CP&L
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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

RESPONSE TO REQUESTS FOR ADDITIONAL INFORMATION ON AMENDMENT REQUEST TO INCREASE AUTHORIZED REACTOR POWER LEVEL (TAC NO. MB5106)

Ladies and Gentlemen:

By letter dated May 16, 2002, Carolina Power and Light (CP&L) Company submitted a request for amendment to the Technical Specifications (TS) to increase the authorized reactor power for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The NRC has requested additional information that is required to complete the review of the proposed amendment by letters dated June 26, 2002, and July 17, 2002.

Attachment I provides an Affirmation as required by 10 CFR 50.30(b). Attachment II provides responses to the requests for additional information (RAI). Attachment III provides Caldon Engineering Report ER-267N, "Bounding Uncertainty Analysis for Thermal Power Determination at CP&L Robinson Nuclear Power Station Using the LEFM Check Plus System," Revision 0, non-proprietary version.

A proprietary version of the Caldon Engineering Report is supplied as Attachment V. The information contained in Attachment V is considered by Caldon, Incorporated, to contain proprietary information. Therefore, Caldon, Incorporated, requests exemption from public disclosure in accordance with 10 CFR 2.790(b). Attachment IV contains an affidavit and application for withholding from public disclosure executed by Mr. Calvin R. Hastings, President and CEO of Caldon, Incorporated, who is authorized to apply for the withholding of the proprietary information for Caldon, Incorporated.

The license amendment request for the increase in authorized reactor power level, as submitted by letter dated May 16, 2002, relies upon the approval of another proposed licensed amendment request to implement alternate source term (AST) methodology. In discussions with the NRC staff, it was determined that review and approval of the power increase license amendment prior to the upcoming refueling outage, which is scheduled to start on October 12, 2002, would be

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facilitated by modifying this request so as to remove the reliance on approval of the AST license amendment request. Therefore, CP&L intends to submit a supplement to this license amendment request to utilize the provisions within Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," that permit the radiological consequences of affected accident analyses to be dispositioned in accordance with NRC-approved reload methodologies.

In accordance with 10 CFR 50.91(b), CP&L is providing the State of South Carolina with a copy of this response.

The responses to the NRC RAI provide additional information that does not affect the basis or justification for the proposed TS change, including the evaluation of No Significant Hazards Consideration provided within the May 16, 2002, submittal.

If you have any questions concerning this matter, please contact Mr. C. T. Baucom.

Sincerely,

B. L. Fletcher III
Manager - Regulatory Affairs

CAC/cac

Attachments:

I. Affirmation
II. Response To Requests For Additional Information On Amendment Request to Increase Authorized Reactor Power Level
III. Caldon, Inc., Engineering Report ER-267N, "Bounding Uncertainty Analysis for Thermal Power Determination at CP&L Robinson Nuclear Power Station Using the LEFM Check Plus System," Revision 0, non-proprietary version
IV. Affidavit and Application for Withholding from Public Disclosure
V. Caldon, Inc., Engineering Report ER-267, "Bounding Uncertainty Analysis for Thermal Power Determination at CP&L Robinson Nuclear Power Station Using the LEFM Check Plus System," Revision 0, proprietary version

c: Mr. L. A. Reyes, NRC, Region II
Mr. H. J. Porter, Director, Division of Radioactive Waste Management (SC)
Mr. R. M. Gandy, Division of Radioactive Waste Management (SC)
Mr. R. Subbaratnam, NRC, NRR
NRC Resident Inspector, HBRSEP
Attorney General (SC)

AFFIRMATION

The information contained in letter RNP-RA/02-0104 is true and correct to the best of my information, knowledge and belief; and the sources of my information are officers, employees, contractors, and agents of Carolina Power and Light Company. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: JUL 25 2002

JMoyer

J. W. Moyer
Vice President, HBRSEP, Unit No. 2

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

RESPONSE TO REQUESTS FOR ADDITIONAL INFORMATION ON AMENDMENT REQUEST TO INCREASE AUTHORIZED REACTOR POWER LEVEL

By letters dated June 26, 2002, and July 17, 2002, the NRC issued Requests for Additional Information (RAI) regarding Carolina Power and Light (CP&L) Company's request for an amendment to the Technical Specifications (TS) to increase the licensed reactor power level, submitted by letter dated May 16, 2002. Responses to the RAI are provided below.

NRC Question:

1. The licensee should provide a discussion of allowed outage time for the Leading Edge Flow Meter (LEFM) CheckPlus™ system. This discussion should include the following:

   a. The length of time the plant can be operated at a power level above 2300 MWt if the LEFM CheckPlus™ system becomes unavailable.

   b. The actions needed to be taken to continue operation above 2300 MWt after the LEFM CheckPlus™ system becomes unavailable (i.e., if the LEFM can be returned to operation prior to the expiration of the time limit in Question 1.a).

   c. Identification of the Technical Specification (TS) Surveillance Requirement that governs the time and actions involved in returning the LEFM CheckPlus™ system to operation.

   d. The actions needed to be taken to increase the power level above 2300 MWt after reduction of power because of an inoperable LEFM CheckPlus™ system.

   e. The impact on power level of a single spool piece (or single LEFM CheckPlus™ channel) versus multiple spool pieces (or multiple LEFM CheckPlus™ channels) being unavailable.

CP&L Response:

TS Surveillance Requirement (SR) 3.3.1.2 requires a comparison of the calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output every 24 hours. A proposed change to the Technical Requirements Manual (TRM) is being processed in support of the modification to install the Feedwater Ultrasonic Flow Measurement (FWUFM) system (the LEFM CheckPlus™ system). The proposed TRM change provides the conditions and required compensatory measures to be taken if the FWUFM is unavailable for the completion of SR 3.3.1.2.

The FWUFM system is considered available when the necessary flow measurement instrumentation and supporting hardware and software are available to provide feedwater mass flow data for each feedwater line. Therefore, if one or more spool pieces (i.e., one or more of the three LEFM channels) were unavailable, the entire FWUFM system would be considered unavailable. The proposed TRM change requires that the availability of the FWUFM system be verified using self-diagnostics prior to the performance of SR 3.3.1.2.

When the FWUFM system is available, the plant may operate in a manner consistent with the FWUFM-based secondary calorimetric calculation, at the uprated power level of 2339 MWt. When the FWUFM system is unavailable, the proposed TRM requirements would allow operation at the uprated power level until the next performance of SR 3.3.1.2. The maximum length of time the plant could be operated at a power level above 2300 MWt, if the FWUFM system becomes unavailable immediately after performance of SR 3.3.1.2, could be up to 30 hours, which is based on the SR 3.3.1.2 Frequency requirement of 24 hours and the SR 3.0.2 allowance that the specified Frequency is met if the SR is performed within 1.25 times the interval specified in the Frequency.

If the FWUFM system is unavailable at the time SR 3.3.1.2 is to be performed, alternate venturi-based flow measurement methods would be used for performance of SR 3.3.1.2. In this case, the proposed TRM requirements will limit the maximum allowable power level to 2300 MWt prior to the performance of SR 3.3.1.2, and the NIS would be adjusted to indicate 100% at 2300 MWt in accordance with SR 3.3.1.2 and the proposed TRM requirements. The TRM will require that thermal power be maintained ≤ 2300 MWt until the FWUFM system is restored to available status and SR 3.3.1.2 is performed using the FWUFM system.

The proposed TRM requirements do not contain specific actions for the loss of part of the FWUFM system (e.g., one of the three main feedwater line instruments). Therefore, the TRM compensatory measures for unavailability of the FWUFM are required for the partial failure of the FWUFM system, if that failure causes the FWUFM system to fail the self-diagnostics.

NRC Question:

2. The licensee should address the verification and validation of software.

CP&L Response:

The FWUFM system software quality assurance (SQA) activities are being conducted in accordance with the CP&L Nuclear Generation Group (NGG) SQA program procedures. Additionally, the FWUFM system software verification and validation will be documented in a software verification and validation report, in accordance with NGG SQA program procedures.

NRC Question:

3. In Attachment II, Section 3.2, page 12, the licensee stated that one spool piece will be installed in each of the feedwater headers that supply the steam generators. Robinson has two feedwater pumps and three steam generators. Therefore, it is not clear if two or three spool pieces will be installed. Please specify how many spool pieces are being installed.

