not more than 3% at significant modes of vibration was used in the time history analysis of the surge line.

No analysis was previously performed to determine pipe break vibratory motion effects on surge line response for the OSG configuration.

5.7.3 Results of This Evaluation

RCS component and support loads were maximized over the response time history for each BLPB analyzed.

Time history response accelerations and displacements were generated at pre-determined locations listed below. The response displacements at the following RCS locations were maximized over time for each pipe break case. These results were used in the evaluations of tributary piping, RSG design, and evaluations of the RVI and RV head area components.

RSG tributary nozzle interfaces
RCS tributary nozzle interfaces including surge line nozzle
RV upper head
RV closure flange
RV CSB snubber

Maximized RSG shell moments at various RSG shell elevations were calculated for each break case for use in the detailed RSG design.

Response spectra at the hot leg-surge line interface and at the RV upper head elevation were generated for use in the surge line and RV head area analyses, respectively.

Response of the surge line to RCS vibratory motion due to the five non-surge line BLBPs was calculated. Surge line piping and support response loads were calculated and were maximized over time. In addition, the time maximized results were maximized over all five pipe break cases. These results were used in the surge line evaluation for the faulted condition.

5.8 RCS TRIBUTARY LINE RECONCILIATION ANALYSIS

The purpose of the RCS tributary line reconciliation analysis was to evaluate the changes resulting from the RSGs at power uprate conditions and to reconcile the resultant loads against applicable code allowables. These analyses included stress analyses of the piping and applicable pipe supports. Inputs to the piping analyses included Loss of Coolant Accident (LOCA) displacements and seismic response spectra. LOCA displacements due to branch line pipe breaks (BLPBs) were calculated for the postulated breaks of the largest tributary and secondary lines of the RCS (safety injection, shutdown cooling, and pressurizer spray, main steam and main feedwater lines). The seismic response spectra were calculated from the coupled reactor building/RCS analyses with the variable damping of ASME Code Case N-411.

5.8.1 <u>Regulatory Basis</u>

5.8.1.1 Piping

The piping was analyzed in accordance with the rules of ASME Code, Section III, Subsection NB/NC/ND and with additional criteria set forth in ASME Code Case N-411. High energy postulated pipe break location analysis utilized the criteria of NRC Standard Review Plan (SRP) 3.6.2 Revision 1, and NRC Generic Letter 87-11.

5.8.1.2 Supports

The repair and replacement activities for the tributary line pipe supports were analyzed in accordance with the applicable AISC Manual of Steel Construction.

5.8.2 <u>Methodology Used for This Evaluation</u>

5.8.2.1 Pipe and Pipe Support Data Input

Data input to the pipe and pipe support analyses included the following:

a. Pipe and pipe support information was based on the latest revisions (as-built) of the piping isometric drawings and pipe support drawings.

b. Seismic accelerations were based on the response spectra with Code Case N-411 damping for the reactor coolant loop nozzles, containment internals, and where applicable, containment shell wall.

c. Seismic anchor movement (SAM) data was taken from the coupled containment/RCS seismic analysis.

d. LOCA displacements were taken from the Branch Line Pipe Break (BLPB) analysis.

e. Although basically unchanged from the OSG design, the thermal loads for the RSG were analyzed using the temperatures and pressures required for power uprate.

f. Design transients for fatigue evaluations were taken from the design transient evaluations.

5.8.2.2 Pipe Evaluation

The methodology followed in the analysis and evaluation of the tributary piping system was in accordance with ANO design engineering specifications that provide the general requirements for piping analysis at ANO - Units 1 & 2. The analyses were divided between ANO design engineering and Westinghouse, the RSG supplier.

The ME101 (which includes ME101C1 to perform ASME NB-3600 Class 1 analysis) pipe stress analysis program was used to create mathematical models of the designated lines. ME101 is a family of computer programs that performs linear elastic analysis of piping systems using standard beam theory techniques. ME101 has advanced static and dynamic analysis capabilities including detailed uniform and multilevel response spectrum analysis, time history calculations, fatigue calculations, and multiple load cases and combinations. In addition to Class 1 stress analysis, ME101 also performs ASME Section III Class 2 & 3 and ANSI B31.1 code stress checks and high energy line break location analysis.

