NMC
Committed to Nuclear Excellence

**Palisades Nuclear Plant**
Operated by Nuclear Management Company, LLC

June 3, 2003 10 CFR 50.90

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

PALISADES NUCLEAR PLANT
DOCKET 50-255
LICENSE No. DPR-20
LICENSE AMENDMENT REQUEST: INCREASE RATED THERMAL POWER

Pursuant to 10 CFR 50.90, Nuclear Management Company, LLC (NMC) requests Nuclear Regulatory Commission (NRC) review and approval of a license amendment for the Palisades Nuclear Plant. NMC proposes to revise Facility Operating License DPR-20, including Appendix A, Technical Specifications (TS) to increase rated thermal power (RTP) by 1.4% from 2530 Megawatts thermal (MWt) to 2565.4 MWt.

Enclosure 1 provides a description of the proposed change, background, No Significant Hazards Consideration Determination, and Environmental Review Consideration. Enclosure 2 provides the revised Operating License (OL) and TS pages reflecting the proposed change. Enclosure 3 provides the annotated OL and TS pages showing the changes proposed. Enclosure 4 provides a summary of the measurement uncertainty recapture evaluation following the guidance provided in Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

One previous submittal affects the technical basis for this submittal and requires approval to support this submittal: Lahti (NMC) to NRC, "Nuclear Management Company, LLC – Palisades Nuclear Plant, Docket 50-255, License DPR-20 - License Amendment Request: Thermal Margin/Low Pressure Trip," October 17, 2002.

NMC requests approval of this proposed license amendment by December 16, 2003 in order to accommodate implementation in a timely manner. NMC further requests a 90-day implementation period following amendment approval.

A copy of this request has been provided to the designated representative of the State of Michigan.



A001

## SUMMARY OF COMMITMENTS

This letter contains the following new commitments and no changes to existing commitments:

- NMC will conduct operator training on the proposed power uprate prior to implementation of the proposed power uprate.

- NMC will revise plant procedures to address operation with the Crossflow ultrasonic flow measurement system out of service prior to implementation of the proposed power uprate.

I declare under penalty of perjury that the foregoing is true and accurate. Executed on June 3, 2003.

Douglas E. Cooper
Site Vice-President, Palisades

CC Administrator, Region III, USNRC
Project Manager, Palisades Plant, NRR, USNRC
NRC Resident Inspector – Palisades Plant

Enclosures

**ENCLOSURE 1**

**NUCLEAR MANAGEMENT COMPANY, LLC**
**PALISADES PLANT**
**DOCKET 50-255**

**LICENSE AMENDMENT REQUEST PURSUANT TO 10 CFR 50.90:**
**INCREASE RATED THERMAL POWER**

5 Pages Follow

PALISADES NUCLEAR PLANT
DOCKET 50-255

## 1.0 INTRODUCTION

Nuclear Management Company, LLC (NMC) requests to amend Operating License DPR-20, including Appendix A, Technical Specifications (TS), for the Palisades Nuclear Plant. The Palisades Nuclear Plant is presently licensed to operate at 2530 Megawatts thermal (MWt). The request to increase the rated thermal power (RTP) by 1.4% to 2565.4 MWt is based on reduced core power measurement uncertainty resulting from the use of more accurate feedwater flow measurement instrumentation.

## 2.0 DESCRIPTION OF THE PROPOSED AMENDMENT

NMC requests to:

- Revise paragraph 2.C.(1) of Facility Operating License DPR-20 to authorize operation at steady-state reactor core power levels not in excess of 2565.4 Megawatts thermal (100 percent rated power).
- Revise the definition of rated thermal power (RTP) in Appendix A, TS 1.1, from 2530 MWt to 2565.4 MWt.
- Revise the maximum allowable value for the Variable High Power Trip from 111% to 109.4% in Appendix A, TS Table 3.3.1-1, item 1.

## 3.0 BACKGROUND

Palisades Nuclear Plant is presently licensed for steady-state reactor core power of 2530 MWt. In June 2000, the Nuclear Regulatory Commission (NRC) approved a change to 10 CFR 50, Appendix K, providing licensees the option of maintaining the 2 percent power margin between the licensed core power level and the assumed core power level for emergency core cooling system (ECCS) evaluations, or applying a reduced margin to the ECCS evaluations.

The Crossflow ultrasonic flow measurement system has been in use at Palisades Nuclear Plant for feedwater measurement since 1997. The core power measurement uncertainty using Crossflow has been determined to be less than 0.59 percent. Therefore, it is proposed to reduce the power measurement uncertainty required by 10 CFR 50, Appendix K, from 2% to ≤ 0.5925% to permit an increase in the licensed power level by 1.4% from 2530 MWt to 2565.4 MWt. The impact of the proposed changes has been evaluated on the nuclear steam supply system (NSSS) and balance of plant (BOP) systems, components and safety analyses. Enclosure 4 provides this evaluation, which was prepared in accordance with the guidance in NRC Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," January 31, 2002.

The proposed power uprate also requires the maximum allowable value for the Variable High Power Trip (VHPT) be changed from 111% to 109.4% in TS Table 3.3.1-1, item 1. The current VHPT allowable value was determined for 2530 MWt. The proposed VHPT allowable value was determined for the proposed uprated power level of 2565.4 MWt.

## 4.0 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Nuclear Management Company, LLC (NMC) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment." The following evaluation supports the finding that operation of the facility in accordance with the proposed change would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed increase in power level is achieved by the taking credit for the accuracy of the existing feedwater flow measurement instrumentation, including the Crossflow ultrasonic flow measurement (UFM) system, which results in a more accurate feedwater flow used in the heat balance calculation. The increased flow accuracy utilizing the Crossflow UFM system improves the uncertainty in the core power level from the existing 2% margin to ≤ 0.5925%. The probability of an accident previously evaluated is not increased by the proposed change because the flow measurement instrumentation is not an initiator of design-basis accidents evaluated in the updated final safety analysis report.

   The plant design and licensing basis has been evaluated for operation at the proposed increased value of 2565.4 Megawatts thermal (MWt). All systems and components continue to acceptably perform their structural and operational functions.

There are no changes as a result of the proposed measurement uncertainty recapture power uprate to the design or operation of the plant that could affect system, component, or accident mitigative functions. All systems and components will function as designed and the applicable performance requirements have been evaluated and found to be acceptable. The proposed variable high power trip allowable value will ensure that the maximum actual steady state power at which a trip would be actuated is within safety analysis limits.

Therefore, there is no significant increase in the probability of an accident previously evaluated.

The reduction in power measurement uncertainty is bounded by the safety analyses since they were performed or evaluated at 2580.6 MWt. Radiological consequences of Chapter 14 accidents were assessed previously and continue to be bounding. The FSAR Chapter 14 analyses continue to demonstrate compliance with the relevant accident analysis acceptance criteria. Therefore, there is no significant increase in the consequences of any accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed change. All systems, structures and components previously required for the mitigation of an event remain capable of fulfilling their intended design function at the proposed uprated power level. The proposed change has no adverse effects on any safety-related systems or component and does not challenge the performance or integrity of any safety-related system. The proposed variable high power trip allowable value will ensure that the maximum actual steady state power at which a trip would be actuated is within safety analysis limits. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The maximum steady-state reactor power of 2580.6 MWt assumed in the accident analysis, including uncertainties, remains the same as previously analyzed. Therefore, the change in rated thermal power to 2565.4 MWt does not involve a significant reduction in the margin of safety.

The current accident analyses and system and component analyses had been previously performed at core powers that exceed the proposed measurement uncertainty recapture uprated core power. Evaluations have been performed for analyses that were done at nominal core power and have been found acceptable for the proposed measurement uncertainty recapture power uprate. Analyses of the primary fission product barriers at uprated core powers have concluded that all relevant design basis criteria remain satisfied in regard to integrity and compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been either reviewed and approved by the Nuclear Regulatory Commission or are in compliance with applicable regulatory review guidance and standards. The proposed variable high power trip allowable value will ensure that the maximum actual steady state power at which a trip would be actuated is within safety analysis limits. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the evaluation above, NMC has determined that the proposed change does not involve significant hazards consideration.

## 5.0 ENVIRONMENTAL REVIEW CONSIDERATION

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

## 6.0 CONCLUSION

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. The Palisades Plant Review Committee has reviewed this amendment request and has determined that the change involves no significant hazards consideration. The Palisades Offsite Safety Review Committee has also concurred in this determination.

**ENCLOSURE 2**

**NUCLEAR MANAGEMENT COMPANY, LLC**
**PALISADES PLANT**
**DOCKET 50-255**

**LICENSE AMENDMENT REQUEST PURSUANT TO 10 CFR 50.90:**
**INCREASE RATED THERMAL POWER**

REVISED FACILITY OPERATING LICENSE PAGE 3
REVISED TECHNICAL SPECIFICATION PAGES 1.1-5 AND 3.3.1-6
AND
PAGE CHANGE INSTRUCTIONS

4 Pages Follow

**ATTACHMENT TO LICENSE AMENDMENT NO.**

**FACILITY OPERATING LICENSE NO. DPR-20**

**DOCKET NO. 50-255**

Replace the following page of Operating License No. DPR-20 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

| REMOVE | INSERT |
|---|---|
| 3 | 3 |

Replace the following pages of Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

| REMOVE | INSERT |
|---|---|
| 1.1-5 | 1.1-5 |
| 3.3.1-6 | 3.3.1-6 |

C. This license shall be deemed to contain and is subject to all applicable provisions of the Act; 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) NMC is authorized to operate the facility at steady-state reactor core power levels not in excess of 2565.4 Megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) The Technical Specifications contained in Appendix A, as revised through Amendment No. 2xx, and the Environmental Protection Plan contained in Appendix B are hearby incorporated in the license. NMC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) NMC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the SERs dated 09/01/78, 03/19/80, 02/10/81, 05/26/83, 07/12/85, 01/29/86, 12/03/87, and 05/19/89 and subject to the following provisions:

a. NMC may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

b. NMC may alter specific features of the approved fire protection program provided:

- Such changes do not result in failure to complete the fire protection program as approved by the Commission. NMC shall maintain in auditable form, a current record of all such changes, including an analysis of the effects of the change on the fire protection program and shall make such records available to the Commission Inspectors upon request. All changes to the approved program shall be reported along with the FSAR revision as required by 10 CFR 50.71(e); and

- Temporary changes to specific fire protection features which may be necessary to accomplish maintenance or modifications are acceptable provided interim compensatory measures are implemented.

(4) Upon implementation of Amendment No. 189, the schedule for performance of new or revised surveillance requirements (SRs) shall be as follows:

- For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval that begins on the date of implementation of this amendment.

## 1.1 Definitions

| | |
|---|---|
| PHYSICS TESTS | PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:<br><br>a. Described in Chapter 13, Initial Tests and Operation, of the FSAR;<br><br>b. Authorized under the provisions of 10 CFR 50.59; or<br><br>c. Otherwise approved by the Nuclear Regulatory Commission. |
| QUADRANT POWER TILT ($T_q$) | $T_q$ shall be the maximum positive ratio of the power generated in any quadrant minus the average quadrant power, to the average quadrant power. |
| RATED THERMAL POWER (RTP) | RTP shall be a total reactor core heat transfer rate to the primary coolant of 2565.4 MWt. |
| REFUELING BORON CONCENTRATION | REFUELING BORON CONCENTRATION shall be a Primary Coolant System boron concentration of ≥ 1720 ppm and sufficient to assure the reactor is subcritical by ≥ 5% $\Delta\rho$ with all control rods withdrawn. |
| SHUTDOWN MARGIN (SDM) | SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:<br><br>a. All full length control rods (shutdown and regulating) are fully inserted except for the single rod of highest reactivity worth, which is assumed to be fully withdrawn. However, with all full length control rods verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation. With any full length control rods not capable of being fully inserted, the reactivity worth of these rods must be accounted for in the determination of SDM; and<br><br>b. There is no change in part length rod position. |

Table 3.3.1-1 (page 1 of 2)
Reactor Protective System Instrumentation

| FUNCTION | APPLICABLE MODES | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|
| 1. Variable High Power Trip | 1,2,3[(a)],4[(a)],5[(a)] | SR 3.3.1.1<br>SR 3.3.1.2<br>SR 3.3.1.3<br>SR 3.3.1.4<br>SR 3.3.1.5<br>SR 3.3.1.6<br>SR 3.3.1.8 | ≤ 15% RTP above current THERMAL POWER with a minimum of ≤ 30% RTP and a maximum of ≤ 109.4% RTP |
| 2. High Startup Rate Trip[(b)] | 1,2 | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.8 | NA |
| 3. Low Primary Coolant System Flow Trip[(c)] | 1,2,3[(a)],4[(a)],5[(a)] | SR 3.3.1.1<br>SR 3.3.1.5<br>SR 3.3.1.8 | ≥ 95% |
| 4. Low Steam Generator A Level Trip | 1,2,3[(a)],4[(a)],5[(a)] | SR 3.3.1.1<br>SR 3.3.1.5<br>SR 3.3.1.8 | ≥ 25.9% narrow range |
| 5. Low Steam Generator B Level Trip | 1,2,3[(a)],4[(a)],5[(a)] | SR 3.3.1.1<br>SR 3.3.1.5<br>SR 3.3.1.8 | ≥ 25.9% narrow range |
| 6. Low Steam Generator A Pressure Trip[(c)] | 1,2,3[(a)],4[(a)],5[(a)] | SR 3.3.1.1<br>SR 3.3.1.5<br>SR 3.3.1.8 | ≥ 500 psia |
| 7. Low Steam Generator B Pressure Trip[(c)] | 1,2,3[(a)],4[(a)],5[(a)] | SR 3.3.1.1<br>SR 3.3.1.5<br>SR 3.3.1.8 | ≥ 500 psia |
| 8. High Pressurizer Pressure Trip | 1,2,3[(a)],4[(a)],5[(a)] | SR 3.3.1.1<br>SR 3.3.1.5<br>SR 3.3.1.8 | ≤ 2255 psia |

(a) With more than one full-length control rod capable of being withdrawn and PCS boron concentration less than REFUELING BORON CONCENTRATION.

(b) Trip may be bypassed when Wide Range Power is < 1E-4% RTP or when THERMAL POWER is > 13% RTP.

(c) Trips may be bypassed when Wide Range Power is < 1E-4% RTP. Bypass shall be automatically removed when Wide Range Power is ≥ 1E-4% RTP.

**ENCLOSURE 3**

**NUCLEAR MANAGEMENT COMPANY, LLC**
**PALISADES PLANT**
**DOCKET 50-255**

**LICENSE AMENDMENT REQUEST PURSUANT TO 10 CFR 50.90:**
**INCREASE RATED THERMAL POWER**

MARK-UP OF REVISED FACILITY OPERATING LICENSE PAGE 3
MARK-UP OF REVISED TECHNICAL SPECIFICATION PAGES 1.1-5 AND 3.3.1-6
(Showing proposed changes)
(additions are double underlined; deletions are strikethrough)

3 Pages Follow

C. This license shall be deemed to contain and is subject to all applicable provisions of the Act; 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:

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$T_q$ shall be the maximum positive ratio of the power generated in any quadrant minus the average quadrant power, to the average quadrant power.

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the primary coolant of ~~2530~~2565.4 M~~w~~Wt.

REFUELING BORON CONCENTRATION

REFUELING BORON CONCENTRATION shall be a Primary Coolant System boron concentration of ≥ 1720 ppm and sufficient to assure the reactor is subcritical by ≥ 5% Δρ with all control rods withdrawn.

SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All full length control rods (shutdown and regulating) are fully inserted except for the single rod of highest reactivity worth, which is assumed to be fully withdrawn. However, with all full length control rods verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation. With any full length control rods not capable of being fully inserted, the reactivity worth of these rods must be accounted for in the determination of SDM; and

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| 2. High Startup Rate Trip[b] | 1,2 | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.8 | NA |
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| 4. Low Steam Generator A Level Trip | 1,2,3[a],4[a],5[a] | SR 3.3.1.1<br>SR 3.3.1.5<br>SR 3.3.1.8 | ≥ 25.9% narrow range |
| 5. Low Steam Generator B Level Trip | 1,2,3[a],4[a],5[a] | SR 3.3.1.1<br>SR 3.3.1.5<br>SR 3.3.1.8 | ≥ 25.9% narrow range |
| 6. Low Steam Generator A Pressure Trip[c] | 1,2,3[a],4[a],5[a] | SR 3.3.1.1<br>SR 3.3.1.5<br>SR 3.3.1.8 | ≥ 500 psia |
| 7. Low Steam Generator B Pressure Trip[c] | 1,2,3[a],4[a],5[a] | SR 3.3.1.1<br>SR 3.3.1.5<br>SR 3.3.1.8 | ≥ 500 psia |
| 8. High Pressurizer Pressure Trip | 1,2,3[a],4[a],5[a] | SR 3.3.1.1<br>SR 3.3.1.5<br>SR 3.3.1.8 | ≤ 2255 psia |

(a) With more than one full-length control rod capable of being withdrawn and PCS boron concentration less than REFUELING BORON CONCENTRATION.