CP&L Response:

A total of three spool pieces will be installed. One spool piece will be installed in the main feedwater line to each of the three steam generators.

NRC Question:

4. In Attachment II, Section 3.2, page 13, the licensee refers to the currently installed feedwater temperature instrumentation. Please indicate what is the currently installed feedwater temperature instrumentation.

CP&L Response:

The currently installed feedwater temperature instrumentation consists of 100-ohm platinum resistance temperature detectors (RTDs). These RTDs are located in thermowells in the piping downstream of the main feedwater regulating valves. These temperature instruments are shown on Updated Final Safety Analysis Report (UFSAR) Figure 10.1.0-5 as TE-3004, TE-3005, and TE-3006.

NRC Question:

5. In Attachment II, Section 3.2.1.3, page 16, the licensee references Caldon Engineering Report ER-267, "Bounding Uncertainty Analysis for Thermal Power Determination at CP&L Robinson Nuclear Power Station Using the LEFM Check Plus System," as the site-specific bounding uncertainty analysis that is the basis for the Robinson Uncertainty values. Please submit ER-267 to the staff.

CP&L Response:

Caldon Engineering Report ER-267, "Bounding Uncertainty Analysis for Thermal Power Determination at CP&L Robinson Nuclear Power Station Using the LEFM Check Plus System," Revision 0, is provided as Attachment V to this submittal. Caldon, Incorporated, considers information contained within ER-267 to be proprietary in nature and requests exemption from disclosure in accordance with 10 CFR 2.790(b). Attachment IV provides an affidavit and application for withholding from public disclosure executed by Mr. Calvin R. Hastings of Caldon, Incorporated, who is authorized to apply for the withholding of the proprietary information for Caldon, Incorporated. A non-proprietary version of ER-267 is supplied as Attachment III.

NRC Question:

6. In Attachment II, Section 3.10.1.2, page 57, the licensee describes the change in Allowable Value for Technical Specification 3.3.2 Engineered Safety Feature Actuation System Instrumentation, Table 3.3.2-1, Function 1.e, "Safety Injection – Steam Line High Differential Pressure Between Steam Header and Steam Lines." This Allowable Value is being changed from a number with an upper bound to a number with both upper and lower bounds. Please indicate the rationale for the change from a number with an upper bound to a number with both upper and lower bounds.

CP&L Response:

The proposed upper and lower limits are being specified to insure that the input to the applicable containment pressure and temperature analyses are bounded and maintained. The basis for the proposed limits is further described, as follows:

During a review of the calculations associated with the steam line high differential pressure, it was determined that the total loop uncertainty was not properly implemented in the setpoint value for the Safety Injection – Steam Line High Differential Pressure Between Steam Header and Steam Lines ESFAS trip assumed in the containment analysis. Westinghouse was contacted and requested to perform a sensitivity analysis using the revised values for the trip setpoint (100 ± 60 psi). The sensitivity analysis showed that for a Main Steam Line Break (MSLB) at hot zero power with an electrical bus failure, and for a MSLB at 102% power with a main feedwater regulating valve failure, a slightly higher containment pressure, on the order of 0.03 psig and 0.08 psig, respectively, resulted from an earlier ESFAS actuation.

According to Westinghouse, the greatest sensitivity to the steam line and steam header high differential pressure signal lies with the auxiliary feedwater start time. Since steam line breaks are sensitive to main feedwater and auxiliary feedwater delivery rates and times, it is clear that auxiliary feedwater start time would have an effect. Due to the conclusions of the sensitivity analysis, the containment analysis has been revised to assume a bounding actuation setpoint of 100 ± 60 psi. The proposed TS upper bound of ≤ 116.24 psig and lower bound of ≥ 83.76 psig are bounded by the containment analysis assumptions.

NRC Question:

7. Identify the LOCA and non-LOCA transient analyses of record that rely on the following isolation valves, reactor trip, and Engineered Safety Features (ESF) for consequence mitigation. Provide the values for the valves isolation time, reactor trip, and ESF actuation system setpoints assumed in the applicable analyses and justify that the current analyses bound the cases with the revised values below.

   (a) Page 37* indicates that the stroke time requirements specified in Technical Specification (TS) 3.7.3.1 for the main feedwater regulation valves and the main feedwater bypass valves are changed from ≤ 30 seconds to ≤ 20 seconds.

   (b) Page 53* indicates that the values of T_{avg} in the reactor trip equations for the OTDT trip (TS Table 3.3.1-1, Note 1) and the OPDT trip (TS Table 3.3.1-1, Note 2) are changed from 575.4°F to 575.9°F.

   (c) Page 57* indicates that the allowable value in TS Table 3.3.2-1, Function 1.e for the "Steam Line High Differential Pressure Between Steam Header and Steam Lines – Safety Injection" is revised from ≤ 108.95 psig to an upper bound of ≤ 116.24 psig with an added lower bound of ≥ 83.76 psig.

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* The page numbers in the RAI refer to Attachment II to the licensee's letter dated May 16, 2002.

Also, page 55* states that "it will be necessary to revise calculations for the Steam/Feedwater Flow Mismatch Trip to reflect the new nominal flow rates for feedwater and steam flow. The nominal trip setpoint is provided in TS Table 3.3.1-1." Based on the statement, it appears to the staff that the trip setpoint needs to be changed. However, TS Table 3.3.1-1 is not changed to reflect the new trip setpoint. Provide clarification to the statement and the TS changes related to the Steam/Feedwater Flow Mismatch Trip.

CP&L Response:

The Main Feedwater (MFW) regulating valve bypass valves are not modeled in the Chapter 15 safety analyses. Therefore, changing the Technical Specifications valve stroke time limit will not impact the analyses of record.

The ESFAS signal on "Steam Line High Differential Pressure Between Steam Header and Steam Lines -- Safety Injection" is not modeled in the Chapter 15 safety analyses. Therefore, changing the Technical Specifications allowable setpoint values will not impact the analyses of record.

The following table provides the requested information for the non-LOCA analyses that do not involve containment mass and energy release.

UFSAR Chapter 15 Event	MFW Regulating Valve Stroke Time	T-Average (T_{avg}) Value Used in OTΔT and OPΔT Trip Equations	Disposition of Change
Main Steam Line Break (15.1.5)	30 seconds	OTΔT and OPΔT Trips were not used to mitigate event.	MFW is modeled to isolate 30 seconds after receiving the isolation signal. Prior to isolation, all MFW is conservatively modeled to be delivered to the affected steam generator. The modeling of the MFW response bounds a lower MFW regulating valve stroke time of 20 seconds because the cooling of the primary system is maximized. Therefore, the current licensing basis analysis remains bounding.
Loss of External Load (15.2.2)	1 second	575.4°F	The current licensing basis analysis conservatively assumes a nearly instantaneous MFW isolation (i.e., ≤ 1 second) coincident with turbine trip. Changing the TS MFW regulating valve stroke time limit from 30 to 20 seconds will not adversely impact the analysis of record. As a result of the Appendix K power uprate, the vessel T_{avg} at rated power will be 575.9°F. Changing the reference T_{avg} for the OTΔT and OPΔT trips from 575.4°F to 575.9°F results in the timing of the reactor trip in the Chapter 15 event analyses being insignificantly impacted by the power uprate. In addition, the analysis of record bounds the effect of the slight increase in Tavg on the minimum Departure from Nucleate Boiling (DNB) ratio. Therefore, the analysis of record remains bounding.

UFSAR Chapter 15 Event	MFW Regulating Valve Stroke Time	T-Average (T_{avg}) Value Used in OTΔT and OPΔT Trip Equations	Disposition of Change
Loss of Normal Feedwater (15.2.7)	1 second	OTΔT and OPΔT Trips were not used to mitigate event.	The analysis of record conservatively assumes a nearly instantaneous MFW isolation coincident with turbine trip. Changing the TS MFW regulating valve stroke time limit from 30 to 20 seconds will not adversely impact the analysis of record.
Bank Withdrawal at Power (15.4.2)	Not used to mitigate event	575.4°F	As a result of the Appendix K power uprate, the vessel T_{avg} at rated power will be 575.9°F. Changing the reference T_{avg} for the OTΔT and OPΔT trips from 575.4°F to 575.9°F results in the timing of the reactor trip in the Chapter 15 event analyses being insignificantly impacted by the power uprate. In addition, the analysis of record bounds the effect of the slight increase in Tavg on minimum DNB ratio. Therefore, the analysis of record remains bounding.

