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W3F1-2002-0009

January 21, 2002

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

SUBJECT: Waterford Steam Electric Station, Unit 3
Docket No. 50-382
Technical Specification Change Request, NPF-38-238
Appendix K Margin Recovery – Power Uprate Request
Response to Request for Additional Information

REFERENCES:

1. Entergy letter dated September 21, 2001, TSCR 38-238, "Appendix K Margin Recovery – Power Uprate Request" (W3F1-2001-0091)
2. NRC letter dated November 6, 2001 (I&C Branch)
3. NRC letter dated November 8, 2001 (Electrical Branch & Radiological Consequences)
4. NRC letter dated November 28, 2001 (Human Performance Branch)
5. NRC letter dated December 21, 2001 (Material & Chemical Engineering, Mechanical & Civil Engineering, and Reactor Systems Branches)
6. Entergy letter dated December 10, 2001, TSCR 38-238, Response to Requests for Additional Information (W3F1-2001-0117)
7. Entergy letter dated January 16, 2002, TSCR 38-238, Response to Request for Additional Information (W3F1-2002-0006)

Dear Sir or Madam:

In accordance with 10CFR50.90, Entergy Operations, Inc. (Entergy) submitted, by letter dated September 21, 2001 (Reference 1), a request for changes tothe Waterford Steam Electric Station, Unit 3 (Waterford 3) Operating License and Technical Specifications associated with an increase in the licensed power level. The changes involve a proposed increase in the power level from 3,390 MWt to 3,441 MWt representing a measurement uncertainty recapture power uprate. The NRC Staff has returned four Requests for Additional Information (RAI) (References 2-5). Entergy provided a response to the first two RAIs in Reference 6 and to the third RAI in Reference 7. The response to the remaining RAI is provided in the Attachment.

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The proposed change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. The attached responses do not impact that conclusion.

Entergy requests that the effective date for this TS change to be within 60 days of startup from Refueling Outage (RF) 11. Although this request is neither exigent nor emergency, your prompt review and approval prior to startup from RF 11 is requested. Entergy would like to implement the increased power level upon startup from our upcoming RF11 scheduled to start on March 22, 2002.

There are no new commitments associated with the attached responses. Summary listings of other commitments associated with this request were provided in References 1, 6, and 7. Should you have any questions or comments concerning this response, please contact Jerry Burford at (601) 368-5755.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 21, 2002.

Sincerely,

John T. Herron
Vice President, Operations
Waterford 3

JTH/FGB/cbh

Attachment: Response to Request for Additional Information

cc: E.W. Merschoff, NRC Region IV
N. Kalyanam, NRC-NRR
J. Smith
N.S. Reynolds
NRC Resident Inspectors Office
Louisiana DEQ/Surveillance Division
American Nuclear Insurers

Attachment to

W3F1-2002-0009

Response to Requests for Additional Information Related to Power Uprate

RESPONSE TO REQUESTS FOR ADDITIONAL INFORMATION

By letter dated September 21, 2001, Entergy Operations, Inc. (the licensee), proposed a license amendment to change the Technical Specifications (TS) for Waterford Steam Electric Generating Station, Unit 3 (Waterford 3). The proposed amendment addresses modifications necessary to increase the rated thermal power of Waterford 3 from 3,390MWt to 3,441 MWt, an increase of 1.5%. These changes result from increased feedwater flow measurement accuracy to be achieved by utilizing high accuracy ultrasonic flow measurement instrumentation to be installed in the main feedwater system piping. The NRC Staff has returned a request for additional information (RAI) in a letter December 21, 2001 (the Materials & Chemical Engineering Branch, Mechanical & Civil Engineering Branch, and Reactor Systems Branch questions below). The response to the RAI is provided below.

Materials & Chemical Engineering Branch

1. The Waterford 3 submittal for the 1.5% power uprate does not provide any discussion concerning the effects of the proposed uprate on the integrity of the reactor vessel with respect to:

    a. Pressurized thermal shock
    b. Heatup and cooldown pressure-temperature (P/T) limits
    c. Upper shelf energy

    Please address the effects of the uprate on the above topics.

    Response:

    Neutron fluence projections on the vessel were evaluated for the uprated power level. The fluence projections serve as input to the reactor vessel (RV) integrity evaluations. Specifically, fluence values are used to:
    - evaluate the end-of-life (EOL) transition temperature shift for development of the surveillance capsule withdrawal schedules,
    - determine EOL upper shelf energy (USE) values,
    - adjust reference temperature values for determining the applicability of the heatup and cooldown curves, and
    - determine compliance with the pressurized thermal shock (RT_{PTS}) screening criteria.

    Additional clarification of the fluence discussion provided in Section 3.6.4 of the original submittal is provided in the response to question 19 below. The following information provides additional discussion related to the effects of the power uprate on RV integrity.

    - <u>Pressurized Thermal Shock (PTS)</u>

      The RT_{PTS} values were evaluated for reactor vessel beltline plates, axial weld seams, and circumferential weld seams for end-of-license operation based on the NRC screening criteria for pressurized thermal shock (10CFR50.61). The RT_{PTS} values for beltline region materials of the Waterford 3 reactor vessel used as the basis for

the current pressure-temperature curves (see TS 3.4.8.1), as previously calculated, bound the conditions for the 1.5-percent uprate. The Waterford 3 RT_{PTS} values remain below the NRC screening criteria values using projected fluence values through 16 effective full power years (EFPY).

- Heatup and Cooldown Pressure-Temperature Limits:

  Heatup and cooldown pressure-temperature curves and limits are addressed in Section 3.6.4 of the original submittal.

- Upper Shelf Energy (USE):

  Since the bounding neutron fluence values for the 1.5-percent uprate have decreased, the projected 16 EFPY USE values will exhibit a smaller decrease than previously predicted for Waterford 3. It was previously determined that the beltline materials in the Waterford 3 reactor vessel will have an USE greater than 50 ft-lb through the period of the currently licensed curves (16 EFPY) as required by 10CFR50, Appendix G.

2. In Section 3.6.2.6.1 to Attachment 2 of its application, the licensee discussed structural integrity of the steam generators (SGs) under power uprate conditions. It appears that the structural integrity evaluation of SG tube degradation was focused on satisfying the stress and fatigue specifications of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code).

   a) The NRC staff requests that the licensee evaluate the structural integrity of the SG tubes in terms of Regulatory Guide 1.121, "Bases of Plugging Degraded PWR (Pressurized Water Reactor) Steam Generator Tubes."
   b) The licensee also needs to evaluate and discuss the acceptability of the leakage integrity of the SG tubes under power uprate conditions.

   Response:

   a) The Waterford 3 Degradation Assessment evaluated structural integrity under three conditions. The criteria used for the assessment are consistent with the guidance of Regulatory Guide 1.121. The conditions and the criteria established to evaluate flaws with respect to structural integrity based on Waterford 3 tubing material properties are presented below:

      Conditions:
      1 - Normal operating differential pressure (NOP)
      2 - Accident condition or 1.4 times MSLB (MSLB)
      3 - Regulatory Guide 1.121 or 3 times the Normal Operating Differential Pressure (3ΔP)

Table 1 for Question 2
Waterford 3 SG Tube Structural Integrity Criteria (post-power uprate)

Condition	Normal Value	Temperature-Compensated Value
NOP	1420 psi	1612 psi
MSLB	1750 psi	1975 psi
$3\Delta P$	4260 psi	4736 psi

The NOP value above has been adjusted for the power uprate primary and secondary pressure differential (pre-uprate value was 1402 psi.) The temperature compensated values have been adjusted 10% for temperature and an additional 50 psi added for instrumentation error.

b) SG leakage integrity is evaluated and screened in accordance with the table below (Table 2). The most limiting criteria for leakage is that for an axial flaw (MD-THR-L Minimum Depth Threshold for Leak Testing). These criteria for flaw evaluation are not affected by the power uprate.

Critical to the success of in-situ tube selection is properly identifying the structural length of flaws and threshold values for critical NDE parameters. The table summarizes the planned threshold values to be used during the Waterford 3 2002 Outage.