For analyses provided by Westinghouse, the PIPESTRESS code was used to create models of the designated pipe lines. PIPESTRESS has advanced static and dynamic analysis capabilities including detailed uniform and multilevel response spectrum analyses, time history calculations, fatigue calculations, and multiple load cases and combinations.

The primary assumption made in the evaluation and reconciliation analyses was the original design basis qualification criteria for load combinations, as required per ASME code 1971 Edition, plus the effects due to LOCA loads, with the following exceptions and clarifications:

- Seismic qualification is based on response spectra with Code Case N-411 damping.

- LOCA loads are included and combined with the seismic SSE loads by the Square Root Sum of the Squares (SRSS) method.

- Fatigue evaluations were reconciled back to the code of record, ASME code 1971 Edition.

The seismic inertia load analysis utilized the modal response spectra method, taking into consideration closely spaced modes per NRC R.G. 1.92 and the total response obtained from the three spatial directions by the SRSS combination method.

The total seismic load for each of the tributary lines analyzed was developed, using the SRSS method, utilizing the seismic inertia and seismic anchor movement loads.

For all tributary lines except the cold leg drain line, a static load calculation was made for the pipe break effect loads (LOCA). The calculation enveloped the nozzle displacements resulting from postulated breaks for each of the three directional displacements. The results were then combined using the square root sum of the squares (SRSS) method. A conservative dynamic load factor (DLF) of ≥2.0 was applied in the analysis to account for the dynamic effects. The following tributary lines were evaluated using static LOCA displacements:

a. Shutdown cooling
b. Safety injection
c. Main steam
d. Main feedwater
e. Charging
f. Letdown
g. Hot leg drain
h. SG blowdown
i. PZR spray

For the cold leg drain lines, a time-history analysis was performed for LOCA. The global loads at the reactor coolant loop nozzle and the tie-back support were obtained. This analysis of the drain lines bounded the most limiting drain line configuration and input boundary conditions based on reviewing the displacements and accelerations of the three drain lines.

In order to provide adequate flexibility for the increased LOCA movements or to minimize the support modifications for increased loads, it was necessary to delete a few snubbers and supports on some of the lines. Deletion of these snubbers and supports was acceptable for seismic loading based on the analyses using the response spectrum with Code Case N-411 damping.

5.8.2.3 Pipe Support Evaluation

The methodology followed in the analysis and evaluation of the tributary system pipe supports was in accordance with ANO design engineering specifications that provides the general requirements for the design of supports at ANO - Units 1 & 2. Pipe support components and structures were qualified for the increased loads calculated in the piping analysis due to the effects of the RSG.

5.8.3 Results of This Evaluation

The RCS tributary lines have been evaluated and reconciled for RSG and power uprate by ANO design engineering and Westinghouse. The analysis results demonstrate that the tributary piping remains in compliance with the applicable ASME Code, Section III, Subsections NB/NC/ND allowables. Therefore, these lines satisfy the acceptance criteria required per applicable regulatory codes.

All supports evaluated due to the effects of the RSGs met the structural acceptance criteria and therefore required no modifications. A total of four supports were deleted of which three were snubbers.

For safety injection lines, high energy line break locations were selected at the terminal ends and at the intermediate locations based on the criteria in SRP 3.6.2 Revision 1 and NRC Generic Letter 87-11. Use of the ASME Code, 1980 Edition resulted into substantial reduction of the originally postulated intermediate break locations. A few new locations were however needed to be postulated due to the reanalysis. These new locations are all in the vicinity of the originally postulated locations. The whip restraints installed on these lines for OSG are however left in place. Therefore, the protection from dynamic effects has not changed.

For shutdown cooling and pressurizer spray lines, breaks were originally postulated to occur at the terminal ends and at every intermediate pipe fitting, welded attachment and valve. Therefore, the reanalysis had no effect on the postulated break locations on these lines, and the protection from dynamic effects has not also changed.