The following table provides the requested information pertaining to the containment mass and energy analyses:

UFSAR Chapter 6 (Containment Evaluation)	MFW Regulating Valve Stroke Time	MFW Bypass Valve Stroke Time	T-Average Value used in OTΔT and OPΔT Trip Equations	Steam Line High Differential Pressure – Safety Injection	Steam / Feedwater Flow Mismatch Trip
Containment LOCA Mass and Energy Calculation	≤ 20 seconds	≤ 20 seconds	OTΔT and OPΔT Trips were not used to mitigate event.	Nominal: 100 psi Upper: 160 psi Lower: 40 psi	Not used in containment evaluation
Containment MSLB Mass and Energy Calculation	≤ 20 seconds	≤ 20 seconds	OTΔT and OPΔT Trips were not used to mitigate event.	Nominal: 100 psi Upper: 160 psi Lower: 40 psi	Not used in containment evaluation

The Steam Flow/Feedwater Flow Mismatch trip does not need to be revised. The statement on page 55 that "it will be necessary to revise calculations for the Steam/Feedwater Flow Mismatch Trip to reflect the new nominal flow rates for feedwater and steam flow" is based on the increase in full power steam flow and feed flow rates, as a result of the Measurement Uncertainty Recapture (MUR) power uprate, to approximately 3.430E6 lbm/hr per steam generator.

The Steam Flow/Feedwater Flow Mismatch trip generates a reactor trip when it is coincident with a Steam Generator Low Level trip. The current Technical Specification Nominal Trip

Setpoint (NTS) is 6.4E5 lbm/hr and the Allowable Value (AV) is ≤ 7.06E5 lbm/hr. UFSAR Table 15.0.7-1 shows that this reactor trip is not directly credited to mitigate any UFSAR Chapter 15 events. This trip is also not used in the containment evaluation; therefore, there is no "analysis value."

This trip serves as an anticipatory trip for the Steam Generator Low-Low Water Level trip. The trip setpoint is based on less than or equal to a mismatch equivalent to 40% of full power steam flow per steam generator. The full power steam flow will increase, as a result of the MUR power uprate, to approximately 3.430E6 lbm/hr per steam generator. Therefore, the flow rate equivalent to ≤ 40% full power steam flow under MUR power uprate conditions would be ≤ 1.372E6 lbm/hr, compared to the current value of ≤ 1.348E6 lbm/hr. The current full power steam flow is used to calculate the current design input value of the mismatch, and this portion of the CP&L calculation is being revised, as necessary, to incorporate the MUR power uprate changes to steam flow, steam generator pressure, and feedwater temperature. The MUR power uprate will not alter the calibration or drift characteristics of the equipment. No changes to the TS NTS of 6.4E5 lbm/hr or AV of ≤ 7.06E5 lbm/hr are considered necessary, because the TS values provide sufficient margin for this anticipatory trip function.

NRC Question:

8. Provide results of an ATWS analysis demonstrating that the plant at power uprate conditions is within the bounds considered by the staff during the licensee's documentation of compliance with the ATWS rule. If the licensee chooses to rely on the Westinghouse generic ATWS analyses to demonstrate the acceptance of the analytical results, the licensee is requested to provide a discussion of the ATWS analyses that are applicable to the specific plant and power uprate conditions, and justify that the assumptions for the applicable ATWS analyses are adequate as they relate to input parameters such as the initial power level, Moderator Temperature Coefficient (MTC), pressurizer safety and relief valves capacity, RCS volume, steam generator pressure, Auxiliary Feedwater (AFW) flow rate and its actuation delay time, and the Setpoint for the AMSAC system to actuate the AFW and trip the turbine. The submittal should include a discussion and applicable values of the unfavorable exposure time for the MTC assumed in the analyses.

CP&L Response:

No new Anticipated Transient Without Scram (ATWS) analyses were performed to support power uprate. The generic analyses were evaluated based on the HBRSEP, Unit No. 2, power uprate conditions. The basis for the ATWS Mitigating System Actuation Circuitry (AMSAC) is provided in WCAP-10858-P-A, "AMSAC Generic Design Package." Revision 1 of this report was approved for application to HBRSEP, Unit No. 2, via the NRC Safety Evaluation Report (SER) for "H.B. Robinson Steam Electric Plant, Unit No. 2 – ATWS Rule," dated October 14, 1988.

The 14 key elements, identified in the NRC SER, remain unaffected by the plant changes associated with this license amendment request.

As indicated in the NRC SER, HBRSEP, Unit No. 2, "has elected to implement the WCAP-10858-P-A AMSAC design associated with monitoring the steam generator water level and activating the AMSAC when the water level is below the low-low setpoint established for the reactor protection system (RPS)." Section 3.0 of WCAP-10858-P-A describes the functional requirements for AMSAC actuation on low steam generator water level. The following is an evaluation of the effects on power uprate on the functional requirements identified in Section 3.0 of WCAP-10858-P-A:

1) "...in order to minimize the amount of reactor coolant system voiding during an ATWS, AMSAC should operate at and above 40% of nominal power" and "the AMSAC signal shall be automatically blocked below 40% power."

   The C-20 permissive setpoint is currently set to $\leq$ 40% of 2300 MWt and will be changed to reflect $\leq$ 40% of 2339 MWt.

2) "...the C-20 permissive signal shall be maintained for approximately 360 seconds after a turbine trip has occurred" and "removal of the C-20 permissive signal will be delayed by approximately 360 seconds."

   The proposed increase in authorized reactor power level does not affect this delay parameter, therefore, this delay is not being changed.

3) "The turbine trip response time from an AMSAC signal must be less than or equal to 30 seconds, including sensor delays."

   The circuitry used to generate a turbine trip is not being altered by the proposed increase in authorized reactor power level. Therefore, there will be no affect on the turbine trip response time.

4) "The auxiliary feedwater flow response time from an AMSAC signal must be less than or equal to 90 seconds, including sensor delays."

   The circuitry to initiate auxiliary feedwater flow is not being altered by the proposed increase in authorized reactor power level. Therefore, there will be no affect on the auxiliary feedwater response time.

5) "AMSAC actuation is required at a setpoint below the RPS steam generator lo-lo level setpoint" and "the AMSAC setpoint shall be no more than 5% of narrow range span (NRS) below the RPS steam generator level setpoint. In all cases, the AMSAC setpoint shall never be less than 5% of NRS."

   The RPS Low-Low Steam Generator level setpoint is set at 16% and the AMSAC Steam Generator level setpoint is set at 11%. Neither of these setpoints is being changed as a result of proposed increase in authorized reactor power level.

Section 3.0 of WCAP-10858-P-A does not link the specific power rating of the plant to the AMSAC setpoints, delays, or time responses.

[Note: As requested by the RAI, the following information is a description of the evaluation of parameters associated with the generic analyses. As evaluated in the preceding information, HBRSEP, Unit No. 2, utilizes the general approach described in WCAP-108580-P-A. The design parameters and settings for AMSAC provided in WCAP-10858-P-A are considered bounding and evaluation of the generic analysis is not necessary to verify compliance with 10 CFR 50.62. The following information is provided only in response to the RAI and does not constitute new licensing basis requirements for HBRSEP, Unit No. 2.]