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**ENCLOSURE 4**

**NUCLEAR MANAGEMENT COMPANY, LLC**
**PALISADES PLANT**
**DOCKET 50-255**

**LICENSE AMENDMENT REQUEST PURSUANT TO 10 CFR 50.90:**
**INCREASE RATED THERMAL POWER**

Measurement Uncertainty Recapture Evaluation Following Guidance Provided in Regulatory Issue Summary 2002-03

42 Pages Follow

**Measurement Uncertainty Recapture Evaluation Following Guidance Provided in Regulatory Issue Summary 2002-03**

This enclosure provides the evaluation, which was prepared in accordance with the guidance in NRC Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," January 31, 2002.

## I. Feedwater flow measurement technique and power measurement uncertainty

1. Best estimate feedwater flow is determined using the Crossflow ultrasonic flow measurement (UFM) System. The measured UFM feedwater flow is then used to correct the feedwater flow that is continuously measured by feedwater venturi and input to the plant heat balance calculation.

   A. The UFM system at Palisades is described in the Combustion Engineering topical report "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," CENPD-397-P-A, Revision 01, May 2000.

   B. The Nuclear Regulatory Commission (NRC) safety evaluation and approval to use the Crossflow ultrasonic flow measurement technology is included in the approved report and can be found in correspondence Richards (NRC) to Rickard (ABB), "Acceptance for Referencing of CENPD-397-P, Revision 01-P, 'Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology' (TAC No. MA6452)," March 20, 2000.

   C. All guidelines specified in Topical Report CENPD-397-P-A, Revision 01 and the NRC staff letter/safety evaluation have been implemented at Palisades.

   D. The NRC safety evaluation specifically identified that the following information should be addressed.

      *(1) The licensee should discuss the development of maintenance and calibration procedures that will be implemented with the Crossflow UFM installation. These procedures should include process and contingencies for an inoperable Crossflow UFM and the effect on thermal power measurement and plant operation.*

      Maintenance and calibration of the UFM System components are performed using Nuclear Management Company, LLC (NMC) site work control processes. These processes direct the performance of NMC site-specific procedures, or a combination of NMC and vendor specific procedures.

      The Crossflow System is not connected to the plant process computer. It does not perform any automatic safety related or plant control functions. The Crossflow System is used to determine best estimate feedwater flow in support of calculating correction factors that are manually input into the Plant Process Computer and used to correct the feedwater flow as measured by venturi. If the UFM System is inoperable, then the system is either repaired to operable status within the allowed outage time, or power is reduced and the Plant Process Computer feedwater flow correction factors are manually reset (i.e., no feedwater flow correction or credit for UFM calculations).

*(2) For plants that currently have the Crossflow UFM installed, the licensee should provide an evaluation of the operational and maintenance history of the installed UFM and confirm that the instrumentation is representative of the Crossflow UFM and is bounded by the requirements set forth in Topical Report CENPD-397-P.*

The Crossflow System has been in use since 1997 at Palisades. The system has proven to be very reliable and is presently used to correct for fouling of feedwater flow venturi nozzles allowing operation at near 100% rated thermal power. Considerable experience has been gained setting up and tuning the system, such that minimal interaction is now required. Confidence in system reliability has developed significantly over this time period, resulting in a reduction in the system surveillance frequency.

UFM system reliability has been good following replacement of probes and brackets during the 1999 refueling outage (beginning of Cycle 15). No work has been required on Loop A and only three Work Orders (W/O) have been performed on Loop B since this probe replacement. The root cause of the first W/O was a failed channel on the multiplexer, (multiple spare channels are available). Definitive root causes were not established on the remaining two work orders; however, optimization of software settings by the vendor's representative and troubleshooting in the field corrected the problems (indicating probable loose connector). A possible cause was a bent pin found on a hundred-pin connector between the signal conditioning unit and computer. The specific pin was unused but may have prevented proper connection of the data cable. The cable was replaced and no further problems have occurred. These three work orders were relatively minor problems readily addressed with vendor support. UFM system reliability has been very good during the past two years.

A recurrent communications problem experienced with the UFM system computer in containment is not reflected in the work order history (work orders were not required to correct the problem). Data for analysis is transferred from the system in containment to an office computer via modem. Communications failures do not affect UFM system operability but required containment entries (to a low dose area) to download data and reboot the computer. This communications problem was corrected by an upgrade of computers and software. No containment entries for UFM data collection have been required since these upgrades were implemented.

The currently installed Crossflow system is representative of the Crossflow UFM described in Topical Report CENPD-397-P-A, Revision 01, and is bounded by the requirements set forth in the Topical Report.

(3) *The licensee should confirm that the methodology used to calculate the uncertainty of the Crossflow UFM in comparison to the current feedwater flow instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative methodology is used, the application should be justified and applied to both the venturi and the Crossflow UFM for comparison.*

The Crossflow uncertainty calculation indicates a mass flow accuracy of better than 0.5% of rated flow for the Palisades site-specific installation. The calculations are consistent with the methodology described in Topical Report CENPD-397-P-A, Revision 01. The uncertainty calculations specify requirements for 95% confidence interval flow measurement including:

- Inside pipe diameter measurement and associated uncertainty
- Transducer spacing measurement and associated uncertainty
- Velocity profile correction factor and justification
- Crossflow time delay calibration data and associated uncertainty

The Crossflow flow uncertainty calculation supports an uncertainty in the reactor power measurement of less than 0.59%. These calculations are based on accepted plant instrument uncertainty methodology, which incorporates the aspects of ANSI/ISA-S67.04.01-2000, "Methodologies For the Determination of Setpoints for Nuclear Safety-Related Instrumentation," and are included as Attachment 1 to this enclosure. Crossflow system implementing procedures ensure the assumptions and requirements of the uncertainty calculation remain valid.

(4) *The licensee of a plant at which the installed Crossflow UFM was not calibrated to a site-specific piping configuration (flow profile and meter factors not representative of the plant-specific installation) should submit additional justification. This justification should show that the meter installation is either independent of the plant-specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibration and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed and calibrated Crossflow UFM, the licensee should confirm that the plant-specific installation follows the guidelines in the Crossflow UFM topical report.*

At Palisades, the Crossflow installation is not explicitly calibrated to the site-specific piping configuration. The installation is equivalent to known calibration and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. The velocity profile correction factor is calculated as described in Section 5.6 of the Topical Report. The transducers are installed on straight pipe runs and are far enough from disturbances to conform to the proprietary installation requirements of Topical Report CENPD-397-P-A, Revision 01.

E. The total power measurement uncertainty has been revised and is included as Attachment 1 to this enclosure, "Uncertainty Calculation for the Secondary Calorimetric Heat Balance," EA-ELEC08-0001, Revision 1.

F. Calibration and Maintenance

i. Calibration of the feedwater flow and feedwater temperature instrument loops is performed at 18-month intervals. Calibration of the steam generator pressure instrument loops is performed at 18-month intervals. Calibration of the steam generator blowdown flow indicator (flow reading is manually input into the plant computer calorimetric calculation) is performed at 24-month intervals.

ii. Normal plant processes to ensure successful implementation and maintenance of plant configuration control are used to control software and hardware configuration. For example, changes made to software affecting the plant calorimetric are performed in accordance with a plant procedure, which describes the actions and requirements necessary to ensure appropriate control of software changes and configuration is maintained. Changes to instrumentation that affect the plant calorimetric are processed in accordance with plant procedures which describe the actions necessary to implement intended design related changes, or enhancements, and ensure the plant's design basis is maintained. Maintenance of plant calorimetric instruments is performed in accordance with plant procedures.

iii. Corrective Actions are determined in accordance with plant procedures. The need for corrective actions is determined as part of the formal evaluation of a specific condition adverse to quality. Equipment and instruments affecting the plant calorimetric may be declared inoperable pending repairs performed in accordance with plant procedures.

iv. Instruments associated with the calorimetric are monitored by the previously described surveillance procedures in paragraph I.1.F.i. If acceptance criteria are not met, the issue is entered into the plant's corrective action program in accordance with plant procedures in order to evaluate the deficiency, or potential condition adverse to quality, and to render determination of operability. As part of the evaluation, or in parallel to the evaluation, troubleshooting or repairs may be performed on the instruments. During the process, notifications and information are provided to the vendor, as necessary. Site approved vendor manuals are maintained and used, which describe the troubleshooting that may be performed on-site.

The manuals typically provide recommendations to contact the vendor when troubleshooting options are exhausted. A similar process is used for software related deficiencies with entry into the corrective action program in accordance with plant procedures. NMC software quality assurance procedures require the software custodian notifies all users of the code if the code results are incorrect. This implies notification of the vendor, if the software is vendor supplied.

v. Manufacturer deficiency reports are processed per plant procedures. Deficiency reports in the form of Bulletins, Technical Notes, 10 CFR 21 notices, among others, are initially screened as to the need for applicability and/or further evaluation. If an evaluation is needed, the item is documented into the plant's corrective action program in accordance with plant procedures. The evaluation process ensures that notifications to the NRC are made if the issue meets the requirements of 10 CFR Part 21.

G. The allowed outage time for the UFM System is administratively controlled in the same manner as Technical Specification surveillances. The UFM System may remain out of service for a period no longer than the normal calibration interval, currently specified as 31 days, with up to 25% grace period.

H. Prior to exceeding the allowed outage time, the reactor power level will be reduced to 2550.0 MWt, or 99.4% of rated thermal power. This power level is consistent with the feedwater flow uncertainty analysis based on feedwater flow measurement with venturi instrumentation only. Reactor power, plus the uncertainty in reactor power, remains less than the analyzed power level of 2580.6 MWt. Power measurement uncertainty calculations are contained on page 17 of Attachment 1 to this enclosure.

| Crossflow UFM | Rated Thermal Power (%) | Power Level (MWt) | Power Measurement Uncertainty (%) | Power Measurement Uncertainty (MWt) | Power level plus uncertainty (MWt) |
|---|---|---|---|---|---|
| In service | 100.0 | 2565.4 | 0.49% | 12.6 | 2578.0 |
| Out of service | 99.4 | 2550.0 | 1.13% | 28.815 | 2578.8 |

## II. Accident/transient analyses for which the existing analyses of record bound plant operation at the proposed uprated power level

The plant parameters listed below are the accident and transient analysis design input conditions for rated thermal power of 2530 MWt and for the proposed rated thermal power of 2565.4MWt:

| | Pre-Uprate | Post-Uprate |
|---|---|---|
| Reactor Power | 2530 MWt | 2565.4 MWt |
| Cold Leg Temperature | 537.3 °F | 537.0 °F |
| Hot Leg Temperature | 582.7 °F | 583.0 °F |
| Steam Generator Pressure | 770 psia | 765.8 psia |
| Main Feedwater Temperature | 439.5°F | 440.7°F |
| Main Steam Flow | 11.114 Mlbm/hr | 11.297 Mlbm/hr |
| Main Feed Flow | 11.174 Mlbm/hr | 11.357 Mlbm/hr |
| Steam Generator Liquid Inventory | 133,593 lbm | 132,531 lbm |
| Steam Generator Vapor Inventory | 8,545 lbm | 8,534 lbm |

1. For the accidents and transients included in the following matrix, the proposed uprate in power level continues to be bounded by the existing analyses of record. The NRC has previously approved these analyses, or they were conducted using methods or processes that the NRC has previously approved. A reference to the document approving the analysis, or method, is included in the NRC Approval column.

| **Accident/Transient** | **FSAR Section** | **Validity of Bounding Event Determination** | **Assumed Reactor Power Level** | **NRC Approval** |
|---|---|---|---|---|
| Uncontrolled Control Rod Withdrawal | | | | |
| Uncontrolled Control Rod Bank Withdrawal from a Subcritical or Low Power Start-up Condition | 14.2.1 | Remains Valid | Near 0 MWt | II.2.B<br>II.2.C<br>II.2.D |
| Uncontrolled Control Rod Bank Withdrawal at Power | 14.2.2 | Remains Valid | Up to 2580.6 MWt | II.2.B<br>II.2.C<br>II.2.D<br>II.2.E |
| Single Control Rod Withdrawal | 14.2.3 | Remains Valid | Up to 2580.6 MWt | II.2.B<br>II.2.C<br>II.2.D<br>II.2.E<br>II.2.G |
| Boron Dilution | | | | |
| Boron Dilution | 14.3 | Remains Valid | 0 to 2580.6 MWt | II.2.A |
| Control Rod Drop | | | | |
| Dropped Rod/Bank Event | 14.4.1/ 14.4.2 | Remains Valid | Up to 2580.6 MWt | II.2.B<br>II.2.C<br>II.2.D<br>II.2.E<br>II.2.G |
| Core Barrel Failure | | | | |
| Core Barrel Failure | 14.5 | Bounded[1] | | |

[1] Bounded by Control Rod Ejection Event (14.16)

| **Accident/Transient** | **FSAR Section** | **Validity of Bounding Event Determination** | **Assumed Reactor Power Level** | **NRC Approval** |
|---|---|---|---|---|
| | | Control Rod Misoperation | | |
| Malposition of the Part-Length Control Rod Group | 14.6.1 | Not Credible[2] | | |
| Statically Misaligned Control Rod/Bank | 14.6.2 | Remains Valid | Up to 2580.6 MWt | II.2.B<br>II.2.C<br>II.2.E<br>II.2.G |
| | | Decreased Reactor Coolant Flow | | |
| Loss of Forced Reactor Coolant Flow | 14.7.1 | Remains Valid | Up to 2580.6 MWt | II.2.B<br>II.2.C<br>II.2.D<br>II.2.E |
| Reactor Coolant Pump Rotor Seizure | 14.7.2 | Remains Valid | Up to 2580.6 MWt | II.2.B<br>II.2.C<br>II.2.D<br>II.2.E |
| | | Start-Up of an Inactive Loop | | |
| Start-Up of an Inactive Loop | 14.8 | Not Credible[3] | | |
| | | Excessive Feedwater Incident | | |
| Excessive Feedwater Incident | 14.9 | Bounded[4] | | |
| | | Increase in Steam Flow (Excess Load) | | |
| Increase in Steam Flow (Excess Load) | 14.10 | Remains Valid | Up to 2580.6 MWt | II.2.B<br>II.2.C<br>II.2.D<br>II.2.E |
| | | Postulated Cask Drop Accidents | | |
| Postulated Cask Drop Accidents | 14.11 | Remains Valid | Up to 2580.6 MWt | II.3.A |
| | | Loss of External Load | | |
| Loss of External Load | 14.12 | Remains Valid | Up to 2580.6 MWt | II.2.B<br>II.2.F |
| | | Loss of Normal Feedwater | | |
| Loss of Normal Feedwater | 14.13 | Remains Valid | Up to 2580.6 MWt | II.2.H |
| | | Steam Line Rupture Incident | | |
| Steam Line Rupture Incident | 14.14 | Remains Valid | Up to 2580.6 MWt | II.2.B<br>II.2.C<br>II.2.G<br>II.2.H<br>II.3.A |

[2] Not credible because part-length rods are not used.
[3] Prevented by operating procedures.
[4] Event bounded by Increase in Steam Flow (14.10).