Table 2 for Question 2
In-Situ Screening Threshold at Waterford 3

Orientation	**Surface**	**Parameter**	**Threshold**
Axial	ID/O	LSTR	0.416
Axial	ID	MDTHR-P	45%
Axial	OD	MDTHR-P	58%
Axial	ID	ADTHR-P	CC2 used at site
Axial	OD	ADTHR-P	CC2 used at site
Axial	ID/O	MDTHR-L	80%
Circumferential	ID/O	PDATHR	68%
Circumferential	ID/O	CATWSL	273 degrees
Circumferential	OD	AV	Not given

Nomenclature	
LSTR	Length Based structural limit for freespan, straight length tube sections, in inches
MDTHR-P	Maximum depth threshold for pressure testing, in percent through-wall
MDTHR-L	Minimum depth threshold for leak testing, in percent through-wall
ADTHR-P	Average indication depth threshold for pressure testing which is a function of length, in percent through-wall per unit length

PDATHR	Percent degraded area, integrated over 360 degree at 3dP burst limit
CATWSL	Crack angle through-wall structural limit that satisfies the 3dP burst requirement
AV	360 degree average of the vertical amplitude (in volts)- also called voltage integral

3. NRC has issued the following generic communications regarding SG tube plugs:

- NRC Information Notice 89-65, "Potential for Stress Corrosion Cracking in Steam Generator Tube Plugs Supplied by Babcock and Wilcox"
- NRC Information Notice 89-33, "Potential Failure of Westinghouse [Electric Corporation] Steam Generator Tube Mechanical Plugs"
- NRC Bulletin No. 89-01, "Failure of Westinghouse Steam Generator Tube Mechanical Plugs," and Supplements 1 and 2, and
- NRC Information Notice 94-87, "Unanticipated Crack in a Particular Heat of Alloy 600 Used for Westinghouse Mechanical Plugs for Steam Generator Tubes"

The application discusses SG tube plugs in Section 3.6.2.6.3 of Attachment 2.

a) Discuss if any of the above NRC generic communications are applicable to the tube plugs used in the Waterford 3 SGs and the steps that have been taken to meet the NRC staff's recommendations in the above generic communications.

b) Discuss any degradation detected in tube plugs and the associated repair method other than those in discussed in Item 3a.

Response:

a) As part of Waterford 3's commitment to remove all Westinghouse Inconel-600 mechanically rolled ribbed plugs, Framatome, Inc. removed the last 11 cold leg plugs during RF8 (Spring 1997). The repaired locations were re-plugged with FTI's Inconel-690 long threaded mechanical plugs that were seated and torque rolled in virgin tube inner diameter (ID) surface. This process avoided the potential of seating a replacement plug in a tube location where the previous Westinghouse ribbed plug could have damaged the tube ID surface.

   The hot leg Westinghouse Inconel-600 mechanical tube plugs were removed Spring 1991 by machining the locations with a drill and re-inserting a Westinghouse Inconel-690 mechanical ribbed plug.

b) The generic communications discussed above dealt with Inconel-600 mechanical-ribbed plugs and Primary Water Stress Corrosion Cracking (PWSCC). Waterford 3 did not experience any tube plug degradation specific to Westinghouse Inconel-600 mechanical ribbed plugs or potential PWSCC prior to removal.

4. Discuss the impact of the power uprate on each of the degradation mechanisms of the SG tubes and on the inspection intervals for the SG tubes.

Response:

The following modes of degradation are currently the only active damage mechanisms identified and depth sized for the Waterford 3 CE Model 3410 SGs:

- Wear at the square bend locations (BWs) and upper bundle supports
- Top of the tubesheet ODSCC, PWSCC and Volumetric (SCIs & SAIs)
- Egg-crate axial and volumetric
- Tight radius U-Bend PWSCC, Rows 1 – 3 (Critical Area)

Waterford 3 SGs have experienced wear at diagonal and vertical supports since start-up and the first cycle of operation. As a result, Waterford 3 has administratively plugged 305 tubes in each SG surrounding the stay cylinder region as a preventive measure. To date Waterford 3 is tracking over 600 wear indications at structures. The majority of the wear indications are located at square bend supports. The balance of the indications are located within straight sections of tubing were located at or below the seventh hot and cold leg egg-crates.

The Appendix K power up-rate will increase primary water bulk temperature by as much as 1 ^{O}F. This additional increase above the present 600 ^{O}F bulk water temperature will affect the temperature dependent damage mechanisms (i.e., outer-diameter stress corrosion cracking (ODSCC) and PWSCC). Waterford 3 currently inspects SGs every scheduled refueling outage (18-month cycle). The 1.5% power uprate is expected to have negligible effects with regard to ODSCC and PWSCC or on the inspection schedule.

The Waterford 3 inspection program subsequent to RF 11 (Spring) 2002 will continue to be comprehensive and adhere to industry recommendations as applicable with regards to potential damage mechanisms and industry qualified inspection techniques.

5. In Section 4.1.2, Flow Accelerated Corrosion (FAC), the application states the following:

> ".... CHECWORKS models will be revised, as appropriate, to incorporate flow and thermodynamic states that are projected for uprated conditions. The results of these models will be factored into future inspection/pipe replacement plans consistent with the current FAC Program requirements."

The staff requests additional information on the revisions to the current CHECWORKS models. Specifically, the staff requests details on the revisions to the models and details on how the impact of these changes will be factored into future inspections or pipe replacements. These details should include a comprehensive list of changes to the models, the means by which the new results will be captured into future inspections or pipe replacements, and the basis for the scheduling of the pipe replacements.

Response:

Waterford 3 uses EPRI CHECWORKS version 1.0F as the Flow Accelerated Corrosion (FAC) ultrasonic thickness data management system. There will be no changes to the CHECWORKS model, methodology, or software used to quantify the effects of FAC on piping systems at Waterford 3; however, inputs to the model will be adjusted to reflect plant operating conditions. The existing plant model parameters will be updated to reflect the power uprated conditions using the Plant Data Management task in CHECWORKS. Plant Data Management allows editing of the parameters (temperature, pressure, enthalpy, steam quality etc.) data in the database. These changes are necessary to assure that the model reflects the new plant operating conditions and are vital to maintaining the accuracy of the Wear Rate Analysis function in CHECWORKS. The overall impact of updating the model parameters will be minimal with respect to component inspection / replacement.

Examples of updated parameters include:

- Power Level (%)
- Steam Flow Rate (Mlb/hr)
- Pressure (psia)
- Temp. (°F)
- BD Rate (Mlb/hr)
- Carryover (%)
- Steam enthalpy

6. In Section 3.6.1, with regard to loss-of-coolant accident (LOCA) hydraulic loads produced by the tributary lines, confirm that the current design basis LOCA produced by the as-built tributary lines are bounded by the design basis LOCA resulting from the mechanistic failure of main coolant loop piping.

Response:

Westinghouse assessed the impact of changed RCS conditions on the LOCA-induced hydraulic blowdown loads. This evaluation demonstrated that the original UFSAR design basis LOCA hydraulic loads resulting from the mechanistic failure of main coolant loop piping with an initial inlet temperature of 553°F would bound analogous loadings resulting from tributary line breaks with an inlet temperature as low as 533°F. Consequently, the original controlling set of time history forcing functions acting on the reactor vessel (RV) shell and internals structures, which were determined from the LOCA hydraulic loads, would also bound any RV LOCA forcing functions resulting from the tributary line breaks.

This evaluation was based on earlier analyses performed for Waterford 3 for a possible extended 8% power uprate. Those analyses had compared the LOCA induced pressure loadings on the fuel and reactor vessel internals for the original design basis mechanistic cold leg break to design basis induced pressure loadings produced by tributary line breaks at higher power and lower inlet temperatures.

The RV shell and internals forcing functions are part of the overall set of input LOCA loadings applied to various RCS structural model locations to determine major component and component support loads. Other applied time historyloadings are those due to pipe tension release and jet impingement at the break location, and, where applicable, those due to subcompartment pressurization. The fact that the tributary line breaks have smaller break opening sizes than the corresponding mechanistic breaks leads to the conclusion that the original design basis loadings would also be more severe than those resulting from any of the tributary line breaks. Therefore, the original design basis RCS structural loads due to LOCA remain bounding for Waterford 3 under leak before break (LBB) and Appendix K power uprate conditions.

7. In Section 3.6.2.2.2, with regard to flow and pump-induced vibration, you state that the current analysis uses a mechanical flow that changes by less than 1 percent for the revised operating condition. Provide the basis for your conclusion. You also state that the revised operating conditions after the T_{hot} fluid density, but did not provide the magnitude of the change in T_{hot} fluid density. Provide the basis and magnitude of the change in T_{hot} fluid density and confirm that there is no increase in the potential for flow-induced vibration.

Response:

The following table compares fluid densities at design and at Appendix K uprate conditions:

	Design Temp (°F)	Design Density (lb/ft^3)	App K Temp (°F)	App K Density (lb/ft^3)	% increase
T_{hot}	611.0	42.50	600.2	43.42	2.16
T_{cold}	553.0	46.82	545.0	47.32	1.07

Note – all density values are determined at the design pressure of 2500 psi

<u>Flow-Induced Vibration Effects</u>

Due to the overall reduction in operating temperatures, there are small increases in the fluid densities, which in turn increase the severity of any flow-induced vibration effects on the reactor vessel (RV) core support barrel at a given flow rate. However, the original design analyses were based on a conservatively low RV inlet temperature (i.e., T_{cold}) of 500 °F, as opposed to the design temperature of 553 °F, which is typically used in RCS structural analyses. The use of a 500 °F T_{cold} produced conservative fluid densities, when compared to the fluid densities associated with actual RV operating temperatures. The comparison to Appendix K uprate conditions is shown below.

	Analysis Temp (°F)	Analysis Density (lb/ft^3)	App K Temp (°F)	App K Density (lb/ft^3)	% increase
T_{cold}	500.0	49.86	545.0	47.32	-5.10

Therefore the effects of fluid density on RV core support barrel loading are less severe at Appendix K conditions.