The generic Westinghouse ATWS analyses are contained in WCAP-8404, "Anticipated Transient Without Trip Analysis for Westinghouse PWRs with 44 Series Steam Generators." WCAP-8404 used the same analysis methodologies as established in WCAP-8330, "Westinghouse Anticipated Transients with Trip Analysis." Comparison of relevant corresponding HBRSEP, Unit No. 2, data to the 3-loop plant used in the generic analysis follows:

Parameter	**WCAP-8404 Generic 3 Loop Plant**	**HBRSEP, Unit No. 2, Value at Current Power Level (2300 MWt)**	**HBRSEP, Unit No. 2, Value at Uprated Power Level (2339 MWt)**
Core Power	2300 MWt	2300 MWt	2339 MWt
Reactor Coolant System (RCS) Volume	9072 ft^3	9343 ft^3	9343 ft^3
Pressurizer Safety Valve Capacity	288,000 lbm/hr	293,330 lbm/hr	293,330 lbm/hr
Pressurizer PORV Capacity	179,000 lbm/hr	210,000 lbm/hr	210,000 lbm/hr
Steam Generator Pressure	833 psia	824 psia	821 psia
AFW Flowrate	1200 gpm	1209 gpm	1209 gpm
AFW Actuation Delay	60-180 sec	$\leq$90 sec	$\leq$90 sec
Turbine Trip	30 sec	$\leq$30 sec	$\leq$30 sec
MTC at 100% power	$\leq$ -8 pcm/°F for $\geq$ 95% of the cycle	$\leq$ -8 pcm/°F for 100% of the cycle	$\leq$ -8 pcm/°F for $\geq$ 99% of the cycle

In WCAP-8404, the most limiting ATWS event for a 3 loop plant was the Loss of Feedwater without a Reactor Trip. The peak RCS pressure for the Generic 3 Loop Plant during this event was 2887 psia.

WCAP-8404 provides sensitivity studies for core power level for this event as follows:

Core Power	Peak RCS Pressure
100% (2300 MWt)	2887 psia
90% (2070 MWt)	2735 psia
80% (1840 MWt)	2636 psia

For a 230 MWt decrease in core power (100% to 90%), the peak RCS pressure decreased 152 psi or 0.661 psi/MWt. A similar 230 MWt decrease in core power from 90% to 80% only produced a 99 psi decrease or 0.430 psi/MWt. Using the larger 0.661 psi/MWt value, a 39 MWt increase would cause an approximately 26 psi increase in peak pressure resulting in a peak RCS pressure of approximately 2913 psia.

WCAP-8404 also provides sensitivity studies for pressurizer PORV and safety valve relief capacity for this event as follows:

Pressurizer PORV Capacity	Pressurizer Safety Capacity	Peak RCS Pressure
0	864,000 lbm/hr	3247 psia
358,000 lbm/hr	864,000 lbm/hr	2887 psia
420,000 lbm/hr	864,000 lbm/hr	2824 psia

The increase in the pressurizer PORV capacity from 0 to 358,000 lbm/hr (2 PORVs x 179,000 lbm/hr) caused a 360 psi decrease in the peak RCS pressure (0.001 psi/lbm/hr). The further increase from 358,000 lbm/hr to 420,000 lbm/hr (2 PORVs x 210,000 lbm/hr) produced an additional 63 psi decrease in the peak RCS pressure (0.001 psi/lbm/hr). As noted in the preceding table, HBRSEP, Unit No. 2, has a total pressurizer PORV capacity of 420,000 lbm/hr; therefore, the 26 psi increase in peak RCS pressure caused by the small increase in core power level would be more than offset by the 63 psi reduction due to the larger HBRSEP, Unit No. 2, PORVs. Furthermore, the HBRSEP, Unit No. 2, pressurizer safety valves are larger than those modeled in the Generic 3 Loop Plant analysis, which would provide an additional reduction in the peak RCS pressure.

HBRSEP, Unit No. 2, is not requesting a change to the Technical Specification limits on the allowable magnitude of a positive MTC. The power uprate is not expected to result in any significant change in the nominal value of the MTC. Therefore, following power uprate the MTC will be more negative than the analysis value for $\geq$ 95% of the time as specified for the generic Unfavorable Exposure Time.

Based on the MTC being maintained within the bounds of the generic analyses, and the additional pressurizer PORV and safety valve capacity over that assumed in the generic analyses (offsets the increase in core power level), the generic Westinghouse analyses remain bounding and show significant margin to the reactor coolant system pressure limit of 3200 psia.

NRC Question:

9. Table 3.2-1 on page 83* lists uncertainties for six components used to calculate the total secondary calorimeter power measurement uncertainty. The six component uncertainties are represented by: A for the feedwater mass flow (LEFM), B for feedwater temperature, C for main steam pressure, D for main steam pressure (SG blowdown), E for blowdown flow, and F for CPC factor. Provide documents to show how each of the six component uncertainties (items A through F) are determined and explain the differences of uncertainties contributed from items C, D, and E.

   The staff finds that in the equation used to calculate the total power measurement uncertainty, item A is not included and item H is not defined. It is also not clear why the items for the squares of C, D, E, and F in the equation are multiplied by 3, and item B is not multiplied by 3 even though the licensee states that items B through E represents instrument uncertainties for each RCS loop (in a total of 3 loops) while F represents uncertainties from various heat gain and loss. The licensee is requested to clearly define each item used in the equation and confirm the use of multiplier of 3 (instead of a divider of 3) for each item is correct and acceptable.

CP&L Response:

As noted by this question, there is an error in the original submittal relative to the description of the equation that is used to demonstrate the calculation of the two-sigma secondary calorimetric power measurement uncertainty. The equation below Table 3.2-1 on page 83 is restated as follows:

$$\text{Uncertainty}_{\text{LEFM}} = [(B)^2+(C)^2\times3+(D)^2\times3+(E)^2\times3+(F)^2\times3+(H)^2]^{1/2}$$

This equation is corrected as follows:

$$\text{Uncertainty}_{\text{LEFM}} = [(A)^2+(B)^2\times3+(C)^2\times3+(D)^2\times3+(E)^2\times3+(F)^2]^{1/2}$$

The parameters and uncertainties provided in Table 3.2-1 remain valid, including item G, the Total Uncertainty, which was calculated in accordance with the corrected version of the uncertainty equation.

The determination of the six uncertainty terms (A through F), listed above, are described in plant-specific calculations and are summarized as follows:

A. Feedwater Mass Flow Rate Uncertainty is from Caldon, Inc., Engineering Report ER-267, Revision 0, "Bounding Uncertainty Analysis for Thermal Power Determination at CP&L Robinson Nuclear Power Station Using the LEFM Check Plus System."

B. Feedwater Temperature Uncertainty is from Caldon, Inc., Engineering Report ER-267, Revision 0, "Bounding Uncertainty Analysis for Thermal Power

* The page numbers refer to Attachment II of the CP&L letter dated May 16, 2002.

Determination at CP&L Robinson Nuclear Power Station Using the LEFM Check Plus System."

C. Main Steam Pressure Uncertainty is the uncertainty impact due to main steam pressure measurement instrumentation accuracy.

D. Main Steam Pressure (SG Blowdown) is the uncertainty impact due to changes in main steam pressure on SG Blowdown enthalpy.

E. Blowdown Flow is the uncertainty impact due to SG Blowdown flow measurement instrumentation accuracy.

F. CPC Factor is the Core Power Correction (CPC) factor, which is the impact associated with system heat inputs and losses that are separate from the system heat input due to the reactor core thermal power level.

The explanation of differences in uncertainties contributed from items C, D, and E is provided as follows:

Parameter C – Main Steam Pressure Uncertainty defines a band of pressures around a nominal base steam pressure. This defines the range of enthalpy to be considered.

Parameter D – Main Steam Pressure (SG Blowdown) is related to main steam pressure as SG Blowdown pressure is assumed to be less than main steam pressure. Using the lower pressure value and the 97.5%/2.5% mixture of water/steam, a band of blowdown enthalpies is defined for consideration in the determination of secondary thermal power.

Parameter E - Blowdown Flow Uncertainty defines a range of blowdown flow values to be considered in the determination of secondary thermal power.

In the equation used to calculate the total power measurement uncertainty, the squares of parameters B, C, D, and E are multiplied by 3 because each parameter is an equal single loop uncertainty value. Therefore, it is appropriate to multiply these values in this manner.

Parameter A, Mass Flow Rate Feedwater Uncertainty, is a 3 loop total flow uncertainty. Therefore, multiplication by 3 is not necessary for this parameter, because HBRSEP, Unit No. 2, has three reactor coolant loops.

Parameter F, CPC Factor, is a total uncertainty associated with the correlation of core power to the secondary heat balance. Therefore, multiplication by 3 is not necessary for this parameter.