| Accident/Transient | FSAR Section | Validity of Bounding Event Determination | Assumed Reactor Power Level | NRC Approval |
|---|---|---|---|---|
| Steam Generator Tube Rupture with a Loss of Offsite Power | | | | |
| Steam Generator Tube Rupture with a Loss of Offsite Power | 14.15 | Remains Valid | 2580.6 MWt | II.2.I<br>II.3.A |
| Control Rod Ejection | | | | |
| Control Rod Ejection | 14.16 | Remains Valid | Up to 2580.6 MWt | II.2.B<br>II.2.C<br>II.2.F<br>II.2.J<br>II.3.A |
| Loss-of-Coolant Accident (LOCA) | | | | |
| Large Break LOCA (LBLOCA) | 14.17.1 | Remains Valid | Up to 2580.6 MWt | II.2.K |
| Small Break LOCA (SBLOCA) | 14.17.2 | Remains Valid | Up to 2580.6 MWt | II.2.L |
| Reactor Internals Structural Behavior Following a LOCA | 14.17.3 | Remains Valid | Not dependent on power level | II.3.B |
| Containment Pressure and Temperature Analysis | | | | |
| LOCA Analysis | 14.18.1 | Remains Valid | Up to 2580.6 MWt | II.2.M<br>II.2.N |
| Main Steam Line Break (MSLB) inside Containment | 14.18.2 | Remains Valid | 0 to 2580.6 MWt | II.2.M<br>II.2.O |
| Containment Internal Structure Evaluation | 14.18.3 | Remains Valid | Not dependent on power level | II.3.B |
| Fuel Handling Incident | | | | |
| Fuel Handling Incident | 14.19 | Remains Valid | Up to 2580.6 MWt | II.3.A |
| Liquid Waste Incident | | | | |
| Liquid Waste Incident | 14.20 | Remains Valid | Up to 2580.6 MWt | II.3.A |
| Waste Gas Incident | | | | |
| Gas Decay Tank Rupture | 14.21.1 | Remains Valid | Up to 2580.6 MWt | II.3.A |
| Volume Control Tank Rupture | 14.21.2 | Remains Valid | Up to 2580.6 MWt | II.3.A |
| Maximum Hypothetical Accident | 14.22 | Remains Valid | Up to 2580.6 MWt | II.3.A |
| Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment | | | | |
| Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment | 14.23 | Remains Valid | Up to 2580.6 MWt | II.3.A |
| Control Room Radiological Habitability | | | | |
| Control Room Radiological Habitability | 14.24 | Remains Valid | Up to 2580.6 MWt | II.3.A |

| Accident/Transient | FSAR Section | Validity of Bounding Event Determination | Assumed Reactor Power Level | NRC Approval |
|---|---|---|---|---|
| Miscellaneous | | | | |
| Radiological consequences | | Remains Valid | Up to 2580.6 MWt | II.3.A |
| Natural circulation cooldown | | Remains Valid | 2570 MWt | II.3.C |
| Containment performance | | Bounded[5] | Up to 2580.6 MWt | |
| Anticipated transient without scram | | Remains Valid | 2700 MWt | II.3.D |
| Station blackout | | Remains Valid | 2580.6 MWt | II.3.E |
| Analyses to determine environmental qualification parameters | | Remains Valid | Up to 2580.6 MWt | II.3.F |
| Safe shutdown fire analysis | | Remains Valid | Up to 2580.6 MWt | II.3.G |
| Spent fuel pool cooling | | Remains Valid | Up to 2580.6 MWt | II.3.H |
| Flooding | | Remains Valid | Not dependent on power level | II.3.I |

2. Approved Methods

A. ANF-84-73 Revision 5, Appendix B (P)(A), "Advanced Nuclear Fuels Corporation Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation, July 1990.

B. XN-NF-75-21 (A), Revision 2, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," April 1975[6]

C. EMF-92-153 (P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation, March 1994.

D. XN-NF-74-5 (P)(A), Revision 2 and Supplements 1-6, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR)," October 1986[7]

[5] Containment performance addressed by Containment Pressure and Temperature Analysis (14.18), subsections LOCA Analysis (14.18.1), MSLB inside Containment (14.18.2), and Containment Internal Structure Evaluation (14.18.3).

[6] Incorporated into Technical Specifications by reference per approved methods XN-NF-82-21 (P)(A), Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, September 1983, and EMF-2310 (P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP, May 2001.

[7] Incorporated into Technical Specifications by reference per approved method ANF-84-73 Revision 5, Appendix B (P)(A), "Advanced Nuclear Fuels Corporation Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation, July 1990.

E. EMF-1961 (P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation, July 2000.

F. ANF-89-151 (P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," May 1992.

G. EMF-96-029 (P)(A), Volumes 1 and 2, "Reactor Analysis System for PWRs, Volume 1 - Methodology Descriptions, Volume 2 - Benchmarking Results," Siemens Power Corporation, January 1997.

H. EMF-2310 (P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP, May 2001.

I. CESEC-III approved in Section III of a letter from C.O. Thomas (NRC) to A. E. Scherer (CE), "Combustion Engineering Thermal-Hydraulic Computer Program CESEC-III," April 3, 1984.

J. XN-NF-78-44 (NP)(A), "A Generic Analysis of Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company, October 1983.

K. EMF-2087 (P)(A), Revision 0, "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," Siemens Power Corporation, June 1999.

L. EMF-2328 (P)(A), Revision 0, Framatome ANP, Inc., March 2001, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based."

M. NUREG-800, Standard Review Plan, Section 6.2.1.1.A (PWR Dry Containments, Including Subatmospheric Containments) states that the Containment Systems Branch uses the CONTEMPT-LT computer code. Palisades' version (CONTEMPT-LT/28) was obtained from the Energy Science and Technology Software Center. NMC's 10 CFR 50.59 process controlled the computer code modifications made.

N. The CEFLASH-4A and FLOOD3 computer codes were used for the blowdown and reflood portions of the analysis respectively. The version of CEFLASH-4A that was used in the Palisades analysis was approved by NRC in a letter from D. M. Crutchfield (NRC) to A. E. Scherer (CE), "Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model and Acceptance for Referencing of Related Licensing Topical Reports," July 31, 1986. The FLOOD3 computer code is an enhanced version of the FLOOD-MOD2 methodology that is referenced in SRP 6.2.1.

O. The SGN-III computer code, used for the MSLB containment response blowdown calculation, was approved in NUREG-75/112, "Safety Evaluation

Report Related to the Preliminary Design of the Standard Reference System CESSAR System 80," December 31, 1975.

3. Approved Analyses

A. Radiological Consequences

All radiological consequence design basis analyses have been performed at 102% nominal power (or 2580.6 MWt). FSAR Table 14.1-6 provides the calculated dose for all events in FSAR chapter 14 with radiological consequences. In addition, the spent fuel pool gate drop, which is not described in the FSAR Chapter 14, has also been evaluated at 102% power with acceptable radiological consequences.

NMC remains in compliance with the Standard Review Plan (SRP) for Control Room Habitability and Maximum Hypothetical Accident (MHA) analyses. NMC continues to work with the NRC and industry as part of the Nuclear Energy Institute (NEI) 99-03 Task Force addressing acceptable methods for determining atmospheric dispersion factors.

B. Structural Behavior Following a LOCA

Approved as part of original design basis in the FSAR.

C. Natural Circulation Cooldown

The analysis was approved in a letter "Palisades Plant Response to NRC Generic Letter 81-21 – Natural Circulation Cooldown," dated December 15, 1983, Enclosure - Safety Evaluation Report.

D. Anticipated Transient Without Scram (ATWS)

The analysis was approved in a letter from A. W. De Agazio (NRC) to K. W. Berry (CPCo), "Safety Evaluation Related to Palisades Plant Compliance with 10 CFR Part 50.62," dated December 5, 1989.

E. Station Blackout

The analysis was approved in a letter from B. Holian (NRC) to G.B. Slade (CPCo), "Palisades Plant Station Blackout Analysis; Safety Evaluation (TAC No. 68578)," dated May 20, 1991.

F. Analyses to determine environmental qualification parameters

The environmental qualification program for Palisades Nuclear Plant was approved in a letter from Zwolinski (NRC) to VandeWalle (Consumers), "Environmental Qualification of Electrical Equipment Important to Safety," dated January 31, 1985. The accident and transient analyses input was developed at 102% of current rated thermal power.

G. Safe shutdown fire analysis

The safe shutdown fire analysis and Appendix R program was approved in a letter from Crutchfield (NRC) to VandeWalle (Consumers), "Fire Protection Rule – Alternate Safe Shutdown Capability – Sections III.G.3 and III.L of Appendix R to 10 CFR 50," dated May 26, 1983.

H. Spent fuel pool cooling

Spent fuel heat load is controlled by procedure. Irradiated fuel must decay for a specified period of time before it may be transferred to the spent fuel pool. The time specified takes into account the amount of decay heat contributed by fuel operated at 102% of current rated thermal power. Spent fuel pool heat load was approved in Wambach (NRC) to Berry (CPCo), "Amendment to Provisional Operating License No. DPR-20 (TAC 60844)," dated July 24, 1987.

I. Flooding

The flooding evaluation was approved in a letter from Wambach (NRC) to VandeWalle (Consumers), "Supplement to the Integrated Plant Safety Assessment Report (IPSAR) for the Palisades Plant," dated November 7, 1983.

## III. Accidents and transients for which the existing analyses of record do not bound plant operation at the proposed uprated power level

All accident and transient analyses of record bound plant operation at the proposed power level. This conclusion is predicated on the NRC acceptance of the following prior submittal, which affects the thermal margin/low pressure trip:

- Lahti (NMC) to NRC, "Nuclear Management Company, LLC – Palisades Nuclear Plant, Docket 50-255, License DPR-20 - License Amendment Request: Thermal Margin/Low Pressure Trip," October 17, 2002.

## IV. Mechanical/Structural/Material Component Integrity and Design

1. The actual increase in power to the proposed uprate level will be accomplished by additional opening of the turbine throttle valves. The resulting increase in steam flow will cause the temperature difference across the core to increase. However, at the proposed uprated power level, the primary coolant system (PCS) pressure, PCS average temperature, and PCS flow rate will be no different than they are at the current full power level. The principal plant parameters at the current and proposed uprated power levels are listed in the following table.

| Parameter | Pre-Uprate (2530 MWt) | Post-Uprate (2565.4 MWt) |
|---|---|---|
| Core power | 2530 MWt | 2565.4 MWt |
| PCS operating pressure | 2060 psia | 2060 psia |
| $T_{ave}$ range (0-100%) | 532 °F – 560 °F | 532 °F – 560 °F |
| $T_{cold}$ range (0-100%) | 532 °F – 537.3 °F | 532 °F – 537.0 °F |
| $T_{hot}$ range (0-100%) | 532 °F – 582.7 °F | 532 °F – 583.0 °F |
| SG steam pressure at full power | 770 psia | 765.8 psia |
| SG steam temperature at full power | 513.8 °F | 513.2 °F |
| SG steam flow at full power | 11,114,000 lbm/hr | 11,297,000 lbm/hr |
| SG feed temperature at full power | 439.5 °F | 440.7 °F |
| SG feed flow at full power | 11,174,000 lbm/hr | 11,357,000 lbm/hr |

The nuclear steam supply system (NSSS) components, including the reactor vessel and core support structures, were designed to operate at a core power level of 2650 MWt, which bounds the proposed uprate power level.

A. Components

i. Reactor Vessel, Nozzles, and Supports

The code of record for the reactor vessel, including the vessel nozzles and supports, is the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Class A, 1965 edition, including all addenda through Winter 1965. The design and operating temperatures and pressures used in the analyses of record continue to bound the conditions expected at the proposed uprated power level.

ii. Core Support Structures and Vessel Internals

The core support structures and vessel internals were designed prior to the introduction of specific criteria for these components in ASME Boiler and Pressure Vessel Code, Section III. However, as stated in the FSAR, the vessel internals were designed in accordance with the 1965 Section III criteria where required. The power level and the temperatures and pressures used in the design continue to bound the conditions expected at the proposed uprated power level.

iii. Control Rod Drive Mechanisms

The code of record for the pressure containing members of the control rod drive mechanisms is the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1989. The temperatures and pressures used in the design continue to bound the conditions expected at the proposed uprated power level.

iv. NSSS Piping, Pipe Supports, Branch Nozzles

The code of record for the primary coolant system piping is the Code for Pressure Piping, ASA B31.1, 1955. All stresses meet the appropriate allowables for both this code and the ASME Boiler and Pressure Vessel Code, Section III, Class A, 1965 edition. In addition, all piping nozzles have sufficient reinforcement to meet the requirements of both codes, and all nozzle configurations and relative spacings meet the Section III requirements. The temperatures and pressures used in the design and in the analyses of record continue to bound the conditions expected at the proposed uprated power level.

v. Balance of Plant (BOP) Piping

The code used in the original design of balance of plant piping, including the main steam, condensate, feedwater, auxiliary feedwater, and steam generator blowdown systems, is either the Code for Pressure Piping, ASA B31.1, 1955 edition, or the Power Piping Code, USAS B31.1, 1967 edition. All safety-related piping, including safety-related balance of plant piping, has been re-analyzed to ANSI B31.1, 1973 edition, through the Summer 1973 addenda.

The temperatures and pressures used in the design and in the analyses of record for the balance of plant piping continue to bound the conditions expected at the proposed uprated power level. The proposed power uprate will result in a slight increase of the normal operating temperature of the feedwater piping. System design temperature of 450°F bounds the expected normal operating temperature and the impact of this condition has been evaluated and found acceptable.

vi. Steam Generator Tubes, Secondary Side Internal Support Structures, Shell, and Nozzles

The code of record for the steam generator tubes, secondary side internal support structures, shell, and nozzles is the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1977 edition.

The primary coolant system temperatures and pressures used in the design and in the analyses of record continue to bound the conditions expected at the proposed uprated power level.

The steam and feedwater pressures and flow rates used in the design and analyses of record continue to bound the conditions expected at the proposed uprated power level. The difference between the primary coolant system operating pressure and the slightly reduced steam pressure at the proposed uprated conditions remains less than the "normal plant variation" pressure differential specified for the fatigue analysis of the steam generator. The maximum steam flow rate at the proposed uprated conditions will continue to be bounded by the design flow rate given in the table in section IV.1.

Flow Induced Vibration

As indicated in the table in section IV.1, the feedwater and steam flows at the proposed uprated conditions continue to be bounded by those used in the design of the steam generators. The design phase included significant flow induced vibration modeling to ensure stability of the tube bundle. Therefore, the proposed power uprate will have no effect on the magnitude or likelihood of flow induced vibration.

Steam Generator Tube Integrity

Steam generator tube integrity is monitored and maintained in accordance with the plant Technical Specifications and with the guidance provided in NEI 97-06, "Steam Generator Program Guidelines," Revision 1. Six active damage mechanisms have been identified:

- structural wear
- axial outside diameter stress corrosion cracking (ODSCC) at the hot leg top of the tubesheet
- circumferential ODSCC at the hot leg top of the tubesheet
- axial primary water stress corrosion cracking (PWSCC) in the u-bend
- axial PWSCC in the tubesheet in non-expanded tubes
- axial PWSCC within expanded tubesheet region

The main steam generator tube degradation mechanism to date at Palisades is mechanical wear at the following support structures: the vertical straps, the diagonal bars, and the eggcrate lattice supports. The increase in feedwater flow can be expected to result in some additional mechanical wear at these locations. However, the existing steam generator program ensures that these areas are tested and monitored each refueling outage. Predicted steam generator tube wear rates conservatively bound the actual wear rates found during subsequent tube inspections. In addition, recent operating experience involving higher than normal (by approximately 3.5%) steam flow from one steam generator prompted the performance of a Monte Carlo simulation in order to estimate the effects on tube integrity. The predicted wear at this flow rate, which bounds the flow rate expected at the uprated power level, was, when compared with the actual results from the subsequent tube inspection, found to be conservative. The steam generator program will continue to adequately monitor and maintain steam generator tube integrity under the proposed uprate conditions.

Stress corrosion cracking has the potential to be affected by the slight increase in $T_{hot}$. The greatest vulnerability is ODSCC at the top of the tubesheet. A curve developed by the Electric Power Research Institute (EPRI), in connection with the 1999 refueling outage, predicts that the onset of this damage mechanism will not occur until after the end of the license. Since the $T_{hot}$ value used in constructing the curve is the same $T_{hot}$ value expected at proposed uprated conditions, the proposed uprate would not change this conclusion.

Should it become necessary to plug any steam generator tubes, steam generator program procedures require that the issue is documented in the plant's corrective action program and the identified condition is evaluated. In this way, any effects on the plant that result from the plugging of steam generator tubes are assessed and, if necessary, appropriate corrective action is undertaken. In addition, the plant Technical Specifications require verification of the primary coolant system flow rate after each plugging of ten or more steam generator tubes.