For the purpose of application of LBB to the MCL analysis, the postulated breaks beyond the first isolation valve do not cause LOCA and are therefore not included. Breaks in these lines between the loop nozzle and the first isolation valve are also not included since these portions of the lines have stresses and fatigue usage factors below the MEB 3-1 break postulation criteria for intermediate breaks.

Fatigue Evaluation

In accordance with Subsection NB-3600 of 1980 Edition of the ASME Section III Code, fatigue evaluations were performed for each of line. Revised design transients were used in the fatigue evaluations. These design transients included the effects of the replacement steam generators as well as revisions to the transients and the number of cycles to account for actual plant operation and for potential plant life extension to 60 years.

5.9 REPORT ON OVERPRESSURE PROTECTION

SAR Section 5.2.2.3 presents the report on overpressure protection as required by the ASME Code, Section III, 1968 Edition. The specific overpressure protection requirements of the ASME code were evaluated for power uprate. All general requirements and component requirements for pressurizer safety valves and main steam safety valves were found to be in compliance with the code.

Computer code analysis of bounding reactor and steam plant transients causing pressure excursions have been conducted. These transients were evaluated to ensure both peak primary and secondary pressure did not exceed 110% of design pressure.

END OF SECTION

6 NSSS DESIGN TRANSIENTS

This section describes the nuclear steam supply system (NSSS) fluid transients that are used for the structural design of the replacement steam generator (RSG), reactor vessel, RCS main coolant piping and nozzles, pressurizer, and reactor coolant pumps (RCPs). This set of NSSS transients is comprised of the NSSS response to plant heatup, plant cooldown, plant loading, reactor trips and others.

6.1 INTRODUCTION

The design transients are classified into four categories based on how often the transient is expected to occur: normal condition, upset condition, emergency condition and faulted condition. The basis for these transients and the number of occurrences assumed was intended to provide a system/component design which will not be limited by expected cyclic operation over the life of the plant.

The specified design transients represent conservative estimates for design purposes and do not purport to be accurate representations of actual transients, nor do they necessarily reflect actual operating conditions. The entire set of design transients is input into the structural integrity analysis and the RSG component analysis.

The primary reason for redefining and evaluating these transients is to account for any impacts of the RSG and the power uprate. For example, the RSG can affect the RCS thermal hydraulics such that the initial and final steady state temperatures and pressures for any given design transient may be different from those in the original design analysis of the NSSS. The secondary reason for redefining and evaluating these design transients is to account for anticipated differences between the actual operating conditions of the plant and the projected or assumed conditions of the plant used in the design transient analyses.

6.2 REGULATORY BASIS

The methodology for defining the NSSS design transients ensures that they bound both normal and abnormal plant operations. The number of occurrences of any given transient selected for design purposes exceeds the expected number over the life of the plant. The intent is to ensure that no reactor coolant system component is stressed above the allowable limit as described in the ASME Code - Section III. The transients are classified into four categories (normal, upset, emergency and faulted) which are consistent with the ASME Code classification. Again, the design transients are conservative estimates for design purposes and are not necessarily a representation of actual transients or reflective of actual operating conditions. However, in developing the design transients, actual operating conditions, expected operating transients, non-LOCA transients and LOCA transients are used as a basis.

The NSSS transients are defined in accordance with the ASME Code requirements for Class 1 components in that each condition to which a component may be subjected shall be described in the design specification. The set of NSSS transients defined herein form the basis of the design specifications.

10CFR50.55a, Codes & Standards, requires that systems and components meet the requirements of the ASME B&PV Code. Since the design transients are input into the ASME B&PV Code Stress Analysis, this section describes some of the steps needed to meet 10CFR50.55a.

6.3 METHODOLOGY USED FOR EVALUATION

This section defines a set of thermal-hydraulic conditions and thermal-hydraulic transients which can affect the stress level of components. The range of pressures and temperatures to which a component is subjected during a particular transient and the number of cycles for this same transient influence the stress levels for a component. The new NSSS transients were based on conservative estimates of the NSSS response to normal plant operations and upset events. These conservative estimates were provided as input to the structural integrity analyses of Section 5 of this report.