Section 3.6.2.2 of the power uprate licensing report states that Appendix K conditions will cause a small change in the flow rate. This is based on the fact that the Appendix K power uprate is 1.5%. The more significant point, however, is shown in the expanded Table 3.3.1-1 included in the response to Question 11. Table 3.3.1-1 data shows that the post Appendix K uprate primary flow rate is projected to be approximately 8% greater than the original design flow rate, when the plant is at steady state normal operating conditions.

As stated above, the analysis of record (AOR) was conservatively based on a RV inlet design temperature of 500 °F. Similarly, the primary flow rate considered in the AOR is 120% of the design flow rate, which clearly envelops the flow rate projected for Appendix K conditions. Consequently, the flow induced vibration effects considered in the AOR are more severe than the flow induced vibration effects associated with Appendix K uprate conditions, and their contribution to the overall loading of the RV core support barrel remains bounding.

Pump-Induced Vibration Effects

Pump-induced vibration effects for a possible 8% power uprate at Waterford 3 were recently assessed, resulting in the conclusion that the current analyses remain applicable, since power uprate does not affect conditions in the RV downcomer. This conclusion would also apply to Appendix K uprate conditions.

8. In Section 3.6.2.2.3, with regard to the structural integrity of reactor internals for the 1.5, percent power uprate condition, you based your conclusions on results of previous analyses either performed by you or by others. However, details of such analyses were not provided. Please provide a justification for the applicability of these analyses to the 1.5 percent power uprate condition. Provide a summary of evaluation results, including the maximum calculated stresses and cumulative fatigue usage factors (CUFs), for the critical reactor internal components including the baffle/barrel region components, core barrel, baffle plate, baffle/former bolts, and lower core plate for the 1.5 percentuprated power conditions. Also provide the ASME Code and Code Edition used for the evaluation of the reactor internal components, and if different from the Code of Record, please justify and reconcile the differences.

Response:

The Appendix K uprate evaluation considered analyses performed by Westinghouse for both Waterford 3 and SONGS Units 2 and 3. Selected results from the SONGS analyses are applicable because of the similarity of the reactor vessel internals designs and the fact that the hydraulic loads considered in the SONGS analyses are bounding. Conversely, the recently performed Waterford 8% extended power uprate assessment considered bounding thermal gradients.

Current fuel weight and fuel spring loads were also compared. The SONGS calculations bounded current fuel weight and fuel spring loads. The evaluation concluded that stresses in the Waterford reactor vessel internals for Appendix K uprate conditions were bounded by those previously calculated for Waterford and SONGS. The applicable bounding stresses are shown in the table below:

Table for Question 8
RVI Stress Summary for 101.7% Power Uprate (2)

Component & Location	Stress Type (1)	Stress (psi)	Allowable (psi)	Margin (%)
CSB Upper Flange-to-Barrel Weld	MbQ	35,104	48,300	27.3
CSB Lower Flange-to-Barrel Weld	Mb	15,963	24,150	33.9
CSB-to-LSS Flexure Weld	Mb	23,287	23,426	0.6
CSB-to-LSS Flexure	M	12,534	16,100	22.1
CSB Inlet Impingement Area	M	5,073	16,100	68.5
CSB Outlet Nozzle	MbQ	38,225	48,300	20.9
CSB Center Cylinder	MbQ	39,725	48,300	17.8
CSB Cylinder-to-Snubber Weld	mb	18,264	21,735	16.0
Core Support Plate (CSP)	mbQ	46,271	48,300	4.2
LSS Beams	mb	4,200	10,740	60.9
LSS Columns	mbQ	44,819	48,300	7.2
CSP-to-LSS Cylinder Weld	m	2,400	7,245	66.9
CSP-to-Core Shroud Weld	m	4,900	14,490	66.2
Insert Pin-to-LSS Beam Interface	br	2,700	17,900	84.9
CEA Shroud @ UGS Support Plate	mbQ	29,233	48,300	39.5
CEA Shroud @ Flow Channels	mb	11,470	24,150	52.5
Modified Shroud @ Flow Bypass	mb	11,797	24,150	51.2
CEA Shroud Cap Screw	mbQ	53,863	72,900	26.1
UGS Flange-to-Cylinder Weld	mb	17,988	24,150	25.5
UGS Flange Top	mbQ	45,354	48,300	6.1
UGS Grid Beams	mbQ	14,244	48,300	70.5
Fuel Alignment Plate	mbQ	40,201	48,300	16.8
Core Shroud	mbQ	39,800	48,300	17.6
Instrument Tube Support	mb	21,450	24,150	11.2
Water Level Monitoring Support	mbQ	42,046	48,300	12.9
Alignment Key-to-CSB Keyway (4)	br	23,688	17,900	-32.3 (3)
Alignment Keys	mb	31,779	64,950	51.1
Flow Channel Extension	mb	2,935	24,150	87.8

Notes:
1. Stress legend:
   m = primary membrane stress
   mb = primary membrane plus bending stress
   mbQ = primary plus secondary stress
   br = bearing stress
2. All stresses reflect the Normal Operation plus Upset design condition.
3. The negative margin on Alignment Key-to-CSB Keyway bearing stress indicates that the key-to-keyway interference fit could loosen under normal operation plus OBE loading conditions. Note, however, that the Alignment Keys are physically constrained, and that any loosening of the interference fit would have no adverse effect on plant operation. Should

an OBE occur, it would be advisable to check the key-to-keyway interference fit during the next scheduled in-service inspection of the CSB.

4. Allowable stresses for Alignment Key and Cap Screw material were introduced in the 1974 Code.

Fatigue Usage

As described above, reactor vessel internals stresses for the Waterford 3 1.5% power uprate are bounded by previously calculated stresses for Waterford 3 and SONGS. In both of the previous analyses, fatigue usage was addressed by demonstrating that the peak alternating stress required to achieve maximum allowable fatigue usage was greater than the peak alternating stress calculated for any of the internals components. Thus, while exact values for the cumulative fatigue usage factors of individual internals components are not available, it has been demonstrated that they are less than the Code allowable of 1.0.

Applicable ASME Code Edition

Both of the previous analyses reference the 1980 edition of the ASME Code. However, the Code of Record for Waterford 3 is the 1971 edition. The only inconsistency associated with the use of the later Code edition is related to the stress allowables, which are multiples of S_m, σ_y or σ_u. The 1980 edition S_m value for the material of interest, 304SS, is slightly greater than the corresponding 1971 edition value, which would result in a non-conservative comparison to the corresponding as-calculated stresses. The allowables shown in the above table are based on the 1971 edition, with the noted exception of the alignment key and cap screw material. Therefore, the tabulated comparison of calculated stresses to allowable stresses shown above and the fatigue usage assessments are valid with respect to ASME Code requirements.

9. In Section 3.6.2.9, with regard to Nuclear Steam Supply System (NSSS) piping, provide the calculated stresses, CUFs, and allowable stress for the most critical locations in the piping system.

Response:

The following two tables present the Primary Stress Intensities, Primary-plus-Secondary Stress Intensity Ranges, stress allowables, calculated stress margins, and usage factors for the reactor coolant piping, as compiled from the analysis of record (AOR). These results, which are based on the design loads and either design temperature or higher operating temperatures, remain bounding and valid for post Appendix K Uprate conditions.

Table 1 for Question 9
Primary Stress Intensity Summary

Component	Location	Primary S. I. Class.	S. I. (psi)	Allow. S. I. (psi)	Margin (psi)	Margin (%)
Piping	C.L. elbow	$P_m + P_b$	54,400	58,640	4,280	7.3
Safety Injection Nozzle	nozzle-pipe juncture	P_L	19,860	70,370	50,510	71.8
		$(P_m \text{ or } P_L) + P_b$	20,670	70,370	49,700	70.6
		P_b	28,040	49,000	20,960	42.8
Surge Nozzle	nozzle-pipe juncture	P_m	15,490	42,510	27,020	63.6
	nozzle body	$P_m + P_b$	19,080	63,770	44,690	70.1
Charging Inlet Nozzle	safe end	P_m	7,590	41,880	34,290	81.9
	nozzle-pipe juncture	P_m	8,240	52,500	44,260	84.3
		$P_L + P_b$	23,390	78,750	55,360	70.3
Pump Nozzle Safe End	safe end	$P_m + P_b$	15,500	57,900	42,400	73.2
RTD Nozzle - H.L./C.L.	Cold Legs	P_L	8,950	35,000	26,050	74.4
		P_m	10,890	55,900	45,010	80.5
		P_L	11,810	83,900	72,090	85.9
Pressure Measurement & Sampling Nozzle	Hot Leg	P_L	8,460	35,000	26,540	75.8
		P_m	5,990	55,900	49,910	89.3
		P_L	7,640	83,900	76,260	90.9
RTD Nozzle - Surge Line	nozzle weld	P_L	11,500	24,400	12,900	52.9
		P_m	12,250	59,500	47,250	79.4
		P_L	14,830	89,300	74,470	83.4
Sampling Nozzle - Surge Line	nozzle weld	P_L	10,990	24,400	13,410	55.0
		P_m	11,890	59,500	47,610	80.0
		P_L	14,830	89,300	74,470	83.4