The issue of steam generator tube high cycle fatigue is discussed in section IV.1.F, below.

vii. Reactor Coolant Pumps

The primary coolant pumps were designed prior to the introduction of specific criteria for pumps in ASME Boiler and Pressure Vessel Code, Section III. The pressure retaining parts of the pumps were designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class A, 1968 edition. The temperatures and pressures used in the design continue to bound the conditions expected at the proposed uprated power level.

viii. Pressurizer Shell, Nozzles, and Surge Line

The code of record for the pressurizer, including the vessel nozzles, is the ASME Boiler and Pressure Vessel Code, Section III, Class A, 1965 edition, including all addenda through Winter 1966. The code of record for the pressurizer surge line is analyzed to the design requirements of ANSI B31.1, 1973 edition, including the Summer 1973 addenda. The design and operating temperatures and pressures used in the analyses of record continue to bound the conditions, including the slightly lower temperature of the (cold leg) water entering the spray nozzle and the slightly higher temperature of the (hot leg) water entering the surge nozzle, that are expected at the proposed uprated power level.

ix. Safety Relief Valves

The code of record for the pressurizer safety valves is the ASME Boiler and Pressure Vessel Code, Section III, Class A, 1965 edition, including all addenda through Winter 1965. The power level and the temperatures and pressures used in the design continue to bound the conditions expected at the proposed uprated power level. In addition, as discussed in section II, above, the analysis of record for the design basis event used in assessing the capacity of the pressurizer safety valves, the loss of external load, assumes the reactor is initially operating at 2580.6 MWt. 2580.6 MWt amounts to 102% of current rated thermal power.

Since the design temperatures and pressures of the primary coolant system will continue to bound the operating conditions expected at the proposed uprated power level, other safety-related valves in the primary coolant system are not affected by the proposed uprate.

The pressure and temperature ratings of the main steam safety valves will continue to bound the operating conditions expected at the proposed uprated power level. The valves have design relief capability suitable for NSSS operation at 2650 MWt. In addition, as discussed in section II, above, the analysis of record for the design basis event used in assessing the capacity of the main steam safety valves, the loss of external load, assumes the reactor is initially operating at 102% of current rated thermal power.

Therefore, the pressurizer and main steam safety valves are not affected by the proposed uprate.

B. Aspects of Component Design Potentially Affected by the Uprate

i. Stresses

The stresses considered in the design of the components result from combinations of the following: internal pressure, thermal transients, external pipe loads, fluid flow, dead weight, seismic, and bolt preload.

The proposed uprate involves no change in the operating pressure of the primary coolant system. As a result, there will be no change in any primary coolant system internal pressure used in component design. There will be a slight decrease in main steam and feedwater operating pressure at the proposed uprated conditions. These conditions are bounded by system design pressure.

The proposed uprate involves no change in the nature of any thermal transient. At the proposed uprated conditions, there are slight changes in primary coolant system temperatures. However, as indicated in the table in section IV.1, these changes are within the limits of the temperatures originally specified for component design.

At the proposed uprated conditions, the increase in power level and the slight changes in primary coolant system temperatures will continue to be bounded by the power level and temperatures originally specified for component and piping design. The impact of the slight increase in the temperature of the main feedwater piping that is expected at the proposed uprated power level has been evaluated and is bounded by the system design temperature of 450°F. Therefore, there is no change in thermal movements, or safety/relief valve discharges, which could increase the external pipe loads.

The proposed uprate involves no change in the flow rate of the primary coolant. The expected change in core inlet and core exit temperatures will have a negligible effect on the primary coolant density. Furthermore, these changes are within the temperature range for which the equipment was originally designed. Therefore, there are no changes related to analyzed flow for any of the components in the primary coolant flowpath. Operation at the proposed uprated power level will result in an increase in the steam flow and feedwater flow through the steam generators. The feedwater and steam flow at the proposed uprated conditions will continue to be bounded by those used in the design of the steam generators.

The proposed uprate involves no change in loads due to dead weight, seismic, or bolt preload.

ii. Cumulative Usage Factors

The FSAR states that the following design cyclic transients were used in the fatigue analysis required by code: heatup/cooldown, ramp changes in power, step changes in power, hydrostatic testing, leakage testing, normal operating pressure variations, reactor trips, loss of turbine load, and loss of primary coolant flow. The fatigue analyses could be affected if there were (1) a change in the expected number of transients or (2) changes in the system pressures or temperatures or in their rates of change.

The proposed power uprate involves no change in the manner in which the plant is operated and no increase in the likelihood of any plant transient. Thus, the design number of transients continues to bound the expected number of transients.

The proposed power uprate involves no change in the operating pressures of the primary coolant system. The proposed power uprate involves no change in heatup or cooldown rates. At the proposed uprated conditions, there is a slight increase in the core exit and primary coolant hot leg temperatures and a slight decrease in the core inlet and primary coolant system cold leg temperatures. However, as indicated in the table in section IV.1, these changes are within the limits of the temperatures used in the original stress and fatigue analyses.

Therefore, the cumulative usage factors are not affected by the proposed power uprate.

iii. Flow Induced Vibration

The proposed power uprate involves no change in the flow rate of the primary coolant. The expected change in core inlet and core exit temperatures will have a negligible effect on the primary coolant density. Furthermore, these changes are within the temperature range for which the equipment was originally designed. Therefore, there are no changes related to analyzed flow induced vibration for any of the components in the primary coolant flowpath.

Operation at the proposed uprated power level will result in an increase in the steam flow and feedwater flow through the steam generators. A discussion of the susceptibility of the steam generators to flow induced vibration is included in item IV.1.A.vi, above.

iv. Changes in Temperature

The proposed power uprate results in small changes in temperature on both the primary and secondary sides of the plant. The pre-uprate and post-uprate temperatures are presented in the table in section IV.1 and will continue to be bounded by those used in the original design.

v. Changes in Pressure

The proposed power uprate results in small changes in pressure on the secondary side of the plant. The main steam pre-uprate and post-uprate pressures are presented in the table in section IV.1 and will continue to be bounded by the pressure used in the original design.

vi. Changes in Flow Rates

The proposed power uprate results in small changes in flow rates on the secondary side of the plant. The main steam and main feedwater pre-uprate and post-uprate flow rates are presented in the table in section IV.1 and will continue to be bounded by the flow rates used in the original design.

vii. High-Energy Line Break Locations

The Palisades systems requiring evaluation for high energy line breaks are:

- Primary Coolant
- Engineered Safeguards/Safety Injection
- Chemical and Volume Control
- Main Steam
- Main Feedwater
- Auxiliary Feedwater
- Steam Generator Blowdown
- Sampling
- Turbine Extraction
- Heating Steam and Condensate

With the exception of slight increases in operating temperatures (< 1°F) of the primary coolant ($T_{hot}$), main feedwater, condensate, sampling ($T_{hot}$), and turbine extraction systems; the proposed power uprate has no impact on the above systems. The operating pressures of all systems remain bounded by system design pressures.

Outside Containment

The methodology for determining high energy break locations outside containment was reviewed and approved by NRC letter Wambach to VandeWalle, "Palisades Plant - SEP Topic III-5.B Pipe Break Outside Containment," dated February 19, 1982 in connection with their review of the effects of a piping system break outside containment.

Break locations are postulated at terminal ends, branch connections, at the two intermediate locations of highest stress, and at points where the stresses exceed certain allowable values. Critical crack failures of seismic category I piping are also postulated at those locations where the failure could adversely affect essential structures and components.

The design basis for a high energy line break outside of containment is unchanged by the proposed power uprate.

Inside Containment

The locations of postulated breaks inside containment are based on a combination of mechanistic and effects oriented approaches. With the exception of a portion of the 42-inch hot leg and 30-inch cold leg primary coolant pipes, which were analyzed using the mechanistic approach, all other high energy pipe systems were analyzed using the effects oriented approach. Since neither the potential targets nor the structural discontinuities in the piping are affected by a change in the operating parameters of the system, the slight changes in operating temperatures at the proposed uprated power conditions have no effect on the high energy break locations.

viii. Jet Impingement and Thrust Forces

The magnitude of the jet thrust from a failed pipe is a function of the system operating pressure and the cross-sectional area of the pipe. The proposed power uprate involves no change in the cross-sectional area of any pipes. The only appreciable increase in the operating pressure for the high energy systems is in the turbine extraction steam lines. The pressure increase is predicted to be approximately 1.2% of the operating pressure, which is bounded by the piping system's design pressure. Since a break in this system could not have any adverse impact on essential structures or components, the consequences of pipe whip are not investigated. Thus, the proposed power uprate has no effect on the thrust forces resulting from a high energy pipe break.

The magnitude of the jet impingement force from a failed pipe is a function of the jet pressure, angle of the jet axis, and the target cross-sectional area. Again, the only high energy system that will experience an appreciable increase in pressure is the turbine extraction steam system, and the consequences of jet impingement are not investigated for this system. Thus, the proposed power uprate has no effect on the jet impingement forces resulting from a high energy pipe break.

C. Integrity of the Reactor Vessel

i. Pressurized Thermal Shock Calculations

The pressurized thermal shock calculations are those used in determining the pressurized thermal shock reference temperature for each of the reactor vessel beltline materials. The pressurized thermal shock reference temperature for each reactor vessel beltline material, at the end of license fluence, was evaluated and determined to be below the 10 CFR 50.61 screening criteria. This evaluation was reviewed and approved in NRC letter Hood to Haskell, "Palisades Plant – Reactor Vessel Neutron Fluence Evaluation and Revised Schedule for Reaching Pressurized Thermal Shock Screening Criteria (TAC No. MA8250)," dated November 14, 2000.

The pressurized thermal shock reference temperature for a particular material is a function of the following: the reference temperature in the unirradiated condition, the margin (which is, in turn, a function of the unirradiated reference temperature), the chemistry factor (which is, in turn, a function of the copper and nickel content), and the end of license fluence.

The only one of these parameters, which has the potential to be affected by the proposed power uprate, is the end of license fluence. As is discussed in item ii, immediately below, the end of license fluence previously reviewed and approved by the NRC will continue to bound the proposed power uprate conditions. Therefore, pressurized thermal shock calculations are not affected by the proposed power uprate.

ii. Fluence Evaluation

The current Palisades fluence evaluation was reviewed and approved by NRC letter Hood to Haskell, " Palisades Plant – Reactor Vessel Neutron Fluence Evaluation and Revised Schedule for Reaching Pressurized Thermal Shock Screening Criteria (TAC No. MA8250)," dated November 14, 2000. Included in the evaluation were end of license fluence values for the various reactor vessel beltline materials.

The determination of the fluence at end of license was based, in part, on a specified capacity factor for the remaining fuel cycles. In this way, a total number of effective full power days until the end of license was estimated and then incorporated into the fluence evaluation. If the neutron flux is assumed to scale with reactor power, the proposed 1.4% uprate can be considered equivalent to adding an additional 1.4% to the total number of effective full power days.

However, because of the extensive maintenance outage in 2001, the actual number of effective full power days until the end of license will be less than that originally predicted. The current prediction of effective full power days at 2565.4 MWt until the end of the current operating license is 3545.7 days is presented in the following table. This prediction is bounded by the NRC approved value of 3656.3 days. This difference more than offsets the effects of the proposed power uprate so that the fluence values previously reported will continue to be bounding.

The following table lists the reactor operating cycle lengths used when estimating the NRC approved reactor vessel fluence estimate, the most recent plan for operating cycle lengths, and the plan estimate increased for operating at the proposed rated thermal power beginning April 20, 2003.

| Cycle | EFPD (1/24/00) Assumed in NRC approved analysis | EFPD (4/20/03) Actual and planned | EFPD + 1.4% (4/20/03) Actual and planned at proposed uprated power level |
|---|---|---|---|
| 15 | 444.3 | 401.3 | 401.3 |
| 16 | 488.0 | 444.3 | 444.3 |
| 17 | 488.0 | 495.6 | 502.5 |
| 18 | 488.0 | 505.7 | 512.8 |
| 19 | 488.0 | 488.5 | 495.3 |
| 20 | 488.0 | 505.7 | 512.8 |
| 21 | 488.0 | 512.5 | 519.7 |
| 22 | 284.0 | 154.8 | 157.0 |
| Total | 3656.3 | 3508.4 | 3545.7 |

The fluence evaluation is not affected by the proposed power uprate.

iii. Heatup and Cooldown Pressure-Temperature Limit Curves

The current heatup and cooldown curves were reviewed and approved by NRC letter Gamberoni to Haas, "Palisades Plant – Issuance of Amendment Re: Pressure - Temperature Limits (TAC No M90650)," dated March 2, 1995 as part of Amendment No. 163 to the facility operating license.

As explained in the response to item ii, the end of license fluence with the proposed power uprate continues to be bounded by the value established in the current fluence evaluation. This latter value is, in turn, bounded by the fluence value used (2.192E19 n/cm$^2$ at the limiting beltline material) in determining the heatup and cooldown pressure-temperature limit curves. Therefore, these curves are not affected by the proposed power uprate.

iv. Low-Temperature Overpressure Protection

The current low-temperature overpressure protection was reviewed and approved by NRC letter Gamberoni to Haas, "Palisades Plant – Issuance of Amendment Re: Pressure - Temperature Limits (TAC No M90650)," dated March 2, 1995 as part of Amendment No. 163 to the facility operating license.

The low-temperature overpressure protection is intended to protect against the pressure and temperature limits specified on the heatup and cooldown pressure-temperature limit curves. As explained in the response to item iii, these heatup and cooldown curves continue to bound operation at the proposed uprated conditions. Furthermore, the proposed power uprate does not involve operation at the low temperature conditions during which low-temperature overpressure protection is a concern. Therefore, low-temperature overpressure protection is not affected by the proposed power uprate.

v. Upper-Shelf Energy

10 CFR 50, Appendix G requires that reactor vessel beltline materials maintain a Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lbs unless it is demonstrated, in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy upper-shelf energy will provide adequate margins of safety.

The determination of the upper-shelf energy for a material is a function of (1) the copper content of that particular beltline material and (2) the end of license fluence for the material. The proposed power uprate has no effect on the copper content of any reactor vessel beltline material. The end of license fluence values used in the upper-shelf energy determinations continue to bound the currently approved end of license fluence values (which are discussed in item ii, above). Therefore, the values of upper-shelf energy for the reactor vessel beltline materials are not affected by the proposed power uprate.

vi. Surveillance Capsule Withdrawal Schedule

The end of license fluence values, used in determining the surveillance capsule withdrawal schedule, continue to bound the currently approved end of license fluence values (which are discussed in item ii, above). Therefore, the surveillance capsule withdrawal schedule is not affected by the proposed power uprate.

D. Code of Record

The code of record for particular components is included in section IV.A, above. Information on the code of record for the principal components of the primary coolant system was presented in the original FSAR, which was submitted as Amendment 9 to the license application. Amendment 11 to the application contained additional information concerning the specific edition of the code that was used.

The original safety evaluation, "Safety Evaluation by the Division of Reactor Licensing U.S. Atomic Energy Commission in the Matter of Consumers Power Company Palisades Plant Docket No. 50-255," dated March 6, 1970, section 1.0, was based on Amendments 9 through 20. This evaluation noted that the reactor vessel "was designed and constructed in accordance with Section III, Class A of the ASME Code." Although the safety evaluation did not explicitly address the other components, it is clear that the submitted information concerning the code of record formed part of the basis for approval of the provisional operating license.