The bulk of the thermal design transients fall within the two categories of normal conditions and upset conditions. Level A (normal conditions) and Level B (upset conditions) transients were based upon expected normal plant operating conditions. Consistent with normal plant operating conditions, non-safety grade plant control systems have been assumed operational. This is consistent with the ASME B&PV code description of Level A and B categories.

Level C (emergency) and Level D (faulted) transients are rare occurrences. These events are not considered in structural fatigue analysis but are considered in primary stress calculations. The thermal-hydraulic plant responses for these severe events are documented to enable the structural designer to assess the effect of Level C and D transients on component stress. The results of safety analyses were used to help define the NSSS thermal hydraulic responses for Level C and D events.

The frequency of occurrence for design basis events is intended for design purposes only. The frequency of occurrence used in the design transient analyses is expected to exceed the actual number of occurrences over the life of the plant. Actual plant data was used to estimate the design frequency of occurrences for some events. These events included the loss of letdown events, the loss of feedwater flow event, and the feedwater nozzle transient events during lower mode operation.

Conservatism in the thermal-hydraulic responses was provided in several ways. The number of occurrences selected exceeds the expected number. Conservative methods of predicting the range of pressure and temperature for the transients were used. A composite transient was defined with the most severe portion of the transient derived from a group of transients.

The normal maneuvering transients were simulated using a best estimate simulation code based upon industry standard methods with ANO-2 plant specific data. Projected nominal plant operating conditions for Cycle 15 and Cycle 16 were assumed for the normal maneuvering transient evaluations.

6.4 RESULTS OF EVALUATION

6.4.1 Expected Normal Operating and Design Data

For the following equipment, the expected normal operating point data and design data were used to determine the initial conditions for the transients:

1. Reactor vessel
2. Reactor coolant pumps
3. Pressurizer
4. Reactor coolant piping and fittings
5. Replacement steam generator primary side, and
6. Replacement steam generator secondary side.

Variation in RCS temperatures and steam generator pressure as functions of power were also used in some transients.

6.4.2 Thermal Design Transients for Normal Conditions

Table 6-1 provides the number of occurrences expected for the different transients that are qualified as normal condition transients. Pressure and temperature boundaries were determined for each of these transients.

Based upon the plant history, there were less than 100 occurrences of plant heatup and cooldown for the original steam generator (OSG). The original plant design called for 500 plant heatup and cooldown transients during the 40-year life of the plant. Therefore, there remain 400 occurrences of plant heatup and cooldown transients for the original RCS components. The plant heatup and cooldown transient is given separate consideration for the original steam generators and the replacement steam generators.

The maximum pressure differential across the replacement SG tubes and tubesheet is 1600 psi. This limits the maximum RCS pressure during heatup and cooldown, since the SG pressure will be below its normal operating pressure. The maximum RCS pressure is 1614.7 psia when the SG is not being used as a heat sink or is inoperable. The maximum RCS pressure to which both the tube sheet and tubes will be exposed occurs in the SG inlet plenum. The minimum pressure to which the tubes will be exposed occurs at the top of the tube bundle, assumed to be at saturation conditions or 14.7 psia, whichever is greater. Based upon TS 3.4.1.1 and 3.4.1.2, at least one RCP shall be in operation above 300° F. Hence, above 300° F, there is a transition to a higher pressure because of the RCP seal limits or NPSH limits. Otherwise, the lower pressure limit is based upon the saturation pressure at a given temperature. The upper limit on normal system operating pressure is taken to be 2250 psia for these design transients, which is higher than the plant operating pressure of 2200 psia. Use of 2250 psia conservatively covers a larger pressure range for fatigue analyses and covers potential future plant operating procedure changes. The use of 2250 psia affects the upper end of the heatup and cooldown curves but has no effect on other design transients that depend on RCS pressure changes rather than the absolute magnitude of pressure.