Table 2 for Question 9
Primary-Plus-Secondary Stress Intensity Range (S_n) and Usage Factor Summary

Component	Location	Max. S_n (psi)	$3S_m$ (psi)	Margin (psi)	Margin (%)	Location	Max. Usage factor
Piping	N/A	N/A	N/A	N/A	N/A	surge line elbow	0.937
	H.L. elbow	41,020	51,000	9,980	19.6	C.L. elbow	0.052
Safety Injection Nozzle	safe end (outside)	92,010	58,000	-34,010*	-58.6*	safe end	0.281
Surge Nozzle	safe end (outside)	68,570	52,200	-16,370*	-31.4*	safe end (outside)	0.473
Charging Inlet Nozzle	safe end (inside)	91,080	53,700	-37,380*	-69.6*	safe end (inside)	0.884
RTD Nozzle - H.L./C.L.	H.L. (inside)	19,490	69,900	50,410	72.1	H.L.	0.008
Press. Meas. & Samp. H.L.	inside	20,650	69,900	49,250	70.5	outside	0.010
RTD Nozzle - Surge Line	nozzle weld	36,890	49,800	12,910	25.9	nozzle weld	0.896
Sampling Nozzle - Surge Line	nozzle weld	34,110	49,800	15,690	31,500	nozzle weld	0.572
Drain Nozzle - C.L.	bi-metallic boundary	72,490	53,000	-19,490*	-36.8*	bi-metallic boundary	0.772
Shutdown Cooling Outlet Nozzle	bi-metallic boundary	71,010	53,400	-17,610*	-33.0*	bi-metallic boundary	0.888

* acceptable on the basis of an elastic-plastic analysis

10. In Section 3.6.3.2, with regard to primary piping thermal expansion loads, you stated that ΔT values associated with current and uprated conditions are both less than the ΔT value used in the analysis of record (AOR). Provide the ΔT values associated with the current and uprated conditions, and that used in the AOR.

Response:

The primary side ΔT values are equal to ($T_{hot} - T_{cold}$), and can be found in the expanded Table 3.3.1-1 included in the response to Question 11 below. ΔT associated with the AOR is equal to 58 °F, ΔT for the current conditions is equal to 54.5 °F, and ΔT for the uprated condition is equal to 55.2 °F.

Reactor Systems Branch

11. The NSSS Operating Point parameters for power uprate conditions were calculated for a power uprate of 1.7% (3,448 MWt) in order to bound the requested power uprate of 1.5%.
    a. Provide a table comparing the NSSS operating points at the current 100% power (3390 MWt) to the recalculated uprate NSSS Operating points in Table 3.3.1-1.
    b. Provide a listing of the FSAR Chapter 15 accident/transient safety analyses which incorporate these uprate operating point parameters. For those that do not, provide justification that the current values used in the analyses are bounding.

Response

a. An expanded version of Table 3.3.1-1 from the original submittal is provided below to include the current plant operating conditions. The operating point calculation did not include the total SG liquid mass, thus it was not included in the table.

Table for Question 11
Revised Table 3.3.1-1
NSSS Original Design, Current Operating, and Appendix K Uprate Nominal Operating Parameters
(see Response to question 11 above)

Parameter	Original Design Conditions	Current Operating Conditions	CY 12 Nominal Operating Point
Core Power MWt (input)	3,390	3,390	3,448[2]
No. of plugged tubes per SG	50	421	500
Primary Bulk T_h, ^0F	611	600.19	600.2
Primary T_c, ^0F	553	545.72	545
Primary ΔT, ^0F	58	54.49	55.2
Primary Flow Rate, lbm/sec (input)	41,111.1	44,346.8[3]	44,522.4[3]
Primary Pressure, psia	2250	2248.7	2250
Feedwater Temperature, ^0F	445	440.1	442.7
Feedwater Enthalpy, BTU/lbm (input)	424.9	419.3	422.2
FW Flow Rate per SG, lbm/sec	Same as Steam Flow	2,091.1	2,135.9
SG Blowdown Flow per SG, lbm/sec (input)	NA	17.9	17.48
SG Steam Flow per SG. lbm/sec	2,097.2	2,073.1	2.118.4
Steam Pressure, psia	900 (2)	840.5	831.5
SG Total Mass, lbm	176,950[1]	175587.3[4]	174,030[4]
SG Liquid Mass (lbm)	163,844	Not included	159,158

[1] Does not include mass in steam lines from SG to MSIV (approximately 2500 lbm)
[2] This value of Core Power used for analysis purposes only as described in Section 3.3.1
[3] Based on Actual Pump Performance
[4] Includes mass in steam lines from SG to MSIV (approximately 2500 lbm)

b. The FSAR Chapter 15 analyses do not specifically use the nominal operating points. The FSAR Chapter 15 analyses are performed in a bounding fashion; that is, they are typically initiated at the extremes of the Limiting Conditions for Operation (LCOs). Section 3.10.2 and 3.10.3 of the original submittal describe that the FSAR Chapter 15 accidents use a power level equal to or greater than 102% power for events in which a higher power is more adverse. Since the power level and initial conditions used bound the nominal operating point, the corresponding consequences also are bounding.

12. Please provide a quantitative discussion confirming that the Low Temperature Overpressure Protection Relief valves have adequate relief capacity to remove the additional decay heat generated by the 1.5% power uprate such that there is no increase in peak pressure for this transient. Include a discussion of the NRC approved methodology used to perform this analysis.

Response:

The Low Temperature Overpressure Protection (LTOP) system as it relates to adequate relief capacity is described in UFSAR Appendix 5.2B. The Waterford Safety Evaluation Report (NUREG-0787, "Safety Evaluation Report related to the operation of Waterford 3 Steam Electric Station, Unit No. 3," Docket No. 50-382, July 1981) Section 5.2.2.2 describes the approval of the LTOP system.

The limiting transients with respect to Reactor Coolant System (RCS) pressurization are:

1. An inadvertent Safety Injection (SI) actuation (mass input).
2. A Reactor Coolant Pump (RCP) start when a positive Steam Generator (SG) to Reactor Vessel (RV) ΔT exists (energy input).

These transients were determined to be most limiting by conservative analyses which maximize mass and energy additions to a water solid RCS as a function of time.

The increase in decay heat generated by the 1.5% power uprate will not change the conclusions of these analyses because, as seen in UFSAR Figure 5.2B-1 and Figure 5.2B-2, the mass input and energy input transients bound the loss of decay heat removal transient. In addition, for the limiting mass and energy input transients, the Residual Heat Removal System will continue to remove the higher decay heat load. The results of the mass input and energy input transients (UFSAR Figure 5.2B-3) demonstrate that sufficient margin is available between the peak pressure achieved and the Technical Specification (TS) 3.4.8.1 Pressure -Temperature (PT) limits.

The additional decay heat generated by the 1.5% power uprate does not increase the peak pressure for the limiting LTOP transients, therefore the LTOP relief capacity remains adequate.

13. The application states in Sections 3.5.9 and 3.5.10 that Core Protection Calculator System (CPCS) and Core Operating Limit Supervisory System (COLSS) will require changes to "constants" to account for the 1.5% power uprate.

a. Please discuss the constants requiring revision and the NRC approved methodology to be used to calculate the updated constants for a 1.5% power uprate.

b. For a situation where the Caldon LEFM CheckPlus meter becomes inoperable, please discuss the impact of continued use of the 1.5% power uprate constants for the lower rated thermal power (RTP) level. If the constants for these systems need to be converted back to the current 100% RTP values, please discuss how this will be accomplished, including the time needed to revise these constants and any impacts on the ability of CPCS and COLSS to continue to perform their design basis functions.

Response:

a. Two types of CPCS constants will be changed for the Appendix K uprate:

- Those that are based on the power level, core average linear heat rate, and core average heat flux, and
- Those that must be changed in order to maintain the same effective values for the transient analysis.

Although addressable constants could be adjusted to compensate for the increased power, it was decided to modify the CPCS constants that are based on the power level, core average linear heat rate, and core average heat flux. These constants are:

QAVG – core average heat flux (BTU/sec-ft^2)
QHOT – hot pin heat flux (BTU/sec-ft^2, adjusted for fraction of heat generated in fuel)
LPDLM – local power density trip setpoint (% of rated power)
CLPD – local power density pre-trip setpoint conversion factor (from kw/ft to % of rated power)
LPDMAX – local power density upper limit (% of rated power)
KAD3 – scaling factor for local power density margin analog output (% of rated power to counts)

The methodology being used to calculate the values for these constants is standard physics and thermal hydraulics analysis. This analysis is done using the new power level along with the number of fuel rods, the active fuel length, the fuel rod diameter, and the centerline melt design limit, consistent with past practice and NRC approved methods.