E. Component Inspection and Testing Programs

The following ongoing primary coolant system component inspection and testing programs are identified in either the FSAR or Technical Specifications (TS):

- Reactor Pressure Vessel Material Surveillance Program (FSAR 4.5.3)
- Inservice Inspection (FSAR 4.5.6, FSAR 6.9)
- Primary Coolant Pump Flywheel Integrity (TS 5.5.6)
- Steam Generator Tube Integrity (TS 5.5.8)

In addition, the Palisades engineering programs were reviewed to assess the impact of the proposed uprate on the various plant programs. Those programs that have the potential to be affected by the uprate are discussed in the following paragraphs.

i. Reactor Pressure Vessel Material Surveillance Program

This surveillance program is unaffected by the proposed power uprate. Specific reactor pressure vessel issues are discussed in section IV.1.C.

ii. Inservice Inspection

The systems and components subject to the inservice inspection plan are selected in accordance with 10 CFR 50.55a and Regulatory Guide 1.26. System leakage tests, system functional tests, system inservice tests, and system hydrostatic tests are performed in accordance with the latest edition of ASME Section XI approved for use at Palisades and with approved relief requests and requests for code case use. The proposed power uprate has no effect on the selection of systems or components or on the conduct of the inservice tests or on the acceptance criteria. Therefore, the inservice inspection program is not affected by the proposed power uprate.

iii. Primary Coolant Pump Flywheel Integrity

The proposed power uprate involves no change to any of the parameters, which could impact the operation of the primary coolant pumps, except for a slight decrease in the temperature of the primary coolant leaving the steam generator at full power. However, this temperature is bounded by the lower temperatures that occur during operation at reduced power levels. Therefore, primary coolant pump flywheel integrity is not affected by the proposed power uprate.

iv. Steam Generator Tube Integrity

Steam generator tube integrity is discussed in section IV.1.A.vi, above.

v. Motor Operated, Air Operated, and Relief Valve Programs

The slight changes in operating parameters resulting from the proposed uprate have no impact on system design or maximum differential pressures. The valve programs are not affected by the proposed power uprate.

vi. Flow Accelerated Corrosion

The proposed power uprate involves slight increases in the flow rates of the main steam, main feedwater, condensate, heater drain, and extraction steam systems. The piping in these systems is within the scope of the current flow accelerated corrosion program, which uses CHECWORKS software for the prediction of corrosion wear rates and for data analysis. The effects of the increased flow rates associated with the proposed power uprate were assessed using this model. The results indicate marginal changes to the remaining life of the feedwater system. Actual flow rates are incorporated into the flow accelerated corrosion program and the examination schedules of the piping and components are adjusted by the CHECWORKS predicted remaining life.

vii. Heat Exchanger Condition Assessment

The proposed power uprate involves slight increases in the flow rates of the main steam, main feedwater, condensate, heater drain, and extraction steam systems. The heat exchangers in these systems are within the scope of the current heat exchanger condition assessment program. Inspections of these balance of plant heat exchangers are performed during refueling outages. The frequency of the inspections (typically, the time scale is three to ten years) is such that the slight increases in flow rates resulting from the proposed power uprate will have no effect on the timing of the inspection for any of these components. Heat exchanger tube plugging criteria are developed from a set of ten degradation factors, each weighted in accordance with its importance. The increase in flow rate resulting from the proposed 1.4% uprate has the potential to only slightly affect one of the factors (flaw growth rate), thus, the overall effect of the proposed power uprate on tube plugging criteria is insignificant.

F. Steam Generator Tube High-Cycle Fatigue

NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," concerned the susceptibility of Westinghouse steam generators with carbon steel support plates to a rapidly propagating fatigue failure. The failure was thought to result from a combination of a mean stress level in the tube (caused by denting) and a superimposed alternating stress (caused by flow-induced vibration). In the bulletin, the NRC concluded that the presence of the following three factors could lead to a rapidly propagating fatigue failure:

- denting at the upper support plate
- a high fluid-elastic stability ratio
- absence of effective antivibration bar support

NRC Bulletin 88-02 does not directly apply to Palisades, because the Palisades steam generators were not manufactured by Westinghouse, nor do the steam generators use carbon steel support plates.

The Palisades steam generators have stainless steel eggcrate lattice tube supports. Thus, denting at the upper support plate, the condition of concern in NRC Bulletin 88-02, is not expected to occur at Palisades. The latest steam generator degradation assessment, for the 2003 refueling outage, states that this mechanism is considered nonrelevent. Vertical straps and diagonal bars act as antivibration dampers and these, together with the eggcrate design, provide effective antivibration support.

The design of the upper bundle supports was analyzed for the type of failure reported in NRC Bulletin 88-02. It was determined that the design differences provide greater in-plane and out-of-plane tube support and that the stability ratio of the tubes is less than unity for this condition regardless of the boundary condition at the top of the tubesheet.

A recent industry event involving high cycle fatigue in once through steam generators has been reported. The susceptibility of the Palisades steam generators to this failure, fatigue caused by flow-induced vibration of a swollen and restrained tube that had been previously plugged, has been evaluated as part of the plant's operating experience program. Palisades' eggcrate lattice tube supports would produce a different damping effect if a tube were to swell. NMC is actively engaged in owners group and industry activities associated with this issue.

As indicated in the table in section IV.1, the flow rate used in the design of the steam generators will continue to bound the flow rate at the proposed uprated power level. Therefore, the low probability that the steam generator tubes will experience high cycle fatigue is not affected by the proposed power uprate.

## V. Electrical Equipment Design

1. The performance of safety related electrical equipment is not affected by the proposed power uprate. All protective breaker settings and overload limits remain unchanged.

   A. The transient or accident that is the subject of concern is a design basis accident (DBA), which is a LOCA with loss of offsite power. The analysis determines the adequacy of the emergency diesel generators (EDG) in starting, accelerating, and providing steady state power to DBA motor loads. Per the FSAR, the design basis for the emergency diesel generators is to provide a dependable onsite power source capable of starting and supplying the essential loads to safely shut down the plant and maintain it in a safe shutdown condition under all conditions. The reliability of this onsite power is provided by its duplication wherein each emergency generator supplies redundant loads and each is capable of providing power to the minimum necessary safeguards equipment.

   An increase in thermal power prior to an accident will have no effect on starting and accelerating the EDG DBA motor loads. The EDG DBA steady state load conditions are either not affected or are bounded by the safety system hydraulic conditions used in the emergency diesel generators steady state load analysis. Therefore, the proposed uprate in power level continues to be bounded by the existing analyses of record for the plant.

Bounding event determinations continue to be valid. Hydraulic pumping power requirements used as inputs into the EDG steady state load analysis are conservative with respect to hydraulic conditions used in the transient analyses. Since there is no change in the hydraulic power requirement to the EDG analysis, the bounding event determination of a LOCA with loss of offsite power continues to be valid.

B. Station blackout equipment is unaffected by the proposed power uprate. The station blackout assessment remains valid, as identified in Section II.3.E above.

C. Environmental qualification of electrical equipment was approved in a letter from Zwolinski (NRC) to VandeWalle (Consumers), "Environmental Qualification of Electrical Equipment Important to Safety," dated January 31, 1985 and remains valid. The design basis accident temperature and radiation profile remains unchanged from the approved conditions due to the 1.4% power uprate. Therefore, electrical equipment subject to environmental qualification requirements will not be impacted. The approved environmental qualification program adequately monitors and maintains the applicable equipment.

D. Grid stability is not compromised by operation at the proposed uprated power level.

The System Planning and Protection Department of the Michigan Electric Transmission Company performed an offsite power supply reliability analysis specifically for the Palisades Plant. The results are documented in report "Offsite Power Supply Reliability Analysis," dated October 18, 2001. The following conclusions were reached regarding stability of the transmission system:

- The Palisades Nuclear Plant and the offsite power system connected to Palisades Substation are stable for a three phase-to-phase ground fault, anywhere in the system, which will be cleared by primary relays with the most critical element out of service before the disturbance.
- The Palisades Nuclear Plant and the offsite power system connected to the Palisades Substation are stable for a two phase-to-phase ground fault with subsequent breaker failure, anywhere in the system, with all transmission in service before the fault.
- The Palisades Nuclear Plant and the offsite power system connected to the Palisades Substation are stable for inadvertent tripping of three Ludington units in the pumping mode, representing a 1020 MW of sudden load drop, or for sudden loss of 1000 MW of area load.

- The offsite power system connected to Palisades Substation is stable for inadvertent tripping of the Palisades or Covert Plant units.

## VI. System Design

1. In order to assess the effect of the proposed power uprate on the major plant systems, the conditions expected after the proposed power uprate were compared with both the design and the current operating conditions for the system or component. The NSSS post-uprate conditions are those described in the discussion of the NSSS, above. Balance of plant conditions were estimated by increasing steam flow and decreasing steam generator pressure in the current plant heat balance.

### A. NSSS Interface Systems

Main Steam

The 1.4 % proposed power uprate will result in a slight decrease in the value of main steam system pressure and temperature at full power. There will be an increase in the full power main steam flow rate.

At the proposed uprated power level, the steam generator operating pressure and temperature will be slightly lower than the current full power steam generator operating pressure. However, there is no effect on the pressure or temperature rating of the valves or piping of the main steam system, which will continue to be bounded in all modes of operation. The setpoints of the main steam safety valves will continue to be based on steam generator and main steam piping design conditions.

*Main Steam Safety Valves (MSSVs)*

The results of the current heat balance indicate that the expected steam flow, after the 1.4% proposed power uprate, will be less than the capacity of the safety valves. The ASME Boiler and Pressure Vessel Code rating of the relief valves presently installed in the plant is 511,563 lbm/hr per valve at 1000 psia, when the first bank of valves opens, and 532,041 lbm/hr per valve at 1040 psia, when the last of the three banks open. The Technical Specifications require 23 of the 24 relief valves to be operable. Therefore, the Technical Specifications required relief capacity amounts to 11,765,949 lbm/hr flow at the secondary side design pressure, which is substantially more than the estimated steam flow of 11,297,000 lbm/hr following the proposed 1.4 % power uprate.

Therefore, the main steam safety valves are not affected by the proposed power uprate.

*Atmospheric Dump Valves*

The function of the atmospheric dump valves is to limit MSSV operation during a turbine runback and to provide a means of decay heat removal during various plant transients. These valves relieve 30% of the full power steam flow of 11,297,000 lbm/hr or 3,389,100 lbm/hr at the 1.4% proposed power uprate heat balance conditions. The combined rated capacity of the atmospheric dump valves remains at 30% of steam flow with reactor at full power, as stated in the FSAR.

*Turbine Bypass Valve*

The turbine bypass valve relieves 4.5% of the full power steam flow of 11,297,000 lbm/hr or 508,365 lbm/hr. The results of the current heat balance indicate that the 1.4% proposed power uprate will increase the full power steam flow, however, the rated capacity of this valve (528,000 lbm/hr) will remain above 4.5% of steam flow with reactor at full power, as stated in the FSAR.

*Main Steam Isolation Valves (MSIVs)*

The MSIVs were previously analyzed for a steam flow corresponding to a main-steam-line-break flow of 19,000,000 lbm/hr which is significantly higher than the steam flow expected at the proposed uprated power level, thus, the MSIVs will continue to be capable of performing their previously defined safety function.

Therefore, the main steam isolation valves are not affected by the proposed power uprate.

*Steam Generator Blowdown*

Steam generator blowdown is manually controlled by having an operator throttle the appropriate valves to achieve a specified flow rate, which is procedurally controlled to remain within analyzed conditions. At the proposed uprated power level, the steam generator operating pressure will be slightly lower than the current full power steam generator operating pressure. This means the operator may have to simply open the valves slightly more than at present to achieve a specified flow rate, but will not impact analyzed requirements.

Therefore, steam generator blowdown is not affected by the proposed power uprate.

Condensate and Feedwater

The 1.4% proposed power uprate will result in slight changes, typically, increases, in the extraction steam, condensate, and feedwater temperatures at full power. There will be an increase in the extraction steam, condensate, and feedwater flow rates.

*Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves*

The main feedwater regulating valve actuator must be capable of operating against the maximum differential pressure. Although the proposed power uprated feedwater flow will be slightly higher than the current full power feedwater flow, the system conditions assumed in the evaluation of the actuator will continue to bound the valve operating conditions. The valve actuator design basis remains unchanged, thereby ensuring the ability of the valve actuator to perform its safety function of isolating a guillotine break of a main feedwater line in containment.

The main feedwater regulating valve bypass valves are closed during power ascension and are not affected by the proposed power uprate.

The pressure and temperature ratings of the main feedwater regulating and bypass valves will continue to bound all modes of operation.

Therefore, the main feedwater regulating valves and bypass valves are not affected by the proposed power uprate.

*Main Feedwater Pumps*

The expected feedwater flow increases from 11,174,000 lbm/hr to 11,357,000 lbm/hr for the proposed power uprate. Operating experience has shown that the feedwater pump low suction pressure alarm does not actuate when there is a minor increase in flow (of the magnitude expected at the proposed uprated power level), but only when there is a significant system perturbation. Thus, the feedwater pump suction pressure will remain within acceptable limits following the proposed power uprate.

Therefore, the main feedwater pumps are not affected by the power proposed power uprate.

Auxiliary Feedwater

*Auxiliary Feedwater Pumps*

The design basis auxiliary feedwater flow is established by the loss of normal feedwater event. As indicated in section II of this evaluation, the analysis of record for the loss of normal feedwater event was performed at 102% current rated thermal power, or 2580.6 MWt. Therefore, the auxiliary feedwater pumps are not affected by the proposed power uprate.

*Condensate Storage Tank*

Auxiliary feedwater is drawn from the condensate storage and primary makeup storage tanks. As described in the TS Bases, the specified storage tank volume is sufficient to remove decay heat for eight hours following a reactor trip from 102% current rated thermal power. This amount of time allows for a cooldown of the primary coolant system to shutdown cooling entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. The condensate storage requirements for a station blackout were also determined on the basis of an initial power level of 102% of current rated thermal power.

Therefore, the condensate storage and primary makeup storage tanks are not affected by the proposed power uprate.

B. Containment systems

The containment must be capable of withstanding the pressures and temperatures that result from design basis accidents without exceeding the design leakage rate. The analyses of record for the loss-of-coolant accident and the main steam line break demonstrate that the containment is capable of performing this function. As indicated in section II, above, these analyses have been run at a power level of 102% of current rated thermal power, thereby ensuring that the containment will continue to perform this function at the proposed uprated power level.

The safety function of the containment cooling systems, the containment air coolers and containment spray, is to limit the containment pressure and temperature to acceptable values following an accident. The containment air coolers are also designed to maintain the containment temperature and pressure within acceptable limits during normal operation.

The analyses of record for the loss of coolant accident and the main steam line break demonstrate that the containment cooling systems are capable of performing their safety function. As indicated in section II, above, these analyses have been run at a power level of 102% of current rated thermal power, thereby ensuring that the containment cooling systems will continue to perform their safety function at the proposed uprated power level.

The 1.4% proposed power uprate will have an insignificant effect on the amount of heat that must be removed by the containment air coolers during normal operation. The loads on the equipment located inside containment that produces heat, such as motors or transformers, will not change. There will be a negligible change in the losses from the primary coolant system piping since the full power hot leg temperature will increase slightly while the cold leg temperature will decrease slightly. Likewise, the full power feedwater piping temperature will increase slightly while the main steam temperature will decrease slightly. These changes are insignificant and expected to be within the existing capacity of the coolers and the associated service water system. In addition, containment temperatures are continuously displayed in the main control room.

Therefore, the containment systems are not affected by the proposed power uprate.

C. Safety-related cooling water systems

*Component Cooling*

The component cooling water system provides a heat sink for the removal of process and operating heat from safety related components following a design basis event. The accident analyses for the loss-of-coolant accident demonstrate that the system is capable of performing this function. As indicated in section II, above, these analyses have been run at a power level of 102% of current rated thermal power, thereby ensuring that the component cooling water system will continue to perform this function at the proposed uprated power level.

The system also provides a heat sink for the removal of process and operating heat from various non-safety related components during normal operation.

Operation at the proposed uprated power level will result in a 1.4% increase in decay heat. However, the initiation of shutdown cooling will still be controlled by the limits on primary coolant system temperature and pressure. There will be no change in the temperatures, pressures, or flow rates experienced by the low pressure safety injection system during plant shutdown and, therefore, no change in the temperatures, pressures, or flow rates experienced by the component cooling water system during plant shutdown, other than a slight increase in the actual cooldown time.

The non-safety related shield cooling system is cooled by the component cooling system. This system is designed to remove heat from the biological shield surrounding the reactor vessel. The FSAR states that the system is capable of removing 180,000 BTU/hr. The FSAR further states that during normal plant operation, 120,000 BTU/hr of heat is transferred to the biological shield. Only 35,000 BTU/hr of this is due to heating from radiation interactions within the shield, the remainder is due to convective and radiative heat losses from the reactor. The convective and radiative heat loads will not change significantly, because $T_{ave}$ is unchanged. Since the change in reactor power is small, the increase in neutron and gamma heating of the biological shield is expected to be small, leaving appreciable margin.

There will be no change in the operating conditions for most of the other equipment serviced by component cooling water. Although the proposed power uprate will result in a slight reduction in primary coolant cold leg temperature at full power (which could lower the demands on the letdown heat exchanger), the effect on component cooling is considered insignificant.