NRC Question:

10. Westinghouse has issued three Nuclear Service Advisory Letter (NSAL), NSAL -02-3 and revision 1, 02-4 and 02-5, to document the problems with the Westinghouse-designed steam generator (SG) water level setpoint uncertainties. NSAL-02-3 and its revision, issued on February 15 and April 8, 2002, respectively, deal with the uncertainties caused by the mid-

deck plate located between the upper and lower taps used for SG measurements and affect the low-low level trip setpoint (used in the analyses for events such as the feedwater line break, ATWS and steam line break). NSAL-02-4, issued on February 19, 2002, deals with the uncertainties created because the void contents of the two-phase mixture above the mid-deck plate were not reflected in the calculations and affects the high-high level trip setpoint. NSAL-02-5, issued on February 19, 2002, deals with the initial conditions assumed in the SG water level related safety analyses. The analyses may not be bounding because of velocity head effects or mid-deck plate pressure differential pressure that increase in the control system uncertainties. Discuss how Robinson, Unit 2 accounts for all these uncertainties documented in these advisory letters in determining the SG water level setpoints. Also, discuss the effects of the water level uncertainties on the analyses of record for the LOCA and non-LOCA transients and the ATWS event, and verify that with consideration of all the water level uncertainties, the current analyses are still limiting.

CP&L Response:

The condition of concern, as identified in NSAL-02-3, is based on the discovery of a pressure drop at the mid-deck plate at the top of the steam generator primary moisture separator assembly that could cause the steam generator level indication to be higher than the actual level. The NSAL-02-3 and Chapter 15 of the HBRSEP, Unit No. 2, UFSAR state that the Steam Generator Low-Low Level Reactor Trip is used for mitigation of design basis Loss of Off-Site Power (LOOP), Loss of Normal Feedwater (LONF), and Feedwater Line Break accidents. The effects identified in NSAL-02-3 were incorporated into the Total Loop Uncertainty calculation for Steam Generator Low-Low Water Level under normal containment conditions (i.e., containment pressure and temperature within the limits specified in HBRSEP, Unit No. 2, Technical Specifications Limiting Condition for Operation [LCO] 3.6.4 and LCO 3.6.5). This calculation concluded that the Steam Generator Low-Low Level Reactor Trip Allowable Value and Nominal Trip Setpoint, as listed in HBRSEP, Unit No. 2, Technical Specifications LCO 3.3.1, Function 13, remain valid due to sufficient margin in the Total Loop Uncertainty calculation to accommodate the condition of concern in NSAL-02-3. The evaluation of NSAL-02-3 concluded that the Steam Generator Low-Low Level Reactor Trip function remains operable. Additional evaluations will be performed, as needed, to verify that the issues identified in NSAL -02-3 properly account for uprated power conditions.

The condition of concern, as identified in NSAL-02-4, is based on the discovery that the void content of the two-phase mixture above the mid-deck plate may cause additional inaccuracy in the steam generator water level instrumentation for the actuation of the high water level control function. The effects identified in NSAL-02-4 were incorporated into the Total Loop Uncertainty calculation for the Steam Generator High-High Water Level control function. This calculation concluded that the Steam Generator High-High Water Level Feedwater Isolation setpoint should be reduced from 75% to 74%. The Steam Generator High-High Water Level Feedwater Isolation function is not used for mitigation of any design basis accidents as shown in Chapter 15 of the HBRSEP, Unit No. 2, UFSAR. This function is used to prevent steam generator overfill in the event of a failure in the steam generator level control system. An engineering change has been initiated to implement the setpoint change identified in the evaluation of NSAL-02-4. Additional evaluations will be performed, as needed, to verify that the issues identified in NSAL -02-4 properly account for uprated power conditions.

NSAL-02-5 and Revision 1 to NSAL-02-3 are being evaluated in accordance with the Progress Energy and CP&L Nuclear Generation Group (NGG) procedures that establish the methods for evaluation of operating experience (OE) of this type (i.e., vendor technical information). These evaluations will properly account for the identified issues, as applicable to HBRSEP, Unit No. 2, and will accommodate the uprated power conditions.

If the evaluations of the aforementioned Westinghouse Nuclear Safety Advisory Letters conclude that further plant changes are required, these changes will be processed in accordance with the applicable NGG and HBRSEP, Unit No. 2, procedures.

NRC Question:

11. The licensee is requested to discuss the methodology used in the calculationof the current vessel pressure-temperature curves (page 27*) and confirm that it adheres to the guidance in RG 1.190. Also, provide the results of calculations to show the change of the expected end-of-license (EOL) value for RT_{PTS} from the current power level to the proposed power uprate conditions.

CP&L Response:

The NRC issued Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods For Determining Pressure Vessel Neutron Fluence," in March of 2001. The current HBRSEP, Unit No. 2, pressure-temperature (PT) curves were developed in 1990, utilizing surveillance capsule data, and were approved in a Safety Evaluation associated with License Amendment No. 149, dated July 29, 1994. As discussed in the May 16, 2002, submittal, CP&L requested that the PT limit curves be re-designated from 24 effective full power years (EFPY) of operation to 23.96 EFPY to reflect a conservative projection of the increase in neutron fluence associated with the power uprate. CP&L predicts that the 23.96 EFPY operational limit will occur in January 2005 during Cycle 23. New PT curves, using RG 1.190 methodologies and including recently analyzed surveillance capsule data, are being developed for submittal to the NRC by April of 2004.

Comparisons of the methodology used in developing the current PT curves with the RG 1.190 methodology indicate that the fluence levels either decrease or stay the same. For example, the 29 EFPY projected fluence at the vessel clad-base interface for plate midplane at 0 degrees for the current curve is 4.55E+19 n/cm². The RG 1.190 methodology results in a projected fluence for this location of 3.67E+19 n/cm².

The current RT_{PTS} for the limiting material, which is the upper circumferential weld, was calculated based on a conservative fluence of 1.8E+19 n/cm², as compared to the projected fluence of 1.57E+19 n/cm² for that location, based on the current methodology. The RG 1.190 methodology also results in a projected EOL fluence of 1.57E+19 n/cm².

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* The page numbers in the RAI refer to Attachment II to the licensee's letter dated May 16, 2002.

NRC Question:

12. The definition of dose equivalent I-131 should be defined using the effective dose conversion factor for I-131 inhalation taken from Table 2.1, Federal Guidance Report 11, and not the thyroid dose conversion factor taken from NUREG/CR-6604. Please explain.

CP&L Response:

The proposed definition will be revised in accordance with the reviewer comment in a supplement to the proposed license amendment request for full implementation of the alternate source term (CP&L letter dated May 10, 2002).

NRC Question:

13. The appropriate dose limits for the release of the contents of the waste gas decay tank are the radiation dose limits for individual members of the public. When using TEDE criteria, this is 0.1 rem TEDE and not 0.5 rem TEDE as proposed by the licensee. As a side note to this particular aspect, the calculation of the maximum allowable curie content of dose equivalent Xe133 in a waste gas decay tank should be based upon the Xe 133 effective dose conversion factor for air submersion taken from Federal Guidance Report 12.

CP&L Response:

CP&L requests to defer response to this question to the supplement to the proposed license amendment request for full implementation of the alternate source term (CP&L letter dated May 10, 2002).

NRC Question:

14. Provide fission product inventory in the fuel rod gap for each radionuclide of interest (noble gases and halogens) that is available for release after 8 hours and 56 hours after reactor shutdown to the water surrounding the failed fuel assembly. Also, provide assumed amounts of fission product activities (in curies) release to the environment from the containment and from the fuel handling building following the postulated design-basis accident.

CP&L Response:

As discussed in a telephone conversation with the HBRSEP, Unit No. 2, NRC project manager on July 8, 2002, this information is not required.

[The following RAI questions were provided in NRC letter dated July 17, 2002.]

NRC Question:

1. Section 3.6.2 provides an assessment of reactor coolant system components for the uprated power conditions. The reactor pressure vessel was not explicitly addressed. The licensee is requested to provide information (i.e. existing minimum margin in stress and CUF to show

that the existing margins for the reactor vessel will accommodate the RCS temperature change or that the current design/operating temperature range is bounding for the 1.7% uprate condition.