The only other CPCS constants that will be changed for the Appendix K uprate are the variable overpower trip (VOPT) constants. These constants, which are defined relative to the rated thermal power value, will be changed so that the VOPT will behave the same in absolute power units as prior to the uprate. Therefore, the transient analyses that rely on the VOPT will not be impacted by the uprate. The VOPT is described in CEN-308-P-A and CEN-310-P-A.

b. During a limited period after the failure of the LEFM system, the LEFM-based calibration constants for the venturi-based feedwater and steam flows remain applicable. Thus, the calculations of venturi-based power levels maintain an accuracy consistent with the 1.5% power uprate. During this period no penalties or other modifications to either COLSS or CPCS are necessary.

   Beyond the period of complete validity of the calibration constants, an appropriate penalty is applied to the calculated power, thus increasing the indicated power relative to the raw calculated value. This penalty obviates changes to the COLSS constants by ensuring operation at an actual power level below the licensed power limit.

   The CPC margins to trip are based on the power periodically calibrated to the indicated secondary calorimetric power. Since this power includes the appropriate penalty when the LEFM system is out of service, these margins to trip will remain conservative.

14. The licensee refers to a previously proposed 8% power uprate and associated analyses in certain sections of the submittal to justify the 1.5% power uprate. The 8% power uprate analyses are being used to justify that the 1.5% uprate is bounded (by the 8 percent power uprate analyses) for the following topics:

- Shutdown Cooling System
- Emergency Feedwater System
- Condensate Storage Pool/Wet Cooling Tower Basin Requirements

Please provide the following information:

a. References to submitted analyses or NRC Safety Evaluation Report that documents staff review of the 8% power uprate.
b. If the documents requested above do not exist, please provide quantitative results demonstrating that these systems continue to meet their functional design requirements and acceptance criteria at the 1.5% power uprate conditions.

Response:

a. The 8% power uprate NRC Branch Technical Position (BTP) RSB 5-1 analysis has not been submitted to the NRC for approval.

b. The BTP RSB 5-1 analysis was performed for 108% of rated thermal power (3661 MWt). The BTP RSB 5-1 natural circulation cooldown analysis was performed for two different single failures. The failures of an Atmospheric Dump Valve (ADV) and of an Emergency Diesel Generator (EDG) were evaluated as the two limiting single failures. The ADV failure results in a longer time to reach Shutdown Cooling (SDC) conditions since only one steam generator is used to cool the RCS. However, cooldown to 200 °F occurs using both SDC trains. The EDG failure results in the loss of one SDC train. Thus, the cooldown to SDC

entry conditions uses both steam generators, but only one SDC train to reach 200 °F. The event results are listed in the table below.

The analysis demonstrated that the Shutdown Cooling System (SDC) was capable of reducing the RCS temperature to 200 °F. The analysis also demonstrated that the Emergency Feedwater System (EFW) supplied adequate flow and that sufficient inventory was available in the Condensate Storage Pool (CSP) and one Wet Cooling Tower (WCT) basin (total available inventory of 344,000 gallons) to meet the Nuclear Steam Supply System (NSSS) heat removal requirements.

The times in the table below for the 100% power (3390 MWt) and the 108% power (3661 MWt) cases for the failed ADV are the same. This is because the EFW flow rate, ADV capacity, and the SDC capacity is sufficient to maintain the same cool down rate for both power levels.

Section 3.5.4 of the original submittal also addresses the Technical Specification (TS) Surveillance Requirements 4.9.8.1 and 4.9.8.2. This section refers to an analysis performed for the 8% power uprate. The reference to the 8% power uprate is in error and should have stated that an evaluation was performed for the 1.5% power uprate and determined that the SDC flow versus time limits remain unchanged. The 1.5% power uprate evaluation used the same methods and information that were used in the bases calculations but increased the rated thermal power level to determine acceptable results.

Table for Question 14

Summary of Results of BTP RSB 5-1 Analyses

	Time (including initial 4 hour hold)				
Case	To SDC Entry (hours)	SDC to 200 °F (hours)	Total Event (hours)	Cumulative EFW Usage (gallons)	[2]Total Heat Removed by SDC System (Btu)
NC Cooldown with Failed ADV @ 100% rated power	25.1	3.0	28.1	284,000	3.42×10^8
NC Cooldown with Failed ADV @ 108% rated power	25.1	3.0	28.1	303,000	3.55×10^8
Natural Circulation Cooldown with Failed EDG @ 108% rated power	9.6	[1]12.4	[1]22.0	172,000	[1]1.03×10^9

[1] CCW water temperature initially 115 °F, then reduced to 105 °F when the total heat load (decay heat, pump heat, plus sensible heat) reaches 100×10^6 btu/hr, then reduced to 100 °F when the total heat load reaches 80×10^6 btu/hr.

[2] Total heat removed by the SDC system in going from SDC initiation temperature to 200 °F.

15. Please discuss the impacts of the changes in Steam Generator (SG) thermal-hydraulic performance (circulation ratio/bundle liquid flow, damping factor, SG pressure drop, and moisture carryover) and the increase in primary to secondary system pressure differential on the FSAR Chapter 15 accident and transient safety analyses.

Response:

Sections 3.10.2 and 3.10.3 of the original submittal describe that the FSAR Chapter 15 accidents use a power level equal to or greater than 102% of rated thermal power (3458 MWt) for events in which a higher power is more adverse. The FSAR Chapter 15 analyses are performed in a bounding fashion; that is, they are typically initiated at the extremes of the Limiting Conditions for Operation (LCOs). Since the analyses were already performed at a power level greater than or equal to 102% with the corresponding Steam Generator (SG) thermal hydraulic characteristics, the Appendix K power uprate has no affect on the accident analyses.

16. The Main Steam Isolation Valve (MSIV) design is based on the full design pressure differential across the valves at a Rated Thermal Power of 102% of 3,390 MWt.

a. Please confirm that the pressure differential across the valves assuming the 1.5% power uprate operating conditions remains bounded by the assumptions in the original design analyses and operating conditions.
b. Please discuss the plant operating or accident conditions that result in the maximum expected differential pressure for MSIV closure.

Response:

a. Section 3.7.1.3 of the original submittal stated that the MSIV design is based upon 102% rated thermal power (3458 MWt). Since the MSIV design analyses were already performed at a power level greater than or equal to 102% of rated thermal power, the 1.5% Appendix K uprate is bounded. The maximum differential pressure assumed across the MSIV (1117.6 psid) bounds the 1.5% power uprate operating conditions.

b. The limiting events with respect to maximum differential pressure would be the Main Steam Line Break (MSLB) or the Feedwater Line Break (FWLB). The MSIV maximum differential pressure used in the MSIV closure analysis corresponds to the second steam generator safety valve setpoint plus a 3% tolerance (1117.6 psid). The MSLB and FWLB analyses validate that this differential pressure bounds the potential accident conditions even with an initial power level of 102% rated thermal power. Thus, the proposed Appendix K power uprate power of 101.5% (3441 MWt) of the original rated thermal power will remain bounded.

17. In Section 3.10.3 - Non-LOCA/Transient Analyses, the third paragraph includes the following statement: "..., there are adverse changes in the docketed results of the Non-LOCA transient analyses." Based on the discussion which preceded this statement, it appears that the licensee meant to say that "..., there are no adverse changes in the docketed results of the Non-LOCA transient analyses." Please clarify this statement.

Response:

Section 3.10.3 was intended to say, "..., there are no adverse changes in the docketed results of the non-LOCA transient analyses."

18. FSAR Section 15.4.1.1, Uncontrolled CEA Withdrawal from Subcritical Conditions, states that a 5.41% analytical limit is used for the Logarithmic Power Level - High trip setpoint. Section 3.10.1 of the original submittal states that the analytical limit for this setpoint is 4.4% of Rated Thermal Power. Please clarify this discrepancy and discuss any impact resulting from a change in reactor trip timing.

Response:

The current UFSAR analysis was based upon the Cycle 9 analysis. The Cycle 9 analysis used a high log power trip setpoint of 5.41% rated thermal power. The Cycle 10 (to present) analyses have used a trip setpoint of 4.4% rated thermal power. The actual high log power trip setpoint is less than or equal to 0.257% rated thermal power [Technical Specification 2.2.1]. The total setpoint plus uncertainty associated with these instruments is less than 1% rated thermal power. In addition, accounting for potential decalibration effects would increase the analysis setpoint to a value of 2.08% rated thermal power. The trip setpoints modeled in the Cycle 9 and Cycle 10 analyses are conservative and bounding with respect to this value. Thus, the accident consequences given in the FSAR remain bounding and applicable to the 1.5% power uprate. Additional details of this analysis are being provided to the NRC staff to support their review of an amendment request related to part-length CEA replacement.

19. In Section 3.6.4 it is stated that the fluence value for the projected 20 effective full power years of operation (EFPYs) was derived from the results of capsule W-97 which was removed about 1991. The evaluation report (BAW-2177) states that the cross sections used were from an early version of the BUGLE set. Those cross sections result in non-conservative flux and fluence evaluations because the non-conservative ENDF/B-IV cross section data was used. This fact combined with the power uprate raise a question about the validity of the extrapolation fluence value to 20 EFPYs. It is also stated that another surveillance capsule will be removed at the end of the current fuel cycle (11) when the reactor will have accumulated 14 EFPYs. In this context please consider the following:

   a. Describe how the projected fluence values used for the 20 EFPY pressure temperature (PT) curves and cold overpressure protection limits provide sufficient margin when the known nonconservatisms with the cross sections are considered?
   b. If a new capsule is to be removed in the 2002 outage (the results of which will be available in 2003) why do you need to extrapolate to 20 EFPYs while you have the facility to update the fluence for the cycle 13 refueling (Which will occur much earlier than 20 EFPYs)?