Therefore, the component cooling water system is not affected by the proposed power uprate.

*Service Water*

The service water system provides a heat sink for the removal of process and operating heat from safety related components following a design basis event. The accident analyses for the LOCA and the MSLB demonstrate that the system is capable of performing this function. As indicated in section II, above, these analyses have been run at a power level of 102% of current rated thermal power, thereby ensuring that the service water system will continue to perform this function at the proposed uprated power level.

In addition, the service water system provides a heat sink for the removal of process and operating heat from various non-safety related components during normal operation. The 1.4% proposed power uprate will have no effect on most of these components. An increase in generator output will result in increased heat removal requirements for the generator hydrogen coolers and the isophase bus duct coolers. However, this heat removal equipment was designed to be compatible with the generator, which will continue to be operated within its design rating.

Therefore, the service water system is not affected by the proposed power uprate.

D. Spent fuel pool storage and cooling systems

The design basis spent fuel pool heat load has been determined through analysis. The spent fuel pool cooling system is capable of removing this quantity of heat. Spent fuel may not be moved into the spent fuel pool until a specified minimum time has elapsed, thereby guaranteeing that the estimated spent fuel pool heat load is less than the design basis value. Analysis of the spent fuel pool heat load has been revised to ensure that spent fuel pool storage and cooling system temperatures are maintained within acceptable limits following the proposed power uprate.

Procedural controls prevent spent fuel from being moved into the spent fuel pool until it has decayed to the point where the design bases heat load can be met. Therefore, the spent fuel pool storage and cooling systems are not affected by the proposed power uprate.

E. Radioactive waste systems

The radioactive waste systems are designed to safely process radioactive wastes from the plant with a primary coolant activity based on 1% failed fuel rods and continuous purification during plant operation.

The functioning of the liquid radioactive waste system is independent of power level. The system is designed so that liquid releases to the environment are within 10 CFR 20 limits. Administrative controls ensure that the system is properly operated and that no credible failure could result in releases in excess of specified limits.

The gaseous radioactive waste system must be capable of limiting the radiological consequences that result from design basis accidents to acceptable values. The accident analyses for the gas decay tank rupture and the volume control tank rupture demonstrate that the system is capable of performing this function. As indicated in section II, above, these radiological analyses have been run at a power level of 102% of current rated thermal power, thereby ensuring that the gaseous radioactive waste system continues to perform this function at the proposed uprated power level.

Therefore, the radioactive waste systems are not affected by the proposed power uprate.

F. Engineered safety features heating, ventilation, and air conditioning systems

As discussed under "Containment Systems," the containment air coolers are not affected by the proposed power uprate. The 1.4% proposed power uprate results in no additional equipment and no additional loading of existing equipment in the engineered safeguards equipment rooms, control room, emergency diesel generator rooms, fuel handling area, or the electrical equipment, switchgear, cable spreading and battery rooms. As a result, the proposed power uprate will have no impact on the ventilation systems for these areas.

Therefore, the engineered safety features heating, ventilation, and air conditioning systems are not affected by the proposed power uprate.

G. Other (NSSS) Systems

*Chemical and Volume Control*

The chemical and volume control system has chemical, volume, and reactivity control functions.

FSAR section 9.10 states that the chemical and volume control system is designed to prevent the activity of the primary coolant from exceeding a specified value, assuming failed fuel elements. A plant power level of 2650 MWt was used in the original design of the primary coolant and associated radioactive systems. During normal operation, the primary coolant system activity is much less than the specified value. Thus, although the proposed increase in power may cause a slight increase in primary coolant system activity, the ability of the chemical control system to control primary coolant chemistry will not be affected by the proposed power uprate.

FSAR section 9.10 states that the chemical and volume control system must be capable of maintaining the required volume of water in the primary coolant system over the range of full to zero power without requiring makeup. Since neither the primary coolant system inventory, the primary coolant system pressure, the range for $T_{ave}$, nor the allowable rates for power changes, heatup, or cooldown are changing, charging and letdown flows will remain at their current values and the ability of the chemical and volume control system to maintain primary coolant system volume will not be affected by the proposed power uprate.

FSAR section 3.2.3 states that the chemical and volume control system must be capable of adding boric acid to the primary coolant at a rate sufficient to maintain the shutdown margin during a primary system cooldown at the design rate following a reactor trip. The allowable cooldown rate is not changing. Therefore, charging pumps will still be able to supply sufficient boric acid to ensure that the negative reactivity inserted is in excess of that needed to compensate for the reactivity changes resulting from primary system cooldown following a plant trip. Thus, the ability of the system to provide reactivity control will not be affected by the proposed power uprate.

The proposed power uprated full power value of $T_{cold}$ will be slightly lower than the current value. The effect of this change on the performance of the regenerative or letdown heat exchangers is insignificant. The design pressures and temperatures of the system will continue to bound all modes of operation.

Therefore, the chemical and volume control system is not affected by the proposed power uprate.

*Low Pressure Safety Injection*

As described in section II, above, the accident analyses, which involve the low pressure safety injection system, are performed with an initial power level of 102% of current rated thermal power. Therefore, the accident mitigation functions of the system are not affected by the proposed power uprate.

The system is also used during normal operation for shutdown cooling following a reactor shutdown. Since the proposed power uprate will result in a slightly higher decay heat load, this could slightly delay the initiation of shutdown cooling and could slightly increase the load on the system once initiated. However, these effects are expected to be insignificant. The initiation and operation of shutdown cooling will still be governed by the limits on primary coolant system temperature and pressure.

Therefore, the low pressure safety injection system is not affected by the proposed power uprate.

## VII. Other

1. With the exception of the increase in decay heat, the differences in the plant parameters involve operation at full power and do not affect the emergency operating procedures, which, typically, assume the reactor is shutdown prior to execution of the actions.

   In the case of the increase in decay heat, defense in depth and safety margins are demonstrated by means of the accident analyses, which (as explained in section II, above) have been based on an initial power level of 102% of current rated thermal power and, therefore, bound operation at the proposed power level of 101.4%. The operator actions in the emergency operating procedures are consistent with those assumed in the various accident analyses. Where a time-critical action occurs in an emergency operating procedure, it is reviewed to ensure that the accident analysis remains valid. As a result, no changes to the emergency operating procedures are required.

   The abnormal operating procedures were reviewed to assess the impact of operation at the proposed uprated power level. The slight difference in full power operating parameters does not require any change to the operator actions in these procedures. The slight increase in decay heat has been evaluated and no procedure changes are required.

   No changes to the emergency operating procedures, or the abnormal operating procedures, are required and there are no changes in operator actions that could adversely affect defense-in-depth or safety margins.

2. Modifications associated with proposed power uprate

   A. There are no changes to the emergency and off normal operating procedures required by this proposed power uprate, therefore no modifications to the facility are required.

   B. Control room controls and alarms are expected to be unaffected by the proposed power uprate. The control room indications for various plant parameters may change slightly, but will remain within the range of existing instrumentation. Various parameters monitored by the plant equipment operators may also change slightly but, likewise, remain within the range of existing instrumentation. If any operating parameters are discovered to be outside of the normal operating bands of the control room or other indicators, the operations department administrative controls ensure that the appropriate reviews are conducted and that, if necessary, procedure changes are made. Certain plant parameters may move closer to the alarm setpoint at full power, but substantial margin remains. As a result, no changes to the alarms are expected. The operator response to the alarms is unchanged.

      Operation at the proposed increased power conditions will have a negligible effect on the displays of the plant process computer.

   C. The proposed power uprate will not result in changes to the plant that affect operator performance. Although the proposed power uprate will not require changes in operator actions, plant controls, or plant alarms, the values of various plant parameters at full power are expected to change slightly. These expected changes will be reviewed with the operators as part of the operator training program.

3. Plant administrative procedures ensure that the necessary changes to the control room simulator and operator training program are made prior to implementation of the proposed power uprate.

4. The current plant requirement to maintain the four-hour average of reactor steady state power less than 100.1% indicated power will not be changed. An indicated power level of 100.1% represents 100% rated thermal power. Therefore, temporary operation above "full steady-state licensed power levels" is not permitted at Palisades and no procedure revision is required.

5. 10 CFR 51.22 Criteria for Categorical Exclusion

   A. The proposed power uprate was reviewed with respect to the criteria for a "categorical exclusion" from environmental assessment, in accordance with 10 CFR 51.21 and 10 CFR 51.22(c)(9), to ensure there is no significant increase in the types or significant increase in the amounts of any effluents that may be released offsite.

*Radioactive Effluents*

FSAR section 11.1 states that the normal sources of radioactive wastes are fission and activation products generated within the primary coolant system during plant operation. The proposed power uprate involves no change in materials or plant processes that could introduce radioactive wastes or radioactive effluents of a different type.

Although operation at a higher power level may result in a slight increase in activated products, this change in the inventory of radionuclides is expected to be insignificant. The proposed power uprate has no effect on charging or letdown flows. Thus, the effect on radioactive filters and demineralizers is expected to be insignificant and the processing of solid, liquid, and gaseous radioactive waste will not be impacted. As a result, there will be an insignificant change in the frequency or volume of planned releases.

The proposed power uprate will not change the primary coolant system operating pressure or the margin between the operating pressure and the system design rating, as shown in Section IV.1. Further, the proposed power uprate involves no change that could affect the mechanical or structural integrity of the fuel or fuel cladding. There is no effect on the containment systems. Thus, the proposed power uprate results in no additional challenges to any of the fission product barriers and there is no increase in the likelihood that fission or activated products will be released accidentally.

*Non-Radioactive Effluents*

The proposed power uprate involves no change in materials or plant processes that could introduce non-radioactive effluents of a different type. There will be a slight increase in the amount of heat, which is transferred to the cooling tower and circulating water systems.

A review of the existing environmental requirements, as given in the Final Environmental Statement, the Environmental Protection Plan, and the National Pollutant Discharge Elimination System permit, indicates that the effects of the proposed power uprate will continue to be bounded by these requirements.

Therefore, the proposed power uprate will result in no significant increase in the types, or significant increase in the amounts, of any effluents that may be released offsite.

B. The proposed power uprate was reviewed with respect to the criteria for a "categorical exclusion" from environmental assessment in accordance with 10 CFR 51.21 and 10 CFR 51.22(c)(9) to ensure there is no significant increase in individual or cumulative occupational radiation exposure.

FSAR section 11.6 explains that the plant shielding design, which was intended to ensure that occupational exposure remains within allowable limits, was based on a core power level of 2650 MWt, with 1% failed fuel. These design conditions will continue to bound plant operation at the proposed uprated power level.

Plant radiological and work control processes ensure that the radiation levels in the plant are maintained far below the allowable values.

The proposed power uprate can be expected to result in slightly higher radiation levels in containment during power operation. However, entry into containment inside the bioshield wall at full power is an infrequent occurrence and personnel exposure is administratively controlled.

Therefore, the proposed power uprate will produce no significant increase in individual or cumulative occupational radiation exposure.

## VIII. Changes to technical specifications, protection system settings, and emergency system settings

1. Changes to the Facility Operating License and the Technical Specifications (TS) are required to reflect the proposed new power level. The changes affect the maximum power level specified in the Facility Operating License and the definition of rated thermal power (RTP) in the TS and also the maximum allowable value for the variable high power trip (VHPT).

   A. Description of changes

      - Revise paragraph 2.C.(1) of Facility Operating License DPR-20 to authorize operation at steady-state reactor core power levels not in excess of 2565.4 Megawatts thermal (100 percent rated power).
      - Revise the definition of RTP in Appendix A, TS 1.1, from 2530 MWt to 2565.4 MWt.
      - Revise the maximum allowable value for the VHPT from 111% to 109.4% in Appendix A, TS Table 3.3.1-1, item 1.

Explanation of change for VHPT:

The proposed increase in RTP affects the VHPT allowable value, since the current maximum allowable value has been determined for 2530 MWt. The new proposed maximum allowable value of 109.4% was determined as follows:

(2530 MWt/2565.4 MWt)(111%) = 109.4%

B. No safety analyses are affected by the change. The maximum reactor power at actuation of the variable high power trip remains the same (2808.3 MWt).

Attachment 1

Uncertainty Calculation for the Secondary Calorimetric Heat Balance

EA-ELEC08-0001, Revision 1

CONSUMERS
ENERGY

# PALISADES NUCLEAR PLANT
# ENGINEERING ANALYSIS COVER SHEET

EA- ELEC08-0001

Total Number of Sheets 20

Title Uncertainty Calculation for the Secondary Calorimetric Heat Balance

## INITIATION AND REVIEW

| Calculation Status | Preliminary | Pending | Final | Superseded |
|---|---|---|---|---|
| | ☐ | X | ☐ | ☐ |

| Rev # | Description | Initiated By | Initiated Date | Init Appd By | Review Method: Alt Calc | Review Method: Detail Rev'w | Review Method: Qual Test | Technically Reviewed By | Technically Reviewed Date | Rev'r Appd By | Sup'v & S/DR Appd |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 0 | Original Issue | R.A. Bischoff | | RMB | | ✔ | | D.M. Kennedy | 4/5/02 | | RMB 4/18/02 |
| 1 | See Record Of Revision Sheet | RM Hamm | 7/15/02 | RMB | | ✓ | | DMKennedy | 8/15/02 | | RMB 8/24/02 |
| | | | | | | | | | | | |

## RECORD OF REVISION

| Revision Number | Description Of Change |
|---|---|
| 1 | EA-ELEC08-0004, "Uncertainty Calculation for UFM Corrected, Density Compensated, Total Feedwater Flow Measurement (PPC Only)" has been revised which resulted in an increase in the Total Loop Uncertainty used as an input to this calculation. The uncertainty of the temperature input to the heat balance changed due to the revision of EA-ELEC08-0004. The expected power uprate plant operating parameters are included in Section 4.1.1 as a major assumption. |

The indicated calculation status on the cover sheet has been changed to Pending.

Section 1.0; Added reference to power uprate.
Section 3.2.3; Changed $dW_{FWC}$ to 28,614.9 lbm/hr
Section 3.2.7; Changed $T_{FW}$ to ±3.63°F and 3.4134°F to 3.6251°F
Section 4.1.1; Added this section to provide plant parameters expected following the power uprate.
Section 5.1; Changed Total Uncertainty (Uncorrected) to
+1.13% Power -1.21% Power
Section 5.2; Changed Total Uncertainty (Corrected) to
+0.49% Power, -0.55% Power
Section 7.0; Changed Total Uncertainty (Uncorrected) to
+1.13% Power -1.21% Power
Changed Total Uncertainty (Corrected) to
+0.49% Power, -0.55% Power
Section 9.9; Changed the revision level to Rev.1
Section 9.11; Changed to RI-24A Revision 0.
Section 9.12; Added RI-24B Revision 0.

## 1.0 OBJECTIVE / SCOPE

This calculation will compute the uncertainty associated with the secondary calorimetric heat balance calculation with and without the Ultrasonic Flow Meter correction factor. Heat balance uncertainties are computed for the manual heat balance calculation performed through performance of DWO-1 with utilizing the power uprate values (Reference 9.10).

## 2.0 FUNCTIONAL DESCRIPTION

The Steam Generators (SG) serve to remove energy from the Primary Coolant System (PCS) and supply high quality steam to the main turbine and various auxiliary services. Steam exits the SGs through two 36" headers (Main Steam Lines). Each Main Steam Line (MSL) contains a Main Steam Isolation Valve which provides the ability to isolate the MSL from the remainder of the secondary system. An accurate calculation of Reactor power is obtained by performing a secondary heat balance.

The equation used to perform the Secondary Calorimetric power calculation is derived as follows:

Applying the 1st Law of Thermodynamics (Conservation of Energy) to the Primary Coolant System (PCS) yields the following energy equation:

$$\sum \text{Energy}_{IN} - \sum \text{Energy}_{OUT} = 0$$

Note: Steady state conditions are assumed.