Plant loading (heatup) and plant unloading (cooldown) transients are evaluated. Plant loading and unloading conditions apply between 15% to 100% power. Plant operations between 0% to 15% power are covered by the normal plant variation and cold feedwater categories. The plant loading and unloading is at a rate of 5% full power per minute. The evaluation includes the expected feedwater temperature versus percent power both for Cycle 15 and Cycle 16 forward (power uprate).

A reactor coolant pump start-up (or shutdown) transient takes a finite amount of time (≈15 to 20 seconds) for the pump to come up to full speed and flow rate. Flow rates and associated pressure differentials across the RCS will build up smoothly to the steady-state conditions. The resulting steady state pressure differential across the primary head divider plate due to starting and stopping reactor coolant pumps is 92 psi maximum in either direction for normal conditions.

6.4.3 Thermal Design Transients for Upset Conditions

Table 6-3 provides the number of occurrences expected for the different transients that are categorized as upset condition transients. A composite transient with 480 cycles was developed based on plant responses from reactor trip (400 occurrences), loss of primary flow (40 occurrences) and loss of load (40 occurrences). A composite transient was developed because these transients exhibit similar characteristics. A composite transient was utilized for the RCS that bounds the reactor trip, loss of reactor coolant flow and loss of load upset condition. The maximum change in SG pressure occurs for the loss of load transient. Consequently, that transient is used to provide the bounding parameters for the composite SG transient. The feedwater flow and temperature response input to the composite SG transient includes the effect of relatively cold emergency feedwater flow mixing with the main feedwater flow.

6.4.4 Thermal Design Transients for Emergency and Faulted Conditions

Tables 6-4 and 6-5 provide the number of occurrences expected for the different transients defined as emergency condition and faulted condition transients respectively. The original design number of cycles is retained for the various events in these categories, except for the loss of feedwater flow event, whose frequency of occurrence has increased due to plant operating experience. Only one occurrence needs to be considered for a faulted condition.

The loss of secondary pressure transient was evaluated. This transient appears in both Level C and Level D categories for the RSG design. The Level C category covers less severe events such as excess steam demand and the inadvertent opening of a turbine bypass valve. The Level D category covers the complete severance of the main steam line (MSLB). The Level D response bounds the Level C response. The Level C transient is specified to recover to 2250 psia while the Level D transient recovers to 2500 psia.

The loss of feedwater flow transient (Level C) was evaluated including the effects of emergency feedwater (EFW) addition. EFW valve stroke times and pump delay and run-up times are also considered. The EFW response is cyclic in nature due to the small difference in SG level wherein the EFW is actuated and is turned off. It is assumed that the cyclic EFW response remains until 1800 seconds when it was assumed that the EFW is throttled to a constant flow to maintain SG level.

The feedwater line break transient is included as a Level D transient. For small feedwater line break sizes, this event is a heatup event. A large feedwater line break size would make this a cooldown event that is covered by the loss of secondary pressure event.

6.4.5 Leak Testing

Table 6-6 presents transient lifetime occurrences for test conditions. Leak testing is covered under Section XI of the ASME code. Section XI permits leak tests in lieu of hydrostatic tests. Consequently, the hydrostatic tests are no longer required to be analyzed for fatigue requirements. Additionally, system leakage tests can be used in lieu of hydrostatic testing for installation of replacement items by welding. The ANO-2 plant satisfies Section XI by performing leak testing at hot standby (temperatures >400°F.) Normal operating procedures also ensure limited leakage during power operation.

In general, since leak testing at nominal operating pressure is done in conjunction with normal plant operation, there is no requirement to analyze leak testing with respect to fatigue considerations, except for the special secondary side tests associated with the SG. For the RSG, there is only one set of special conditions for which additional pressure cycles due to leak testing must be considered. These are denoted as Cases 1, 2, 3 and 4 in Table 6-6. These cases represent pressurizing the secondary side of the SG (while the primary side is completely depressurized during an outage) to determine which tube is leaking. Since all the tubes are exposed to this pressure cycle, fatigue analysis accounted for these pressure cycles.