CP&L Response:

The original RCS design conditions for T_{cold} and T_{hot} temperatures, as listed in Table 3.6-1, show that the reactor vessel equipment specification bounds the uprated operating conditions.

NRC Question:

2. Section 3.6.2.1 states that, "the values for T_{hot} and T_{cold} under uprated power conditions are bounded by the revised RCS design condition values as shown in Table 3.6-1." Temperature data in Table 3.6-1 indicates that the T_{hot} for the revised RCS design condition (604.6°F) is slightly higher than that of the uprated operating conditions (604.1°F and 604.5°F). However, T_{cold} for the revised RCS design conditions (546.1°F) is slightly lower than that of the uprated operating conditions (547.6°F or 547.3°F). On the basis of the cited temperatures from Table 3-6.1, provide your justification for the above quoted statement.

CP&L Response:

The statement should more accurately read, "the values for T_{hot} and T_{cold} under uprated power conditions are bounded by the combination of the original and revised design condition values as shown in Table 3.6-1." The original design conditions were those used in the original design and evaluation of plant components. The revised design conditions reflect the plant operating conditions at 2300 MWt with minimum RCS flow as reflected by the replacement steam generator equipment specification. The combination of the original design conditions and the revised design conditions create an envelope that bounds the T_{cold} conditions for power uprate (T_{cold} original > T_{cold} uprated conditions > T_{cold} revised). For T_{hot}, the effects of thermal loads on the structural impact are bounded, since both the original and revised T_{hot} are greater than the uprated T_{hot}. Therefore, thermal expansion loads and thermal stresses for the uprated conditions are bounded by existing analyses, and the uprated power conditions will not adversely affect RCS component stress and fatigue.

NRC Question:

3. Section 3.6.2.8 concluded that PWSCC of the CRDM nozzles will not be impacted by the slight increase in RCS T_{hot} resulting from the power uprate. It further stated that an inspection of the CRDM nozzles is planned for the next refueling outage (RO-21) scheduled to begin in October 2002. Please provide your clarification regarding the significance of the planned inspections as related to its impact on the Technical Specification amendment request for power uprate.

CP&L Response:

Section 3.6.2.8 of the amendment request provides a conclusion pertaining to the effects of the higher RCS temperature on the issues identified in NRC Bulletins 2001-01 and 2002-01. The statement pertaining to the planned inspection activities for the upcoming refueling outage (RO-21) is a reiteration of commitments previously made in response to NRC Bulletins 2001-01 and 2002-01. The statements in Section 3.6.2.8 are clarified as follows:

> The details pertaining to reactor vessel head inspection activities are provided in the HBRSEP, Unit No. 2, responses to NRC Bulletins 2001-01 and 2002-01. Based on the insignificant impact of the slight increase in RCS T_{hot} resulting from the power uprate, these inspection activities are not expected to be affected by the change in reactor operating conditions proposed in this license amendment request.

NRC Question:

4. Per the guidance provided in NRC Regulatory Issue Summary 2002-03: Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," Attachment 1, Section IV.C., provide a response which addresses any effects of the propose power uprate on:

   (1) The pressurized thermal shock evaluation for the H. B. Robinson reactor pressure vessel materials, which should demonstrate continued compliance with the requirements in Title 10 of the Code of Federal Regulations Section 50.61.

   (2) The upper shelf energy evaluation for the H. B. Robinson reactor pressure vessel materials, which should demonstrate continued compliance with the requirements in Title 10 of the Code of Federal Regulations Part 50, Appendix G.

   If the proposed power uprate has no effect on these evaluations, please state so in your response.

CP&L Response to (1):

The fluences for the HBRSEP, Unit No. 2, vessel beltline materials for the current and the projected uprate operation are summarized in the following table. The current EOL fluences are docketed and referenced in Reactor Vessel Integrity Database (RVID), and the uprate fluences were projected using ENDF/BVI cross-sections and comply with Reg. Guide 1.190.

Material	EOL Fluence (n/cm^2) (Current)	EOL Fluence (n/cm^2) (Uprated Operation)
Intermediate Shell Plates	4.80×10^{19}	3.67×10^{19}
Upper Circumferential Weld (W5214) & Upper Shell Plates	1.80×10^{19}	1.57×10^{19}
Lower Circumferential Weld (34B009) & Lower Shell Plates	2.00×10^{19}	1.67×10^{19}
Axial Welds (Heat 86054B)	3.93×10^{19}	2.73×10^{19}

Since there are no changes to the material chemistries or the initial $RT_{NDT(U)}$, the currently docketed PTS values for each of the vessel materials remains bounding in accordance with 10 CFR 50.61. The proposed power uprate has no effect on the vessel PTS evaluation.

CP&L Response to (2):

The upper shelf energy (USE) values are determined by reducing the unirradiated USE by an amount of predicted decrease as a function of fluence and copper content per Regulatory Guide 1.99, Revision 2, Position 2.2. There is no change in the chemistries, and the current fluences bound the predicted uprate fluences as shown in the response to Question (1). Therefore, the current USE values are conservative and power uprate operation has no effect on the USE evaluation.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

CALDON, INC.
ENGINEERING REPORT: ER-267N,
"BOUNDING UNCERTAINTY ANALYSIS FOR THERMAL POWER DETERMINATION AT CP&L ROBINSON NUCLEAR POWER STATION USING THE LEFM CHECK PLUS SYSTEM,"
REVISION 0,
NON-PROPRIETARY VERSION

CALDON

ER-267N
REVISION 0
JANUARY 2002

CALDON, INC.
ENGINEERING REPORT: ER-267N

BOUNDING UNCERTAINTY ANALYSIS FOR THERMAL POWER DETERMINATION AT CP&L ROBINSON NUCLEAR POWER STATION USING THE LEFM✓ + SYSTEM

Prepared By: Ed Madera
Reviewed By: Don Augenstein

NON-PROPRIETARY VERSION

Engineering Report: ER-267N Revision 0

BOUNDING UNCERTAINTY ANALYSIS FOR THERMAL POWER DETERMINATION AT CP&L ROBINSON NUCLEAR POWER STATION USING THE LEFM✓ + SYSTEM

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APPENDICES

Engineering Report: ER-267N

BOUNDING UNCERTAINTY ANALYSIS FOR THERMAL POWER DETERMINATION AT CP&L ROBINSON NUCLEAR POWER STATION USING THE LEFM✓ + SYSTEM

1.0 INTRODUCTION

The LEFM✓ and LEFM✓ + are advanced ultrasonic systems that accurately determine the mass flow and temperature of feedwater in nuclear power plants. Using feedwater pressure signal input to the LEFM✓ and LEFM✓ +; its mass flow and temperature outputs are used along with other plant data to compute reactor core thermal power. The technology underlying the LEFM✓ ultrasonic instruments and the factors affecting their performance are described in a topical report, Reference 1, and a supplement to this topical report, Reference 2. The LEFM✓ + is described in another supplement to the topical report, Reference 3. This supplement demonstrates that plants using the LEFM✓ + can increase their licensed thermal power rating by as much as 1.7%. The exact amount of the uprate allowable under a recent revision to 10CFR50 Appendix K depends not only on the accuracy of the LEFM✓ + outputs but also on the uncertainties in other inputs to the thermal power calculation.

It is the purpose of this document to provide an analysis of the uncertainty contribution of the LEFM✓ + to the overall thermal power uncertainty of CP&L Robinson Nuclear Power Station. The total uncertainty contribution is documented in the Interface and Reconciliation document (included with this document as Appendix C) as the term AB. This uncertainty is the aggregate power uncertainty due to the feedwater flow, temperature and pressure measurements.

The uncertainties in mass flow and feedwater temperature are also provided. If desired, these data may also be used in the calculation of the overall thermal power uncertainty. If this method is used, a special procedure is required for combining the mass flow uncertainty and the uncertainty in feedwater enthalpy due to temperature. The procedure is necessary because some elements of the temperature uncertainty are systematically related to elements of the mass flow uncertainty and others are not. Instructions and constants necessary for the use of this procedure are provided.