For the PCS, it is reasonable to assume that no external work is performed and changes in kinetic energy are negligible. Therefore, only the heat sources and heat sinks in the PCS need to be considered. The following heat sources introduce energy (Q) into the Primary Coolant System:

- Reactor ($Q_{RX}$)
- Primary Coolant Pumps ($Q_{PCP}$)
- Pressurizer Heaters ($Q_{PZR}$)
- Charging Flow ($Q_{CH}$)

The following heat sinks remove energy (Q) from the Primary Coolant System:

- Steam Generators ($Q_{SG}$)
- Letdown Flow ($Q_{LD}$)
- Fixed Insulation Losses ($Q_{FL}$)

Substituting the heat sources and heat sinks into the energy equation and solving for $Q_{RX}$ yields the following equation:

$$Q_{RX} = Q_{SG} + Q_{LD} + Q_{FL} - Q_{PCP} - Q_{PZR} - Q_{CH}$$

Note: Energy removed from the Primary Coolant System due to PCS leakage is considered to be negligible and is not included in the Reactor Power equation.

Per Reference 9.7, the energy terms associated with the primary coolant pumps, letdown flow, charging flow, fixed insulation losses, and the pressurizer heaters are combined into one constant value (C). Therefore, the energy equation is simplified as follows:

$$Q_{RX} = Q_{SG} - C$$

Per Reference 9.7, the value of C will vary depending on the number of charging pumps / letdown orifices in service (up to 3 total). Reference 9.7 determines a conservatively low value of C for each combination with the following results:

C = -9.72 MWth (one orifice)
C = -8.52 MWth (two orifices)
C = -7.14 MWth (three orifices)

DWO-1 is utilized to compute the value of $Q_{SG}$ given in the equation presented above.

The determination of the Steam Generator term ($Q_{SG}$) requires the application of the energy equation with the Steam Generator considered as the control volume:

The following heat sources introduce energy into the Steam Generators:

- Energy from the PCS ($Q_{SG}$)
- Feedwater Flow ($Q_{FW}$)

The following heat sinks remove energy from the Steam Generators:

- Blowdown Flow ($Q_{BD}$)
- Steam Flow ($Q_{ST}$)

Substituting the heat sources and heat sinks into the energy equation and solving for $Q_{SG}$ yields the following equation:

$$Q_{SG} = Q_{ST} + Q_{BD} - Q_{FW}$$

## 3.0 ANALYSIS INPUTS

### 3.1 HEAT BALANCE UNCERTAINTY EQUATION

Per Reference 9.4, the following equation represents the heat balance calculation as computed by DWO-1:

$$Q_{SG} = \left(435.804 + 0.1753\,P_{SG} - 1.1045\,T_{FW} + X\left(851.789 - 0.20257\,P_{SG}\right)\right)W_{FW} - X\left(851.789 - 0.20257\,P_{SG}\right)W_{BD}$$

where,

$Q_{SG}$ = Heat removed from the PCS by the Steam Generators (btu / hr)
$P_{SG}$ = Steam Generator Pressure (psia)
$T_{FW}$ = Feedwater Temperature (°F)
$X$ = Steam quality (unit-less)
$W_{FW}$ = Feedwater flow (lbm / hr)
$W_{BD}$ = Blowdown flow (lbm / hr)

Note: The equation given above is simplified in DWO-1 and broken down into multiple steps. Steam quality is not measured when performing the heat balance.

The effects of instrument uncertainties on the heat balance are computed by taking the total derivative of the overall energy equation given above as follows:

$$dQ_{SG} = \frac{\partial Q_{SG}}{\partial W_{FW}}dW_{FW} + \frac{\partial Q_{SG}}{\partial W_{BD}}dW_{BD} + \frac{\partial Q_{SG}}{\partial P_{SG}}dP_{SG} + \frac{\partial Q_{SG}}{\partial T_{FW}}dT_{FW} + \frac{\partial Q_{SG}}{\partial X}dX$$

Using the methodology described in References 9.1 and 9.2, the individual random uncertainty terms are combined using the Square Root Sum of Squares method as follows:

$$dQ_{SG} = \sqrt{\left(\frac{\partial Q_{SG}}{\partial W_{FW}}dW_{FW}\right)^2 + \left(\frac{\partial Q_{SG}}{\partial W_{BD}}dW_{BD}\right)^2 + \left(\frac{\partial Q_{SG}}{\partial P_{SG}}dP_{SG}\right)^2 + \left(\frac{\partial Q_{SG}}{\partial T_{FW}}dT_{FW}\right)^2 + \left(\frac{\partial Q_{SG}}{\partial X}dX\right)^2}$$

The partial derivatives in the equation given above represent the weighting factor of each parameter used for the heat balance calculation. The differentials in the equation given above represent the uncertainty associated with each parameter.

The partial derivatives are as follows:

$$\frac{\partial Q_{SG}}{\partial W_{FW}} = 435.804 + 0.1753\,P_{SG} - 1.1045\,T_{FW} + X\left(851.789 - 0.20257\,P_{SG}\right)$$

$$\frac{\partial Q_{SG}}{\partial W_{BD}} = -\,X\left(851.789 - 0.20257\,P_{SG}\right)$$

$$\frac{\partial Q_{SG}}{\partial T_{FW}} = -1.1045\,W_{FW}$$

$$\frac{\partial Q_{SG}}{\partial P_{SG}} = 0.1753\,W_{FW} + X\left(0.20257\right)\left(W_{BD} - W_{FW}\right)$$

$$\frac{\partial Q_{SG}}{\partial X} = \left(851.789 - 0.20257\,P_{SG}\right)\left(W_{FW} - W_{BD}\right)$$

The following nominal full power values are used to compute the value of each partial derivative:

| | | |
|---|---|---|
| $W_{FW}$ | = 5,678,500 lbm / hr | [Reference 9.10] |
| $W_{BD}$ | = 30,000 lbm / hr | [Reference 9.4] |
| $T_{FW}$ | = 440.7°F | [Reference 9.10] |
| $P_{SG}$ | = 765.8 psia | [Reference 9.10] |
| X | = 0.9989 | [Average of SG A value and SG B value from Reference 9.4] |

Substituting these values into each partial derivative yields the following weighting factors for each parameter used in the heat balance calculation:

$$\frac{\partial Q_{SG}}{\partial W_{FW}} = 779.190 \text{ btu / lbm}$$

$$\frac{\partial Q_{SG}}{\partial W_{BD}} = -695.895 \text{ btu / lbm}$$

$$\frac{\partial Q_{SG}}{\partial T_{FW}} = -6{,}271{,}903 \text{ btu / (hr - °F)}$$

$$\frac{\partial Q_{SG}}{\partial P_{SG}} = -147{,}517 \text{ btu / (hr - psia)}$$

$$\frac{\partial Q_{SG}}{\partial X} = 3{,}935{,}089{,}059 \text{ btu / hr}$$

## 3.2 HEAT BALANCE INPUT UNCERTAINTIES

3.2.1 Per Reference 9.5, the uncertainty associated with the blowdown flow ($W_{BD}$) input to the heat balance uncertainty calculation is as follows:

$$dW_{BD} = \pm 2{,}500 \text{ lbm / hr}$$

3.2.2 Per Reference 9.4, the uncertainty associated with the steam quality (dX) measurement which is used in the heat balance calculation is as follows:

$$dX = \pm 0.00016$$

3.2.3 Per Reference 9.3, the feedwater flow input to the heat balance is obtained by reading PPC points FEEDWTR_FLOW_SGA_AVE and FEEDWTR_FLOW_SGB_AVG. If these points are not available, alternate computer points (FT-0701 and FT-0703) are used to measure feedwater flow. Per Reference 9.6, the Ultrasonic Flow Meter (UFM) corrected uncertainty ($dW_{FWc}$) associated with the PPC feedwater flow reading consists of the uncertainty associated with the flow transmitter and the PPC A/D uncertainty:

$$dW_{FWc} = \pm 28{,}614.9 \text{ lbm / hr}$$

Per Reference 9.10, the nominal post power uprate feedwater flow rate is 5,678,500 lbm / hr. Therefore,

$$dW_{FWc} = \pm 0.50 \text{ \% Flow}$$

3.2.4 Per Reference 9.11, the <u>random</u> uncertainty of the PPC feedwater flow reading (without UFM correction) consists of the uncertainty associated with the flow element, the flow transmitter, the temperature loop error, and the PPC A/D uncertainty:

$$dW_{FW} = \pm 0.24\text{\% Flow}$$

Using the post uprate feedwater flow value per steam generator from Reference 9.10, 5,678,500 lbm / hr, yields the following uncertainty expressed in units of lbm / hr:

$$dW_{FW} = \pm 13{,}628.4 \text{ lbm / hr @ 100\% Power}$$

3.2.5 Per Reference 9.11, the bias uncertainty associated with the feedwater flow measurement (without UFM correction) consists of the bias uncertainty associated with the flow element and the As-Left tolerance of the transmitter. Therefore, the total bias uncertainty is treated as a bi-directional bias as follows:

Bias = (±0. 50% ±0.25%) Flow @ 100% Power
Bias = ±0.75% Flow @ 100% Power

Using the post uprate feedwater flow value from Reference 9.10, 5,678,500 lbm / hr, yields the following uncertainty expressed in units of lbm / hr:

$dW_{FWb}$ = ±42,588 lbm / hr @ 100% Power

3.2.6 At the present time, per Reference 9.3, PPC points PT_0751B and PT_0752B are used to obtain Steam Generator pressure. If these points are unavailable, Steam Generator pressure ($P_{SG}$) indicators PIC-0751A, 0751B, 0751C, and 0751D are averaged to obtain Steam Generator A pressure, and PIC-0752A, 0752B, 0752C, and 0752D are averaged to obtain Steam Generator B pressure. Per Assumption 4.1, the procedure will require the average of at least 3 Steam Generator Pressure readings per Steam Generator when the heat balance is performed in the future. Per Reference 9.8, the uncertainty associated with PIC-0751A, 0751B, 0751C, 0751D and PIC-0752A, 0752B, 0752C, and 0752D is as follows:

$eP_{SG}$ = +27.48 psia -28.44 psia

For conservatism, the Steam Generator pressure indicator uncertainty is rounded to ±29 psia, and one steam generator pressure channel is assumed to be out of service. Therefore, the uncertainty associated with the averaging of three pressure inputs is computed as follows:

$$dP_{SG} = \pm\sqrt{\left(\frac{29\text{ psia}}{3}\right)^2+\left(\frac{29\text{ psia}}{3}\right)^2+\left(\frac{29\text{ psia}}{3}\right)^2}$$

$dP_{SG}$ = ±16.74 psia

3.2.7 Per Reference 9.3, the feedwater temperature input to the heat balance is obtained by reading PPC points HB_TEMP_STEADY_SGA and HB_TEMP_STEADY_SGB. If these points are not available, alternate computer points (TT_0706A and TT_0708A) are used to measure feedwater temperature. Reference 9.5 provides a PPC Feedwater Temperature uncertainty value of
± 1.3°F. However, Reference 9.6 calculates a more conservative uncertainty associated with the PPC feedwater temperature reading of ± 3.63°F (rounded up from 3.6251°F). Though the results of Reference 9.6 (stated in Section 8.0) are valid for restricted use, this value is used as it bounds the value from Reference 9.5. Therefore:

$$dT_{FW} = \pm 3.63° F$$

NOTE: *Temperature input uncertainties are calculated for single point real time FW Temperature measurements. Any time averaging of FW Temperature values prior to use in calorimetric calculations would provide FW Temperature (and uncertainty) values bounded by the single point real time FW Temperature measurements.*

3.2.8 Per Reference 9.4, the Heat Balance Uncertainty equation (stated in Section 3.1) has, as part of its basis, enthalpy calculation equations. Differences between steam enthalpies determined by using these equations and those determined using the ASME Steam Tables could impart a bias uncertainty into the calculation of overall uncertainty for the secondary calorimetric. This additional bias term is computed below by determining steam and feedwater enthalpy errors at various points and choosing a representative bias from the calculated errors.

Attachment A show the determination of bounding values for enthalpy errors in Feedwater and Steam. ASME Steam enthalpies were determined for saturated steam conditions, while feedwater enthalpies were determined for compressed liquid at 830 psia. Minor differences between the assumed feedwater pressure and actual feedwater pressure would result in negligible enthalpy bias differences. Feedwater enthalpy Delta h error is bounded by +0.17 btu / lbm. In other words, calculated hFW is larger than actual hFW. Steam (mixture enthalpy Delta h error is bounded by –0.08 btu / lbm. In the calculation of secondary calorimetric uncertainty, these errors would result in thermal power calculations that are lower than actual thermal power which is non conservative. These errors can be added to yield an overall enthalpy bias term, as follows:

$$hb = -(|hsb| + |hfb|)$$

where:
hb = total enthalpy bias uncertainty
hsb = steam enthalpy bias uncertainty
hfb = feedwater enthalpy bias uncertainty

Therefore:

$$hb = -(0.08 + 0.17) \text{ btu/lbm}$$
$$hb = -0.25 \text{ btu / lbm}$$

## 3.3 HEAT BALANCE INPUTS

3.3.1 Per Reference 9.7, a conservatively determined constant value (C) is used to account for the energy terms associated with the primary coolant pumps, letdown flow, charging flow, fixed insulation losses, and the pressurizer heaters. The constant C varies depending on the number of charging pump / letdown flow orifices in service (up to three total). Parameters such as charging flow, letdown flow, etc. are relatively constant during the performance of the heat balance, and treating these parameters as constants simplifies the heat balance. Reference 9.7 demonstrates that the calculated C values are lower than actual, which would tend to compensate for any actual variations in the parameters that make up the C constant. The following equation from Section 2.0 demonstrates that using a lower than actual C value is conservative.

$$Q_{RX} = Q_{SG} - C$$

This equation shows that using a smaller value of "C" would result in a higher calculated "$Q_{RX}$". As nuclear instrumentation is calibrated to the heat balance results, this would cause indicated power to be greater than actual, which is conservative.

3.3.2 The following plant parameters are used for the heat balance uncertainty calculation:

$P_{SG}$ = Steam Generator Pressure (psia) [Reference 9.10]
$P_{SG}$ = 765.8 psia

$T_{FW}$ = Feedwater Temperature (°F) [Reference 9.10]
$T_{FW}$ = 440.7°F

X = Steam quality (unit-less) [Reference 9.4]
X = 0.9989

$W_{BD}$ = Blowdown flow (lbm / hr) [Reference 9.4]
$W_{BD}$ = 30,000 lbm / hr

## 4.0 ASSUMPTIONS

### 4.1 MAJOR ASSUMPTIONS

4.1.1 Per Reference 9.10, the following plant parameters are anticipated after the power up-rate project. If actual plant conditions are similar to these, this calculation remains valid.

$Q_R$ = Reactor 100% Power
$Q_R$ = 2565.4 MWt

$P_{SG}$ = Steam Generator Pressure (psia)
$P_{SG}$ = 765.8 psia

$T_{FW}$ = Feedwater Temperature (°F)
$T_{FW}$ = 440.7°F

$F_{FW}$ = Feedwater Flow ($Mlb_m$ / hr)
$F_{FW}$ = 11.357 $Mlb_m$ / hr

### 4.2 MINOR ASSUMPTIONS

4.2.1 Per Reference 9.3 the manual heat balance calculation, without the PPC Steam Generator pressure values available, will use the average of at least three pressure indications per Steam Generator each time the Secondary Heat Balance is performed.

4.2.2 All uncertainties associated with the ultrasonic flow meter are random and independent.

## 5.0 ANALYSIS

Computations are performed to an accuracy of several significant digits, but presented in this calculation rounded to two decimal places in most cases. Hand verification of this calculation utilizing the rounded values may result in slightly different results due to round off errors.