6.4.6 Summary Results

The NSSS fluid transients for the reactor vessel, RCS main piping, pressurizer, reactor coolant pumps and steam generators have changed as a result of the replacement steam generator and the 107.5% power uprate for the remaining design life of the ANO-2 plant. The effect of these changes on the fatigue life of the subject components is defined in Section 5 of this report.

Table 6-1
Transient Lifetime Occurrences for Normal Conditions

<table>
<tr><th>Transient</th><th>Reactor Vessel</th><th>Reactor Coolant Pump</th><th>Pressurizer</th><th>Reactor Coolant Pipe and Fittings</th><th>Replacement Steam Generator</th></tr>
<tr><td>Plant heatup, 100 °F/hr</td><td>500</td><td>500</td><td>NA</td><td>500</td><td>350</td></tr>
<tr><td>Plant heatup, 200 °F/hr</td><td>NA</td><td>NA</td><td>500</td><td>NA</td><td>NA</td></tr>
<tr><td>Plant cooldown, 100 °F/hr</td><td>500</td><td>500</td><td>NA</td><td>500</td><td>350</td></tr>
<tr><td>Plant cooldown, 200 °F/hr and flooding(1)</td><td>NA</td><td>NA</td><td>500</td><td>NA</td><td>NA</td></tr>
<tr><td>Plant loading, 5%/min</td><td>15,000</td><td>15,000</td><td>10^6</td><td>15,000</td><td>12,000</td></tr>
<tr><td>Plant unloading, 5%/min</td><td>15,000</td><td>15,000</td><td>10^6</td><td>15,000</td><td>12,000</td></tr>
<tr><td>10% step load increase</td><td rowspan="3">10^6</td><td rowspan="3">10^6</td><td rowspan="3">10^6</td><td rowspan="3">10^6</td><td>2000</td></tr>
<tr><td>10% step load decrease</td><td>2000</td></tr>
<tr><td>Normal plant variation</td><td>10^6</td></tr>
<tr><td>Pump starting and stopping(2)</td><td>NA</td><td>4000</td><td>NA</td><td>NA</td><td>4000</td></tr>
<tr><td>Cold feedwater following hot standby</td><td>NA</td><td>NA</td><td>NA</td><td>NA</td><td>See FW nozzle additional requirements, Table 6-2</td></tr>
</table>

Notes:

(1) Flooding during cooldown: With an initial water volume equal to that required to cover the heaters and at an initial fluid temperature of 400°F, the pressurizer is filled with 40°F spray water at 133 gpm. The temperature of the water in the pressurizer during filling shall be the mixed mean fluid temperature.

(2) A reactor coolant pump start-up (or shutdown) transient takes a finite amount of time (≈15 to 20 seconds) for the pump to come up to full speed and flow rate or to shutdown from full speed. Flow rates and associated pressure differentials across the RCS will build up or down smoothly to the steady-state conditions.

Table 6-2
FW Nozzle Additional Requirements

Event Description	Design Time Duration, hrs	Slugging Flow (On-Off) Time Period, hours	Continuous Flow or Trickle Flow Time Period, hours	Number of Times at This Condition	Number of Slugging Flow Cycles (On-Off)	Maximum Slugging Flow Rate, gpm	Lifetime Slugging Flow Cycles	Continuous or Trickle Flow Rate, gpm
Heatup	24	6	18	350	10	300	3,500	0+ to 100
Hot Standby - Heatup	4	0+	4	350	2	785	700	48 (nominal)
Hot Standby - Cooldown	4	0+	4	500	5	785	2500	128 (nominal)
Cooldown	24	0+	24	350	10	450	3,500	0+ to 250
	Design Time Duration, hrs	**Flow Ramp**	**Cycles per Hour**	**Number of Times at This Condition**	**Fluctuating Flow lbm/hr**		**Lifetime Cycles**	
0 to 15% Power and 15 to 0%	3 to 10	Flow ramps between 24,060 to 835,000 lbm/hr	---	500 up power & 500 down power maneuvers	10% of nominal		---	

Notes:

(1) The FW temperature during heatup, hot standby and cooldown shall be assumed to be 70°F.