Response:

a. Note that while the projected fluence values were calculated based on a 20-year assumption, the actual allowed limits approved for the Waterford 3 Technical Specifications are limited to 16 effective full power years (EFPY) to address the nonconservatisms in the fluence. This was noted in both Amendment 160 to the Waterford 3 Operating License and in the Waterford 3 letter dated April 13, 2001 requesting approval of the use of a revised cooldown curve. This request was recently approved in Amendment 177. In the power uprate request submittal, Waterford 3 had intended that the P/T curves (based on a projected fluence for 20 EFPY but conservatively restricted to apply only until 16 EFPY) still had adequate margin considering the impact of the uprate to be valid for the current 16 EFPY. As noted in the uprate submittal, the expected increase in the fluence at 16 EFPY due to the uprate will be on the order of 1%. It was also noted that we expect to see the effects of our core management changes result in a lower fluence than had been projected when we pull the next sample in our upcoming (Spring 2002) refueling outage.

b. Waterford 3 is not seeking approval for the full 20 EFPY as a part of the power uprate request. That value was referenced only as the basis for the existing calculation that forms the basis for our current technical specifications (TS 3.4.8.1). The power uprate request only intended to note that the existing approved curves, valid through 16 EFPY, still included sufficient margin to accommodate the uprate and permit operation for Cycle 12.

20. The recent experience from Calvert Cliffs has shown that the cladding corrosion is worse in high burnup regime and is consistently underestimated by the CENP corrosion model. Additionally, the power uprate may increase cladding corrosion levels. Please provide cladding corrosion predictions for power uprate conditions and assess the potential impact in fuel operation for Waterford Unit 3.

Response:

It is recognized that recent high duty fuel performance data from Calvert Cliffs, Waterford 3, and Palo Verde have indicated that OPTIN cladding corrosion for some high duty fuel rods is more adverse than originally expected. Increased corrosion and limited oxide spalling have been observed in recent high duty fuel inspections at Calvert Cliffs and Palo Verde and in past high burnup test assemblies at Calvert Cliffs, Palo Verde, and Waterford 3. Increased core crudding has also been observed in poolside measurements for high duty fuel at Palo Verde.

As a result of these observations, preliminary models for predicting corrosion, the threshold for spalling, and the steaming rates associated with crudding, have been developed by W CENP. These models consider the above-mentioned developments at CE plants and have been applied as needed to assess high duty operation of operating W CENP plants.

W CENP has assessed the corrosion performance of Waterford 3 under power uprated conditions with the new models, and has established and applied additional fuel management guidelines for corrosion to Waterford 3 on a cycle-specific basis,

beginning with the first uprated cycle. These fuel management guidelines limit the maximum oxide thickness, the fuel duty, and the maximum radial power peaking factor (Fr) to limit steaming rate and core crudding. The preliminary corrosion models discussed above were applied to assess conformance with these guidelines.

The corrosion performance assessment included analysis of select limiting power fuel rods (including assembly peripheral rods) from Waterford 3 uprated fuel management depletions. The fuel management depletions were constructed specifically to model the more adverse expected core conditions during uprated power operation. The preliminary models developed based on the high duty corrosion performance data were applied and show that predicted maximum oxide thickness is less than 100 microns. A 100-micron limit has been imposed on other fuel vendors/cladding by the NRC and the limit is expected to be imposed on OPTIN cladding for low duty high burnup fuel when CENPD-388 is approved.

In summary, an assessment of the planned power uprate core with the new fuel management guidelines, utilizing the preliminary corrosion models which consider the recent experiences at CE plants, shows acceptable corrosion performance for the planned Waterford 3 1.5% power uprate core.

21. With respect to the impacts of the proposed power uprate on the nuclear fuel core design, thermal-hydraulic design and fuel rod design analyses, please:

    a. Provide a listing of the NRC approved codes and methodologies to be used for the fuel core design process discussed in section 3.13.1 of the submittal.
    b. Confirm that all parameters and assumptions to be used for analyses described in sections 3.13.1 through 3.13.3 remain within any code limitations or restrictions.

Response:

The table below provides a compilation of the nuclear fuel core design, thermal hydraulic, and fuel rod design codes and methodologies employed in support of the Waterford 3 power uprate amendment request.

The Westinghouse Quality Assurance process requires that each analyst confirm that the limitations and/or constraints (L/Cs) imposed by the Nuclear Regulatory Commission (NRC) via its Safety Evaluation Reports (SERs) are satisfied. Thus all of the analyses employed in support of the Waterford 3 power uprate request satisfy applicable SER code limitations or restrictions.

Table for Question 21
Nuclear Fuel Core, Thermal Hydraulics, and Fuel Rod Design Methodologies

Nuclear Design Methodologies

1. CENPD-153-P, Rev 1, "INCA/CECOR Power Peaking Uncertainty," September 1980.

2. CENPD-188-A, "HERMITE A Multi-Dimensional Space-Time Kinetics Code for PWR Transients", July 1976.
3. CENPD-266-P-A, "The ROCS and DIT Computer Codes for Nuclear Design," August 1983.

Fuel Assembly Design Methodologies

1. CENPD-178-P, Rev. 1-P, "Structural Analysis of Fuel Assemblies for Combined Seismic and Loss of Coolant Accident Loading", August 1981.

Fuel Thermal Hydraulic Design Methodologies

1. CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs 1 and 2", December 1981.
2. CEN-214(A)-P, "CETOP-D Code Structure and Modeling Methods for Arkansas Nuclear One – Unit 2", July 1982.
3. CEN-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core", April 1986.
4. CENPD-162-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution", September 1976.
5. CENPD-206-P-A, "TORC Code, Verification and Simplified Modeling Methods", June 1981.
6. CENPD-207-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 2, Non-Uniform Axial Power Distribution", December, 1984.
7. MacBeth, MacBeth Critical Heat Flux Correlation

Fuel Rod Design Methodologies

1. CEN-161(B)-P-A, "Improvement to Fuel Evaluation Model", August 1989.
2. CEN-193(B)-P, "Partial Response to NRC Questions [Nos. 8, 10-131 on CEN-161(B)-P, Improvements to Fuel Evaluation Model."
3. CEN-193(B)-P, Supplement 1-P, "Partial Response to NRC Questions [Nos. 7 and 9] on CEN-161(B)-P, Improvements to Fuel Evaluation Model."
4. CEN-193(B)-P, Supplement 2-P, "Partial Response to NRC Questions [Nos. 1 - 6] on CEN-161(B)-P, Improvements to Fuel Evaluation Model."
5. CEN-205(B)-P, "Response to NRC Questions on FATES-3 and the Calvert Cliffs 1 Cycle 6 Reload."
6. CEN-220(B)-P, "Supplemental Information on FATES-3 Stored Energy Conservatism."
7. CEN-161(B)-P, Supplement 1-P-A, "Improvement to Fuel Evaluation Model", March 1992.
8. CEN-345(B)-P, "Response to Questions on FATES3B."
9. CEN-161(B)-P, Supplement 1-P, "Improvements to Fuel Evaluation Model," April 1986.

10. CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure", May 1990.
11. CEN-386-P-A, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/KgU for Combustion Engineering 16x16 PWR Fuel", August 1992.
12. CEN-136, "High Temperature Properties of Zircaloy and UO2 for Use in LOCA Evaluation Models," July 1974.
13. CENPD-139-P-A, "CE Fuel Evaluation Model Topical Report," April 1975.
14. CENPD-185-P-A, "LOCA Rupture Behavior of 16x16 Zircaloy Cladding," February 19, 1976.
15. CENPD-225-P-A, "Fuel and Poison Rod Bowing," June 1983.
16. CENPD-269-P, Rev. 1-P, "Extended Burnup Operation of Combustion Engineering PWR Fuel," July 1984.
17. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.

22. The licensee reported that the existing UFSAR Chapter 15 Non-LOCA/transient analyses of record bound plant operation at the proposed power uprate level, and therefore, reanalysis was not required. For the UFSAR Chapter 15 accident and transient analyses:

    a. Confirm that the analyses of record either have been previously approved by the NRC or were conducted using methods or processes that were previously approved by the NRC. Provide a reference to the NRC's previous approvals.
    b. Confirm that the analyses as described in the UFSAR, Revision 11, dated May 2001 are the current analyses of record. For those analyses which are not, please provide the following:

        i. Major assumptions used in the re-analyses. Provide justification for any assumptions which deviate from that used in the UFSAR analyses.
        ii. Describe methods and computer codes used for the re-analyses and confirm that they have been previously approved by the NRC staff. Provide justification for any changes in methodology from the existing analyses.
        iii. Provide the results of the re-analyses including primary and secondary system peak pressure, minimum DNBR, Peak Linear Heat Generation Rate, and/or amount of fuel failed.
    c. Confirm that bounding event determinations continue to be valid.