## 5.1 SECONDARY HEAT BALANCE UNCERTAINTY (Without UFM Correction)

**Random Uncertainties**

Per Analysis Input 3.1, the following equation is used to compute the random uncertainties associated with the heat balance:

$$dQ_{SG} = \sqrt{\left(\frac{\partial Q_{SG}}{\partial W_{FW}} dW_{FW}\right)^2 + \left(\frac{\partial Q_{SG}}{\partial W_{BD}} dW_{BD}\right)^2 + \left(\frac{\partial Q_{SG}}{\partial P_{SG}} dP_{SG}\right)^2 + \left(\frac{\partial Q_{SG}}{\partial T_{FW}} dT_{FW}\right)^2 + \left(\frac{\partial Q_{SG}}{\partial X} dX\right)^2}$$

The uncertainties associated with each input parameter (differentials) are as follows:

| | | | |
|---|---|---|---|
| $dW_{FWUr}$ | = feedwater flow (random) | = ±13,628 lbm / hr | [Analysis Input 3.2.4] |
| $dW_{FWUb}$ | = feedwater flow (bias) | = ± 42,588 lbm / hr | [Analysis Input 3.2.5] |
| $dW_{BD}$ | = blowdown flow | = ±2,500 lbm / hr | [Analysis Input 3.2.1] |
| $dT_{FW}$ | = feedwater temperature | = ±3.63 ° F | [Analysis Input 3.2.7] |
| $dP_{SG}$ | = Steam Generator pressure | = ±16.74 psia | [Analysis Input 3.2.6] |
| $dX$ | = steam quality | = ±0.00016 | [Analysis Input 3.2.2] |

Per Analysis Input 3.1, the partial derivatives are as follows:

$$\frac{\partial Q_{SG}}{\partial W_{FW}} = 779.190 \text{ btu / lbm}$$

$$\frac{\partial Q_{SG}}{\partial W_{BD}} = -695.895 \text{ btu / lbm}$$

$$\frac{\partial Q_{SG}}{\partial T_{FW}} = -6{,}271{,}903 \text{ btu / (hr - °F)}$$

$$\frac{\partial Q_{SG}}{\partial P_{SG}} = -147{,}517 \text{ btu / (hr - psia)}$$

$$\frac{\partial Q_{SG}}{\partial X} = 3{,}935{,}089{,}059 \text{ btu / hr}$$

Therefore, the random uncertainty associated with the heat balance calculation is as follows:

$$dQ_{SG} = \pm 25{,}310{,}409 \text{ btu / hr}$$

Per Reference 9.4, the conversion from btu / hr to Wt is performed by multiplying by a factor of 0.29293 Wt – hr/btu. Per Reference 9.10, 100% Power equates to 2,565.4 MWt. Therefore, the random heat balance uncertainty is converted to % Power with the following equation:

MWt. Therefore, the random heat balance uncertainty is converted to % Power with the following equation:

$$dQ_{SG}\,(\%\ \text{Power}) = dQ_{SG}(\text{btu / hr})\left(0.29293\,\frac{\text{Wt}}{\text{btu / hr}}\right)\left(\frac{100\%\ \text{Power}}{2{,}565{,}400{,}000\ \text{Wt}}\right)$$

Therefore,

$$dQ_{SG} = \pm 0.29\ \%\ \text{Power (Single Steam Generator)}$$

The total random uncertainty associated with the heat balance calculation is computed with the following equation:

$$dQ_{SG} = \sqrt{dQ_{SG}^{\,2} + dQ_{SG}^{\,2}}\ \text{(Both Steam Generators)}$$
$$dQ_{SG} = \pm 0.41\ \%\ \text{Power (Both Steam Generators)}$$

## Bias Uncertainties

There are two bias uncertainties that must be considered, Feedwater Flow Measurement bias (FWb) and enthalpy bias (hb). These bias terms are calculated below, then combined to yield the total bias uncertainty for one steam generator. This result can be multiplied by 2 to yield total bias uncertainty for both steam generators.

### Feedwater Flow Measurement Bias Uncertainty

Per Section 3.2.5, the following bi-directional bias uncertainty is associated with the feedwater flow measurement for one Steam Generator:

$$FWb\ (\text{lbm / hr}) = \pm 42{,}588\ \text{lbm / hr}$$

Per Analysis Input 3.1, the weighting factor (partial derivative) associated with the feedwater flow heat balance input is as follows:

$$\frac{\partial Q_{SG}}{\partial W_{FW}} = 779.190\ \text{btu / lbm}$$

The feedwater flow measurement bias term is converted to btu / hr with the following equation:

$$FWb\ (\text{btu / hr}) = FWb\ (\text{lbm/hr})\left(\frac{\partial Q_{SG}}{\partial W_{FW}}\right)$$
$$FWb\ (\text{btu / hr}) = \pm 33{,}184{,}144\ \text{btu / hr}$$

### Enthalpy Bias Uncertainty

Per Section 3.2.8, the following enthalpy bias is associated with the feedwater flow to each steam generator which is 5,678,500 lbm/hr:

hb = -0.25 btu / lbm
hb (btu / hr) = hb (btu / lbm) * $W_{FW}$

hb (btu/hr) = -0.25 btu / lbm * 5,678,500 lbm / hr
hb (btu / hr) = -1,419,625 btu / hr

**Total Bias Uncertainty**

Total bias uncertainty ($Bias_T$) is the sum of the Feedwater Flow Measurement uncertainty bias and the enthalpy bias. Therefore:

$Bias_T$ = FWb + hb
$Bias_T$ = ±33,184,144 btu / hr -1,419,625 btu / hr
$Bias_T$ = +31,764,519 btu / hr - 34,603,769 btu / hr

Per Reference 9.4, the conversion from btu / hr to Wt is performed by multiplying by a factor of 0.29293 Wt – hr/btu. Per Reference 9.10, 100% Power equates to 2,565.4 MWt. Therefore, the bias term is converted to % Power utilizing the following equation:

$$Bias_T\ (\%\ Power) = Bias\ (btu/hr)\left(0.29293\ \frac{Wt}{btu/hr}\right)\left(\frac{100\%\ Power}{2{,}565{,}400{,}000\ Wt}\right)$$

$Bias_T$ (% Power) = +0.36% Power -0.40 % Power (One Steam Generator)

The total bias uncertainty associated with the feedwater flow to both Steam Generators is obtained by multiplying the bias uncertainties for a single Steam Generator by "2". Therefore,

$Bias_T$ (% Power) = + 0.72% -0.80% Power (Both Steam Generators)

Secondary Heat Balance Total Uncertainty (Uncorrected Feedwater Flow)

Total Uncertainty (Uncorrected) = ±0.41% Power + 0.72% Power –0.80% Power
Total Uncertainty (Uncorrected) = +1.13% Power – 1.21% Power

## 5.2 SECONDARY HEAT BALANCE UNCERTAINTY (With UFM Correction)

**Random Uncertainties**

Per Analysis Input 3.1, the following equation is used to compute the random uncertainties associated with the heat balance:

$$dQ_{SG} = \sqrt{\left(\frac{\partial Q_{SG}}{\partial W_{FW}} dW_{FW}\right)^2 + \left(\frac{\partial Q_{SG}}{\partial W_{BD}} dW_{BD}\right)^2 + \left(\frac{\partial Q_{SG}}{\partial P_{SG}} dP_{SG}\right)^2 + \left(\frac{\partial Q_{SG}}{\partial T_{FW}} dT_{FW}\right)^2 + \left(\frac{\partial Q_{SG}}{\partial X} dX\right)^2}$$

The uncertainties associated with each input parameter (differentials) are as follows:

| | | | |
|---|---|---|---|
| $dW_{FW}$ | = UFM corrected feedwater flow | = ±28,615 lbm / hr | [Analysis Input 3.2.3] |
| $dW_{BD}$ | = blowdown flow | = ±2,500 lbm / hr | [Analysis Input 3.2.1] |
| $dT_{FW}$ | = feedwater temperature | = ±3.63°F | [Analysis Input 3.2.7] |
| $dP_{SG}$ | = Steam Generator pressure | = ±16.74 psia | [Analysis Input 3.2.6] |
| $dX$ | = steam quality | = ±0.00016 | [Analysis Input 3.2.2] |

Per Analysis Input 3.1, the partial derivatives are as follows:

$$\frac{\partial Q_{SG}}{\partial W_{FW}} = 779.190 \text{ btu / lbm}$$

$$\frac{\partial Q_{SG}}{\partial W_{BD}} = -695.895 \text{ btu / lbm}$$

$$\frac{\partial Q_{SG}}{\partial T_{FW}} = -6{,}271{,}903 \text{ btu / (hr - °F)}$$

$$\frac{\partial Q_{SG}}{\partial P_{SG}} = -147{,}517 \text{ btu / (hr - psia)}$$

$$\frac{\partial Q_{SG}}{\partial X} = 3{,}935{,}089{,}059 \text{ btu / hr}$$

Therefore, the random uncertainty associated with the heat balance calculation is as follows:

$dQ_{SG}$ = ±32,015,507 btu / hr

Per Reference 9.4, the conversion from btu / hr to Wt is performed by multiplying by a factor of 0.29293 Wt – hr/btu. Per Reference 9.10, 100% Power equates to 2,565.4 MWt. Therefore, the random heat balance uncertainty is converted to % Power with the following equation:

$$dQ_{SG}\ (\%\ \text{Power}) = dQ_{SG}(\text{btu/hr})\left(0.29293\ \frac{\text{Wt}}{\text{btu/hr}}\right)\left(\frac{100\%\ \text{Power}}{2{,}565{,}400{,}000\ \text{Wt}}\right)$$

Therefore,

$$dQ_{SG} = \pm 0.3656\%\ \text{Power (Single Steam Generator)}$$

The total random uncertainty associated with the heat balance calculation is computed with the following equation:

$$dQ_{SG} = \sqrt{dQ_{SG}^{\ 2} + dQ_{SG}^{\ 2}}\ \text{ (Both Steam Generators)}$$

$$dQ_{SG} = \pm 0.52\%\ \text{Power (Both Steam Generators)}$$

**Bias Uncertainties**

Per Section 3.3.3, there are no feedwater flow measurement bias terms associated with the heat balance when UFM corrected feedwater flow is utilized as an input. Per Section 3.2.8, there is an enthalpy bias term that must be considered.

**Enthalpy Bias Uncertainty**

Per Section 3.2.8, the following enthalpy bias is associated with the feedwater flow to each steam generator which is 5,678,500 lbm/hr:

$$hb = -0.25\ \text{btu/lbm}$$
$$hb\ (\text{btu/hr}) = hb\ (\text{btu/lbm}) * W_{FW}$$

$$hb\ (\text{btu/hr}) = -0.25\ \text{btu/lbm} * 5{,}678{,}500\ \text{lbm/hr}$$
$$hb\ (\text{btu/hr}) = -1{,}419{,}625\ \text{btu/hr}$$

Per Reference 9.4, the conversion from btu / hr to Wt is performed by multiplying by a factor of 0.29293 Wt – hr/btu. Per Reference 9.10, 100% Power equates to 2,565.4 MWt. Therefore, the bias term is converted to % Power utilizing the following equation:

$$\text{Bias}\ (\%\ \text{Power}) = Bias\,(btu/hr)\left(0.29293\,\frac{\text{Wt}}{\text{btu/hr}}\right)\left(\frac{100\%\ \text{Power}}{2{,}565{,}400{,}000\ \text{Wt}}\right)$$

$$\text{Bias}\ (\%\ \text{Power}) = -0.0162\ \%\ \text{Power (One Steam Generator)}$$

The total bias uncertainty associated with the feedwater flow to both Steam Generators is obtained by multiplying the bias uncertainties for a single Steam Generator by "2". Therefore,

Bias (% Power) = -0.03% Power (Both Steam Generators)

Secondary Heat Balance Total Uncertainty - UFM Corrected Feedwater Flow

Total Uncertainty (UFM Corrected) = ±0.52% Power -0.03% Power
Total Uncertainty (UFM Corrected) = +0.49% Power – 0.55% Power

## 6.0 SETPOINT EVALUATION

No setpoints are addressed by this calculation.

## 7.0 SUMMARY OF RESULTS

The uncertainty associated with the secondary heat balance utilizing both Ultrasonic Flow Meter corrected feedwater flow and uncorrected feedwater flow measurements were computed in Section 5.1 and 5.2 of this calculation. The uncertainties expressed in terms of % Power are as follows:

Secondary Heat Balance Total Uncertainty (Uncorrected Feedwater Flow)

Total Uncertainty (Uncorrected) = ±0.41% Power + 0.72% Power –0.80% Power
Total Uncertainty (Uncorrected) = +1.13% Power – 1.21% Power

Secondary Heat Balance Total Uncertainty (Ultrasonic Feedwater Flow Corrected)

Total Uncertainty (Corrected) = ±0.52% Power -0.03% Power
Total Uncertainty (Corrected) = +0.49% Power – 0.55% Power

## 8.0 CONCLUSION

This calculation computed the uncertainty associated with the secondary calorimetric heat balance calculation with and without the Ultrasonic Flow Meter correction factor. Heat balance uncertainties were computed for the manual heat balance calculation performed through performance of DWO-1. See Section 7.0 for results. The results of this calculation are subject to the following limitations:

- This calculation assumes that Reference 9.3 will be revised to use the average of PIC-0751A, 0751B, 0751C, and 0751D to obtain Steam Generator A pressure, and the average of PIC-0752A, 0752B, 0752C, and 0752D will be used to obtain Steam Generator B pressure every time the Secondary Heat Balance is performed. The results of this calculation are based on the use of at least 3 Steam Generator Pressure indications per steam generator.

- Per Reference 9.3, the feedwater flow control room indicators (FI-0701 and FI-0703) may also be used as the feedwater flow input to the heat balance. The uncertainty associated with the control room feedwater flow indicators is larger than the uncertainty associated with the PPC computer point indications of feedwater flow. Reference 9.6 does not compute the uncertainties associated with the feedwater flow control room indicators. The results of this calculation are based on the use of the PPC computer points for the feedwater flow measurement.

- Per Reference 9.3, the feedwater temperature control room indicators (TI-0706 and TI-0708) or recorder TR-0706 may also be used as the feedwater temperature input to the heat balance. The uncertainties associated with the control room indicators and the recorder are larger than the uncertainty associated with the PPC computer point indications of feedwater temperature. Reference 9.6 does not compute the uncertainties associated with the analog indications of feedwater temperature. The results of this calculation are based on the use of the PPC computer points for the feedwater temperature measurement.

## 9.0 REFERENCES

9.1 NUREG/CR-3659, "A Mathematical Model for Assessing the Uncertainties of Instrumentation Measurements for Power and Flow of PWR Reactors", Dated February 1985.

9.2 ISA-RP67.04, Part II - 1994, "Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation", May 1995.

9.3 DWO-1, "Operator's Daily / Weekly Items, Modes 1, 2, 3, and 4", Revision 60.

9.4 EA-HAR-91-10, "Heat Balance Adjustment for Moisture Content of Steam", Revision 0.

9.5 EA-AFZ-96-01, "Analysis of Various Heat Balance Input Inaccuracies", Revision 2.

9.6 EA-ELEC08-0004, " Uncertainty Calculation for UFM Corrected, Density Compensated Total Feedwater Flow Measurement (PPC Only)", Revision 1.

9.7 EA-BWB-96-01, "Heat Balance Calculation Using the Ultrasonic Flowmeter Measurement Device", Revision 3.

9.8 EA-ELEC08-97-04, "Uncertainty Calculation Steam Generator Pressure Loops", Revision 1.

9.9 EA-UFM-97-01, "Feedwater Flow Uncertainty with UFM Correction Factor", Revision 1.

9.10 EA-RCH-01-05, "Calculation of Chapter 14 Safety Analysis Parameter Changes Due to FC-977 Power Uprate," Revision 0.

9.11 RI-24A, "Steam Generator Feedwater Temperature Instrument Loop Calibration," Procedure, Revision 0.

9.12 RI-24B, "Steam Generator Feedwater Flow Instrument Loop Calibration," Procedure, Revision 0.

# ATTACHMENT A

The following table establishes bounding differences between steam and feedwater enthalpy taken from ASME Steam Tables and calculated using approximations from Reference 9.4.

**STEAM** **(Saturated**

(SGA) hm = 1220.36-

(SGB) hm = 1219.97-

XA = 0.999135
XB = 0.998668

| PSG (psia) | ASME hg | hf | hmA | Calc. hmA | Delta h | ASME hmB | Calc. hmB | Delta h |
|---|---|---|---|---|---|---|---|---|
| 760 | 1200.44 | 502.7 | 1199.836 | 1199.772 | **-0.065** | 1199.511 | 1199.45 | **-0.061** |
| 765.8 | 1200.29 | 503.75 | 1199.687 | 1199.614 | **-0.073** | 1199.362 | 1199.293 | **-0.069** |
| 770 | 1200.18 | 504.5 | 1199.578 | 1199.501 | **-0.078** | 1199.253 | 1199.18 | **-0.073** |

**FEEDWATER ENTHALPY (Compressed liquid at 830**

hFW = 1.1045*TFW -

| TFW deg. F | ASME hFW | Calc. hFW | Delta h |
|---|---|---|---|
| 442.7 | 422.32 | 422.47 | **0.150** |
| 440.7 | 420.11 | 420.26 | **0.150** |
| 438.7 | 417.89 | 418.05 | **0.160** |