Revision 0 of this analysis is a preliminary bounding analysis for the CP&L Robinson plant. This revision utilizes nominal dimensions for the spool piece(s) and current values for full power mass flow, final feed temperature and steam conditions. Bounding values for the uncertainties in length measurements, time measurements and calibration coefficient (profile factor) are employed.

Revision 1 of this uncertainty analysis will be published following the calibration of the spool piece(s), when a precise estimate of the uncertainty in the profile factor(s) is available. In Revision 1, the as-built dimensions are input for all computations, and it is confirmed that the uncertainties in these dimensions lie within the bounding values used in the Revision 0 analysis. The analysis may use either the bounding dimensional uncertainties or the actual uncertainties, which ever is greater. In Revision 1, bounding values for the uncertainties in time measurements are again used. The commissioning tests for the LEFM✓ +, performed following its installation in the plant, confirm that in fact time measurement uncertainties are within the bounding values used in the Revision 1 analysis.

2.0 SUMMARY

For the CP&L Robinson Nuclear Power Station, Revision 0 results are as follows:

1. The uncertainty in the mass flow of feedwater is [ ].

2. The uncertainty in the final feedwater temperature is less than 0.6°F.

3. The total power uncertainty due to the LEFM✓+ is [ ].

4. For an overall thermal power uncertainty analysis in which mass flow and temperature errors are treated separately, the bounding 0.6°F temperature error should be [

   ].

3.0 APPROACH

[

]

4.0 OVERVIEW

[

]

Appendix A.1, LEFM✓+ Inputs

--- This Appendix is Proprietary in its Entirety ---

[

]

Appendix A.2, LEFM✓ + Mass Flow and Temperature Uncertainties

- - - This Appendix is Proprietary in its Entirety - - -

This appendix calculates the uncertainties in mass flow and temperature as computed by the LEFM✓ + using the methodology described in Appendix E of Reference 1 and Appendix A of Reference 3[1], with uncertainties in the elements of these measurements bounded as described in both references[2]. [

]

Revision 0 of this appendix utilizes the bounding values of Reference 3 for all uncertainty elements[3] in the computation of plant specific uncertainties. [

]

Appendix A.3, Profile Factor (Calibration) Uncertainties

INCLUDED WITH REVISION 1 ONLY – N/A FOR REVISION 0

As noted above, the calibration test report for the spool piece(s) establishes the overall uncertainty in the profile factor of the LEFM✓ +. [

]

[1] Reference 3 (ER 157P) develops the uncertainties for the LEFM ✓ + system. Because this system uses two measurement planes, the structure of its uncertainties differs somewhat that of an LEFM✓ .

[2] [ ]

[3] [ ]

Appendix A.4, Uncertainties in the Measurement of Time Differences (Δt's)

[

]

Appendix A.5, Uncertainties in the Measurement of Transit Times (t's)

[

]

Appendix B, Total Thermal Power Uncertainty due to the LEFM✓+

The total thermal power uncertainty due to the LEFM✓+ is calculated in this appendix, [

]

Appendix C, Interface and Reconciliation Document

For completeness, an Interface and Reconciliation Document for this project is included as Appendix C. An interface and reconciliation document is provided to each LEFM✓+ customer in the initial phase of an upgrade project. This document breaks down the uncertainties that Caldon will calculate and justify for a specific uprate project, as well as those outside Caldon's scope. It provides a preliminary set of uncertainties, based on the results of Revision 0 of this document, for use by the customer or the customer's agent in establishing the amount of the power uprate. Appendix C also reconciles the results of Revision 0 analysis with the uncertainties calculated in the Topical Report and its supplements (References 1, 2, and 3).

5.0 REFERENCES

1) Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System", Rev. 0.

2) Caldon Engineering Report ER 160P Rev. 0, "Supplement to Topical Report ER 80P: Basis for a Power Uprate with the LEFM System", May 2000

3) Caldon Engineering-157P, "Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus", dated October 2001, Revision 5

4) ASME PTC 19.1-1985, Measurement Uncertainty

5) ISA-RP67.04.02-2000, Methodologies for the Determination of Set Points for Nuclear Safety-Related Instrumentation

Appendix A.1 is Proprietary to Caldon in its Entirety

Appendix A.2 is proprietary to Caldon in its Entirety

Appendix A.3
LEFM✓ + Spool Piece(s) Profile Factor and Profile Factor Uncertainty is proprietary to Caldon in its Entirety

Appendix A.4 is Proprietary to Caldon in its Entirety

Appendix A.5 is Proprietary to Caldon in its Entirety

Appendix B is Proprietary to Caldon in its Entirety

Appendix C
LEFM✓ + Interface and Reconciliation Documents, Calorimetric Uncertainties with the LEFM✓ and LEFM ✓+

CPL ROBINSON
LEFM Interface and Reconciliation Document
Calorimetric Uncertainties with the LEFM Check and Check Plus
JANUARY 17, 2002

I. Purpose

It is the purpose of this document to define precisely the uncertainties that Caldon will calculate and justify for a specific Appendix K uprate project. The uncertainties that are outside Caldon's scope are also defined, as well as the method for combining all uncertainties to obtain a power uncertainty. This document also breaks down the relationship between the uncertainties tabulated in Caldon reports covering the operation of the LEFM Check and LEFM CheckPlus instruments and the data that Caldon will provide for a specific uprate project.

II. Background

Reports covering the operation of the LEFM Check and LEFM CheckPlus instruments describe how the use of these instruments reduces the uncertainties in feedwater mass flow and feedwater enthalpy.[1] In combination with the uncertainties in the determination of other variables (steam enthalpy, for example), the uncertainties in feedwater mass flow and enthalpy establish the uncertainty in the core thermal power. The amount of a power increase allowable under an Appendix K uprate is directly dependent on achieving a thermal power uncertainty within bounds defined by the reports cited above.

The uncertainties in the variables measured by an LEFM, either mass flow or derived from its outputs (feedwater temperature and pressure, which are converted to enthalpy) are made up of several elements. These elements relate to the LEFM's measurements of time, to its dimensions, to the hydraulics of the installation and to correlations relating fluid temperature and density to sound velocity and pressure. With respect to the correlations and the measurements of time and dimensions, some of the uncertainties in mass flow are systematically related to the uncertainties in feedwater enthalpy while others are not. The structure and combination methods for power uncertainties are described further below.

III. Structure of the Thermal Power Uncertainties

The core thermal power as determined by a heat balance around the steam supply is given by:

(1) $Q_{RX} = W_{FW}(h_S - h_{FW}) + Q_{LOSS\ NET}$

Where Q_{RX} is the core thermal power
W_{FW} is the mass flow rate of the feed to the steam supply, the product of feedwater volumetric flow rate and feedwater density,

[1] Caldon Engineering Reports ER-80P, ER-160P and ER-157P

h_S is the enthalpy of the steam delivered by the steam supply, a function of its pressure and moisture content for saturated steam supplies and its pressure and temperature for superheated steam supplies,
h_{FW} is the enthalpy of the feedwater, a function of its temperature and pressure, and
$Q_{LOSS\ NET}$ is the net loss or gain in power from coolant pump heating, blowdown and/or reactor water purification, convective and radiant losses, etc.

The contributing uncertainties to the thermal power computation are defined by differentiating equation (1):

(2) $$dQ_{RX} = dW_{FW}\,(h_S - h_{FW}) + W_{FW}\,dh_S - W_{FW}\,dh_{FW} + dQ_{LOSS\ NET}$$

The contributors can be expressed per unit by dividing equation (2) by Q_{RX}.

(3) $$dQ_{RX}/Q_{RX} = dW_{FW}/W_{FW}\,[1 - (Q_{LOSS\ NET}/Q_{RX})] + [dh_S/(h_S - h_{FW})]\,[1 - (Q_{LOSS\ NET}/Q_{RX})] - [dh_{FW}/(h_S - h_{FW})]\,[1 - (Q_{LOSS\ NET}/Q_{RX})] + dQ_{LOSS\ NET}/Q_{RX}$$

Since the net gains and losses term is typically less than 1% of the reactor thermal power, the term $[1 - (Q_{LOSS\ NET}/Q_{RX})]$ may be taken as approximately 1.0. Hence,

(4) $$dQ_{RX}/Q_{RX} = dW_{FW}/W_{FW} + [dh_S/(h_S - h_{FW})] - [dh_{FW}/(h_S - h_{FW})] + dQ_{LOSS\ NET}/Q_{RX}$$

It should be pointed out that equation (4) applies algebraically only if all error contributors are systematically related to each other. Most of these components are *not* systematically related. If all of the components were random errors or biases the power uncertainty of equation (4) would be the square root of the sum of the squares of the individual terms on the right hand side of the equation. In fact, a combination of the two procedures is appropriate as described below.