Response:

a. The following references provide the Safety Evaluation Reports that demonstrated the NRC approval. NUREG-0787 Supplements 6-10 did not specifically address the accident analyses but were included for completeness. NUREG-0787 Table 15.2 lists the topical reports for the codes used in the safety analyses. The analyses of record were either previously approved by the NRC or were conducted using methods and/or codes that were previously approved by the NRC (see list in response to question 21).

1. NUREG-0787, "Safety Evaluation Report related to the operation of Waterford 3 Steam Electric Station, Unit No. 3," Docket No. 50-382, July 1981, including the following supplements:
   Supplement #1 dated October 1981
   Supplement #2 dated January 1982
   Supplement #3 dated April 1982
   Supplement #4 dated October 1982
   Supplement #5 dated June 1983
   Supplement #6 dated June 1984
   Supplement #7 dated September 1984
   Supplement #8 dated September 1984
   Supplement #9 dated December 1984
   Supplement #10 dated March 1985.

2. NRC Safety Evaluation Report dated January 16, 1987, "Reload Analysis Report for Cycle 2 at Waterford 3."
3. NRC Safety Evaluation Report dated September 15, 1992, "Reload Analysis Report – Waterford Steam Electric Station, Unit No. 3 (TAC No. M84175)", Cycle 6.
4. NRC Safety Evaluation Report dated March 7, 2000, "Waterford Steam Electric Station, Unit 3 – Issuance of Amendment Re: Small Break Loss-of-Coolant Accident Model (TAC No. MA3271)."
5. NRC Safety Evaluation Report, "Acceptance for Referencing of the Topical Report CENPD-137(P), Supplement 2, 'Calculative Methods for the C-E Small Break LOCA Evaluation Model' (TAC No. M89400)."

b. The analyses contained in UFSAR Revision 11 are the current analyses of record.

c. The bounding event determinations remain bounding.

23. Certain transients as described in the UFSAR show MDNBR results of 1.19, which does not meet the acceptance criteria listed in Table 3.10.3-1 of the licensee submittal for MDNBR $\geq$ 1.26. Please discuss this discrepancy. If these are not the current analyses of record, then please provide the information requested in item 22.b.i-iii above.

Response:

In general, the UFSAR accident analyses that still list the low DNBR setpoint as 1.19 were not updated because the low DNBR setpoint change did not adversely affect the overall event consequences with respect to the particular acceptance criteria. The table below lists each of the UFSAR sections and the events for which the low DNBR setpoint of 1.19 is still present. The table also lists a description of the reason the event is still acceptable.

Table for Question 23
Events with DNBR Setpoint not ≥ 1.26

UFSAR Section	Event	Description of Reason
Table 15.0-3	Reactor Protective System Trips Used in the Safety Analyses	UFSAR Table 15.0-3 low DNBR value was not updated but a note was added that describes that the Cycle 2 value to present is 1.26. Any Chapter 15 event with a reactor trip on low DNBR of 1.19 will actually result in a reactor trip on low DNBR of 1.26 due to the CPC low DNBR setpoint change.
15.1.1.3 Table 15.1-1	Increased Main Steam Flow	The event was initially performed for a DNBR limit of 1.19. The CPCs initiated a reactor trip on low DNBR such that the DNBR limit was not exceeded. The current CPC low DNBR setpoint is 1.26. The transient behavior tripping from 1.26 or 1.19 will not change the thermal degradation as it corresponds to the safety limit not being exceeded. Since the safety limit is still protected by the CPC trip, no reanalysis was required. UFSAR information was not updated because the event remained bounding.
15.2.1.4 Table 15.2-3	Loss of Normal AC Power	Same reason as for Increased Main Steam Flow event above.
15.3.3.1 Table 15.3-4a Table 15.3-6	Single Reactor Coolant Pump Shaft Seizure / Sheared Shaft	The DNBR limit of 1.19 is still contained in the barrier performance section and the radiological consequences section. The fuel performance section uses a value of 1.26. The barrier performance supplies the steam releases to the atmosphere that is not adversely affected by the low DNBR setpoint change. The radiological consequences section determines the offsite doses and are not affected low DNBR setpoint change due to the fact that the fuel performance validates the fuel failure that is used in the dose calculations.
15.4.3.1.1	Inadvertent Loading of a Fuel Assembly inot the Improper Position	Refer to Question 28
15.4.3.2	Control Element Assembly (CEA) Ejection	The DNBR limit of 1.19 is still contained in the radiological consequences section. The fuel performance section uses a value of 1.26. The radiological consequences section determines the offsite doses and are not affected by the low DNBR setpoint change due to the fact that the fuel performance validates the fuel failure that is used in the dose calculations.
15.4.4.2.5 Table 15.4-7	Uncontrolled CEA Withdrawal at Power	Same reason as for Increased Main Steam Flow event above.
15.4.1.4 Table 15.4-15	Control Element Assembly Misoperation	Same reason as for Increased Main Steam Flow event above. Note - A separate submittal is currently under review by the NRC Staff to replace the Part-Length CEAs (PLCEA). Thus, after approval of that change, the PLCEA drop event will no longer exist.

24. Please provide a detailed discussion regarding the impact of the power uprate conditions on the Uncontrolled CEA Withdrawal at Subcritical analysis. Section 15.4.1.1 of the UFSAR states that the Linear Heat Generation Rate limit is exceeded and relies on a detailed deposited energy calculation to demonstrate that centerline fuel temperature remains below melt temperature. Please provide the analysis results and the fuel melt temperature for the fuel burnup assumed in the analysis.

Response:

The 1.5% power does not change the subcritical CEA Withdrawal consequences because the event is initiated at subcritical conditions. This event was reanalyzed due to other changes (Part-Length CEA replacement) that are also being implemented for Cycle 12. The method of analysis is the same as for previous cycles and has been reviewed and approved by the NRC. A more detailed description of this event is provided in the submittal related to Part-Length CEA replacement.

Per the Waterford 3 Cycle 2 Safety Evaluation Report (SER) [NRC SER dated January 16, 1987, "Reload Analysis Report for Cycle 2 at Waterford 3."] and the Standard Review Plan (SRP) [NUREG-0800], the subcritical CEA withdrawal acceptance criteria are:

1. The thermal margin limits for DNBR are met (DNBR≥ 1.26).
2. Fuel centerline temperatures do not exceed the melting point (T≤ 4900 °F).
3. The RCS pressure is below the emergency limit (P≤ 2750 psia).

The event acceptance criteria continue to be met for this event and the approved methodology was not affected by the power urprate.

25. For the Uncontrolled CEA Withdrawal at Power transient, the licensee states that the reactor trip credited for this event is the Variable Over Power trip. Section 15.4.1.3 of the UFSAR states that a Low DNBR Trip is credited. Please discuss this discrepancy and its impacts.

Response:

UFSAR Section 15.4.1.3.2 states that the uncontrolled CEA withdrawal transient is terminated by one of the following:

a. achieving a stable, steady state condition
b. high power level trip
c. high pressurizer pressure trip
d. low DNBR trip
e. high local power density trip

The UFSAR Section 15.4.1.3 limiting transient is terminated by the low DNBR trip. Table 3.10.3-1 should have stated:

> The CEA withdrawal at power event actuates a DNBR trip just prior to the high pressurizer pressure trip and the high power level trips. The DNBR trip setpoint is not affected by the power uprate, thus the event consequences remain bounding.

26. Please provide additional discussion and detail regarding the impact of the power uprate conditions on the Control Element Assembly Misoperation event (UFSAR Section 15.4.1.4), specifically regarding the ratio of the available thermal margin at the start of the event to the available thermal margin at the termination of the event.

    Response:

    All of the CEA Misoperation events of UFSAR Section 15.4.1.4 are classified as Anticipated Operational Occurrences (AOOs). As such, the combination of the initial thermal margin preserved by the Limiting Conditions for Operation and the action, if any, of the Reactor Protection System must be sufficient to ensure that a specified acceptable fuel design limit (SAFDL) violation will not occur due to these transients.

    In all cases, COLSS and CPCs will use the actual core power in the online calculations being performed. The initial margins set aside for these events are effectively the ratios of the margin prior to and at the worst point during the event. Since the summation of rated thermal power plus power measurement uncertainty is unchanged by the 1.5% uprate, the maximum core power level used in the plant safety analysis is unchanged. Therefore power uprate does not have a direct impact on these events.

27. Please provide the results for the analysis of record for the Uncontrolled Control Element Assembly (CEA) Withdrawal from Subcritical - Mode 3, 4 and 5, All Full Length CEAs on the Bottom (UFSAR Section 15.4.1.7) event. The results for this analysis are not shown in the UFSAR.

    Response:

    The withdrawal of control element assemblies (CEAs) from subcritical conditions with all full length CEAs on the bottom adds reactivity to the reactor core causing the core power level to increase. The withdrawal of CEAs also produces a time dependent redistribution of core power. As the power level continues to increase a trip is generated by the Core Protection Calculators (CPCs) when the CPC bypass is automatically removed at 2.4 x 10^{-4}% of rated thermal power.