(2) During Hot-Standby, it can be assumed that RCS pressure = 2250 ± 100 psi, T_{avg} = 545 ± 10°F, SG pressure = 1000 psia ± 80 psi.

(3) 0+ signifies there is always a small amount of FW flow due to leakage by the FW valves.

(4) Hot Standby-Cooldown includes additional no. of cycles to account for the times when plant is at hot standby for trouble shooting purposes.

(5) During heatup, cooldown, and hot standby it can be assumed that slugging flow is uniformly distributed during these specific time periods. Slugging flow and continuous flow are two distinct modes of operation and both need to be considered.

(6) The 500 power up and power down design cycles is based on engineering judgement and is higher than the 350 heatup/cooldown cycles to account for times when low power FW control may be unstable or when the plant operators are trying to isolate a plant problem.

Table 6-3
Transient Lifetime Occurrences for Upset Conditions

Transient	Reactor Vessel	Reactor Coolant Pump	Pressurizer	Reactor Coolant Pipe and Fittings	Replacement Steam Generator
Reactor trip Loss of reactor coolant flow Loss of load	480	480	480	480	480
Operating basis earthquake + normal operation at full power	200 [(1)]	200 [(1)]	200 [(1)]	200 [(1)]	200 [(1)]

Note:

(1) Both horizontal and vertical loading/accelerations shall be assumed to cycle about a mean value of zero for a total of 200 full cycles. The number of cycles is based on the postulation of five seismic events occurring during the life of the plant with 40 full cycles per occurrence of significant motion peaks.

Table 6-4
Transient Lifetime Occurrences for Emergency Conditions

Transient	Reactor Vessel	Reactor Coolant Pump	Pressurizer	Reactor Coolant Pipe and Fittings	Replacement Steam Generator
Loss of secondary pressure	5	5	5	5	5
Loss of feedwater flow	NA	NA	NA	NA	20

Table 6-5
Transient Lifetime Occurrences for Faulted Conditions

Transient	Reactor Vessel	Reactor Coolant Pump	Pressurizer	Reactor Coolant Pipe and Fittings	Replacement Steam Generator
Design Basis Earthquake + Normal Operation	1	1	1	1	1
Design Basis Earthquake + Normal Operation + RCS Pipe Rupture	1	1	1	1	1
Design Basis Earthquake + Normal Operation + Main Steam Line Break	NA	NA	NA	NA	1
Design Basis Earthquake + Normal Operation + Feedwater Line Break	NA	NA	NA	NA	1

Table 6-6
Transient Lifetime Occurrences for Test Conditions

Test Conditions	Reactor Vessel	Reactor Coolant Pump	Pressurizer	Reactor Coolant Pipe and Fittings	Replacement Steam Generator	
					Primary Side	Secondary Side
Hydrostatic test, 3125 psia 100°F - 400°F	1	1	1	1	---	---
Leak test, 2250 psia 100°F - 400°F	---	---	---	---	---	---
Tube side hydrostatic test by manufacturer 70°F - 150°F	---	---	---	---	1 primary side at 3125 psig	1 secondary side at 0 psig
Shell side hydrostatic test by manufacturer in field. 70°F - 150°F	---	---	---	---	1 primary side at 0 psig	1 secondary side at 1375 psig
Tube side leak test	---	---	---	---		
Shell side leak test	---	---	---	---		
Tube leak test – Case 1 70°F - 250°F	---	---	---	---	400 at 0 psig	400 at 200 psig
Tube leak test – Case 2 70°F - 250°F	---	---	---	---	200 at 0 psig	200 at 400 psig
Tube leak test – Case 3 70°F - 250°F	---	---	---	---	120 at 0 psig	120 at 600 psig
Tube leak test – Case 4 70°F - 250°F	---	---	---	---	80 at 0 psig	80 at 840 psig

END OF SECTION