The feedwater enthalpy is a function of its temperature and pressure. Likewise, the density of the feedwater, which the LEFM combines with the volumetric flow to compute mass flow, is a function of temperature and pressure. Because of this and other factors, certain elements of the uncertainty in feedwater enthalpy are combined systematically with the mass flow uncertainty, while other elements, unrelated to the mass flow measurement, are combined randomly. For convenience in defining the combination of terms, the feedwater enthalpy will be related to its temperature and pressure by the following:

(5) $$h_{FW} = \delta h/\delta p\,\big|_T\,(p_{FW} - p_0) + \delta h/\delta T\,\big|_p\,(T_{FW} - T_0) + h_0$$

The computation of feedwater enthalpy from temperature and pressure by the plant computer—part of the thermal power computation—may be carried out by a more complex algorithm than that of equation (5), or the enthalpy may be determined from a look up table. Equation (5) is used here simply as a convenience for developing the elements of the error contributors to feedwater enthalpy. Using equation (5), the uncertainty in feedwater enthalpy is:

(6) $dh_{FW} = \delta h/\delta p \mid_T dp_{FW} + \delta h/\delta T \mid_p dT_{FW} + dh_0$

Here dh_0 represents the potential bias in the enthalpy algorithm of the plant computer.

Rewriting equation (4) to incorporate equation (6), and rearranging terms:

(7) $$dQ_{RX}/Q_{RX} = \overbrace{\{dW_{FW}/W_{FW}\}}^{\mathbf{A}} - \overbrace{\{[1/(h_S - h_{FW})]\,[\delta h/\delta p \mid_T dp_{FW} + \delta h/\delta T \mid_p dT_{FW}]\}}^{\mathbf{B}}$$

$$+ \overbrace{\{[1/(h_S - h_{FW})]\,dh_0\}}^{\mathbf{C}} + \overbrace{\{[1/(h_S - h_{FW})]\,dh_S\}}^{\mathbf{D}} + \overbrace{\{dQ_{LOSS\ NET}/Q_{RX}\}}^{\mathbf{E}}$$

In the determination of overall thermal power uncertainty, terms **A** and **B** will be provided by Caldon, based in part on a feedwater pressure uncertainty provided by the utility. [ ]

Terms **C**, **D**, and **E** are outside of Caldon's scope, are based on other plant instruments, and are to be provided by others.

Caldon will provide a single uncertainty, **AB**, expressed as a percentage of the rated thermal power, that encompasses terms **A** and **B**. Under normal circumstances, there will not be a systematic relationship between term **AB**, on the one hand, and terms **C**, **D**, and **E**, on the other. Likewise, there will normally not be systematic relationships among terms **C**, **D**, and **E.** Therefore, the utility will normally compute the total thermal power uncertainty from the following.

(8) $dQ_{RX}/Q_{RX} = [(\mathbf{AB})^2 + (\mathbf{C})^2 + (\mathbf{D})^2 + (\mathbf{E})^2]^{1/2}$

IV. Reconciliation of Uncertainties for CPL ROBINSON With the Uncertainties Quoted in Caldon Engineering Reports

Table 1 below compares the expected site-specific bounding uncertainties for CPL ROBINSON to the following Caldon Engineering Reports:

- ER-80P, Rev. 0, the original Caldon topical report from 1997 that requests a 1% power uprate based on an accuracy of the LEFM Check system bounded by 0.6% thermal power accuracy.
- ER-160P, Rev. 0, which presents instrument uncertainties exactly the same as those in ER-80P. ER-160P recognizes that, in accordance with NRC Rulemaking in June 2000, a power uprate up to and including 1.4% power can be requested for the LEFM Check System (since ER-80P demonstrates that its accuracy supports a thermal power uncertainty of ± 0.6%).

- ER-157P, Rev. 5, which describes Caldon's next generation LEFM CheckPlus. ER-157P revises the uncertainty analyses of ER-80P to reflect actual LEFM Check data as applied to a typical single flow measurement application (similar to CPL ROBINSON). It also shows that the LEFM Check system can achieve power uncertainties as low as ± 0.5% thermal power accuracy. Additionally, ER-157P demonstrates that the LEFM CheckPlus can support power uncertainties as small as ± 0.3%.

It should be reiterated that the data of column 5 of Table 1 are bounding. [

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Information on this page is proprietary to Caldon in its entirety.

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United States Nuclear Regulatory Commission
Attachment IV to Serial: RNP-RA/02-0104
6 pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AFFIDAVIT AND APPLICATION FOR WITHOLDING FROM PUBLIC DISCLOSURE

Caldon, Inc.

July 11, 2002
CAW 02-01

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Caldon, Inc. Engineering Report: ER- 267, Bounding Uncertainty Analysis for Thermal Power Determination at CP&L Robinson Nuclear Power Station Using the LEFM✓+ System"

Gentlemen:

This application for withholding is submitted by Caldon, Inc. ("Caldon") pursuant to the provisions of paragraph (b)(1) of Section 2.790 of the Commission's regulations. It contains commercial strategic information proprietary to Caldon and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.790, Affidavit CAW-02-01 accompanies this application for withholding setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information, which is proprietary to Caldon, be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW-02-01 and should be addressed to the undersigned.

Very truly yours,

Calvin R. Hastings

Calvin R. Hastings
President and CEO

Enclosures

1070 Banksville Avenue • Pittsburgh, PA 15216
Tel: 412-341-9920 • Fax: 412-341-9951 • Web: www.caldon.net

July 11, 2002
CAW-02-01

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Calvin R. Hastings, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Caldon, Inc. ("Caldon") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Calvin R. Hastings

Calvin R. Hastings,
President and CEO
Caldon, Inc.

Sworn to and subscribed before me

this 11th day of

July, 2002

Joann B. Thomas

Notarial Seal
Joann B. Thomas, Notary Public
Pittsburgh, Allegheny County
My Commission Expires July 28, 2003
Member, Pennsylvania Association of Notaries

1. I am the President and CEO of Caldon, Inc. and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of Caldon.

2. I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Caldon application for withholding accompanying this Affidavit.

3. I have personal knowledge of the criteria and procedures utilized by Caldon in designating information as a trade secret, privileged or as confidential commercial or financial information.

4. Pursuant to the provisions of paragraph (b) (4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Caldon.

   (ii) The information is of a type customarily held in confidence by Caldon and not customarily disclosed to the public. Caldon has a rational basis for determining the types of information customarily held in confidence by it and, in that connection utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Caldon policy and provides the rational basis required.

   Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential advantage, as follows:

   (a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Caldon's competitors without license from Caldon constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, and assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Caldon, its customer or suppliers.

(e) It reveals aspects of past, present or future Caldon or customer funded development plans and programs of potential customer value to Caldon.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Caldon system, which include the following:

(a) The use of such information by Caldon gives Caldon a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Caldon competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Caldon ability to sell products or services involving the use of the information.

(c) Use by our competitor would put Caldon at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Caldon of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Caldon in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Caldon capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence, and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Caldon, Inc. Engineering Report: ER-267 Bounding Uncertainty Analysis for Thermal Power Determination at CP&L Robinson Nuclear Power Station Using the LEFM ✓+ System". This information is submitted for use by the NRC Staff in their review of the MUR uprate license amendment request of CP&L Robinson Nuclear Power Station.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Caldon because it would enhance the ability of competitors to provide similar flow and temperature measurement systems and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Caldon effort and the expenditure of a considerable sum of money.

In order for competitors of Caldon to duplicate this information, similar products would have to be developed, similar technical programs would have to be performed, and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing analytical methods and receiving NRC approval for those methods.

Further the deponent sayeth not.