    Due to the prompt CPC trip at 2.4 x 10^{-4}% of rated thermal power, the consequences of these events do not result in DNBR or fuel centerline melt Specified Acceptable Fuel Design Limits (SAFDLs) being approached.

    In addition, Waterford 3 provides adequate assurance (through administrative controls) that, with the Control Element Drive Mechanism powered and the CEAs capable of being withdrawn, criticality is not achievable on the withdrawal of the

Shutdown Banks (excluding Technical Specification Special Test Exceptions 3.10.1 and 3.10.3). With these controls in place, an uncontrolled CEA withdrawal event would not result in an approach toward the SAFDLs.

Thus, no UFSAR results are shown because the event has limited consequences that are bounded by the subcritical CEA withdrawal event (CEA regulating bank withdrawal).

28. For the Inadvertent Loading of a Fuel Assembly into an Improper Position event (UFSAR Section 15.4.3.1), please discuss in greater detail the statement that the consequences of these misloads are limited by the initial DNBR margin. Also, please address the acceptance criteria for this event as listed in NUREG-0800, Standard Review Plan, Section 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position."

Response:

The planned 1.5% power uprate will have no impact on the consequences of the Inadvertent Fuel Loading into an Improper Position event because of the following reasons:

a. UFSAR Section 15.4.3.1 reported a minimum DNBR for the Inadvertent Fuel Misloading Accident that is within the SRP Acceptance Criteria (i.e., fuel failure limits not exceeded). The DNBR calculations reported in UFSAR Section 15.4.3.1 were performed at a core average heat flux of 0.187 MBTU/hr-ft^2. The core average heat flux corresponding to the uprated core (conservatively evaluated at 3448 MWt) is expected to be 0.182 MBTU/hr-ft^2. Thus the heat flux used in the FSAR calculation is sufficient to bound the uprate including allowances for power measurement uncertainty.
b. The 1.5% power uprate will not significantly affect either the power distributions or the relative fuel assembly reactivities. Furthermore, neither the burnable absorber loading nor the number of burnable absorber rods per assembly basis is expected to change. Thus the relative reactivities between a shimmed and unshimmed assembly will not increase.

29. In Section 3.6.2.2.1, with respect to Control Element Assembly Drop Time Analyses, the licensee states that, "Uprate to 3441 MWt will slightly increase the power level in leading rodded fuel assemblies, but will not change the burnup levels of those fuel assemblies, since the excess reactivity will be depleted faster." Please clarify this statement, it is not clear what this means.

Response:

This statement was included to note that the planned 1.5% power uprate will not significantly increase the discharge burnup for the fuel assemblies. Thus the range of burnups and exposures of those fuel assemblies containing CEAs will remain within the current experience base. Since neither the assembly exposure nor the core operating conditions are expected to change with this uprate, it is therefore concluded that the control rod drop times will not be affected by the uprate.

30. The core protection calculator system (CPCS) within the reactor protection system initiates reactor trips based on low DNBR and high local power density.

    a. Describe how the CPCS DNBR and VOPT trip functions are modeled in the Chapter 15 safety analyses of the design basis transients and accidents.
    b. Describe how the proposed power uprate with the reduced power measurement uncertainty affect the CPCS and the safety analyses.

Response:

a. The CPC DNBR and VOPT trip functions modeling differs depending upon which analysis method is employed. The modeling methods are consistent with the NRC approved methodologies described in the response to question 22.a above. The discussion below clarifies the application for the modeling of these functions.

<u>CPC VOPT Function</u>

The Nuclear Steam Supply System (NSSS) parameters are modeled using the CESEC computer code [UFSAR Section 15.0] to determine the transient Reactor Coolant System (RCS) response (i.e., RCS power, pressure and temperatures, etc.). The VOPT may be credited in these CESEC simulations in one of two ways. The first is by using a conservative value of the core power level at which trip will occur as a simple setpoint (e.g., hot zero power CEA withdrawal assumes trip occurs at the VOPT floor setpoint).

The second method of crediting the CPC VOPT is used when it is necessary to model in more detail the dynamic response of the VOPT. When this is necessary, the NSSS response as modeled by CESEC is input into the CPC FORTRAN SIMULATION code. This code provides a high fidelity prediction of when the CPC VOPT will occur. That time is then used in a second transient simulation which initiates a reactor trip in CESEC at that time (instrument response time and CEA holding coil decay time are included in the CEA scram curve).

Since the CPC VOPT constants will be adjusted as described in TSCR 3.1.3.1, the trip timing will remain the same and the accident consequences will remain the same. Thus, the Appendix K uprate will have no affect on the results.

<u>CPC DNBR Trip Function</u>

The CPC DNBR trip function is verified to be conservative by ensuring that the outputs of CPC provide conservative thermal margin when compared against the actual NSSS changes occurring during system transients. In performing this verification, the CESEC and CETOP/TORC computer codes [UFSAR Chapter 15.0] are used to perform the simulations of the NSSS transients and calculate the DNBR/thermal margin using the same CE 1 correlation as in CPC. The CPC constants are adjusted as necessary prior to cycle operation to ensure that the CPC DNBR calculations are conservative.

As the maximum power level considered in the safety analysis (rated thermal power plus uncertainty) is unchanged, there is no impact on the transient simulations. Thus, the Appendix K uprate will have no affect on the results.

b. The CPC constants that are affected by the Appendix K power uprate will be adjusted as described in Sections 3.10.1 and 3.1.3.1 (also, see response to question 21 above). The adjustment of the CPC constants will keep the accident analyses and the trip functions consistent. The accident analyses as described in Sections 3.10.2 and 3.10.3 use a power level equal to or greater than 102% power for events in which a higher power is more adverse. Since the analyses were already performed at a power level greater than or equal to 102%, the Appendix K power uprate has no affect on the accident analyses.

31. 10 CFR 50.46(b)(5) establishes long-term cooling requirements following a loss-of-coolant accident. One aspect of long-term cooling following a loss-of-coolant accident is to ensure boric acid accumulation will not prevent core cooling by applying an acceptable evaluation model (EM) to analysis of boric acid accumulation and to determination of the time available for switchover to hot leg injection. If you have <u>not</u> reanalyzed these topics in support of your power uprate request <u>and</u> you have documented application of a staff-approved EM to these topics, then please provide references to this documentation. If you <u>have</u> reanalyzed these topics in support of your power uprate request <u>or</u> you do not have a staff-approved EM, then please supply a complete description of your methodology. If you will be referencing CENPD-254, please describe the volume in which boric acid is assumed to accumulate and provide the bases for using those volumes.

Response:

UFSAR Section 6.3.3.4 describes the post-LOCA long-term cooling (LTC) analysis. This analysis was not revised for the 1.5% power uprate. The post-LOCA LTC analysis was performed using the NRC-approved post-LOCA long-term cooling evaluation model [CENPD-254-P-A, "Post-LOCA Long Term Cooling Evaluation Model," June 1980.] and complies with the requirements of 10CFR50, Appendix K. The analysis was performed at 102% of rated thermal power, which bounds the requested power uprate condition.

The inner vessel volume in which the boric acid mixes is 11,900 gallons based on the above methodology.

32. To show that the referenced generically approved LOCA analysis methodologies apply specifically to the Waterford-3 plant, provide a statement that Waterford-3 and its vendor have ongoing processes which assure that LOCA analysis input values for peak cladding temperature- sensitive parameters bound the as-operated plant values for those parameters.

Response:

Waterford-3 and Westinghouse have ongoing processes that assure that LOCA analysis input values for peak cladding temperature sensitive parameters bound the as-operated plant values for those parameters.

33. The Waterford-3 power uprate submittal references CENPD-137, Supplement 2-P-A, April 1998, as the generically approved SBLOCA methodology and will become the methodology included in licensing documentation and used to perform the Waterford-3 SBLOCA licensing analyses for the uprated power. The NRC approved CENPD-137, Supplement 2-P-A invoking unique criteria for that specific methodology and the then-existing or then-proposed plant conditions. Provide documentation that demonstrates how all the terms and conditions for use of that methodology have been satisfied and explain how this methodology continues to be applicable to Waterford-3 at the uprated power.

Response:

The Waterford 3 SBLOCA analysis using CENPD-137, Supplement 2-P-A was approved in an NRC Safety Evaluation Report dated March 7, 2000, "Waterford Steam Electric Station, Unit 3 – Issuance of Amendment Re: Small Break Loss-of-Coolant Accident Model (TAC No. MA3271)." This evaluation model was approved in NRC Safety Evaluation Report, "Acceptance for Referencing of the Topical Report CENPD-137(P), Supplement 2, 'Calculative Methods for the C-E Small Break LOCA Evaluation Model' (TAC No. M89400)."

The Waterford 3 NRC approved SBLOCA analysis was performed at 102% of rated thermal power and all of the Safety Evaluation Report required conditions continue to be met for the Appendix K uprate.