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W3F1-2002-0013

February 1, 2002

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

SUBJECT: Waterford Steam Electric Station, Unit 3
Docket No. 50-382
Supplement to Amendment Request NPF-38-234
Replacement of Part-Length Control Element Assemblies

REFERENCES:
1. Entergy letter dated July 9, 2001, TSCR NPF-38-234, "Replacement of Part-Length Control Element Assemblies"
2. Entergy letter dated October 23, 2001, Supplemental Information in Support of TSCR NPF-38-234, "Replacement of Part-Length Control Element Assembly"
3. Entergy letter dated January 17, 2002, Supplement to Amendment Request NPF-38-234, "Replacement of Part-Length Control Element Assemblies"

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Specifications to delete the requirements for the part-length control element assemblies. Additional information was subsequently provided in Reference 2 and Reference 3.

On December 3, 2001 Entergy and members of the NRC staff participated in a call to discuss the additional technical information that would be required in support of the proposed change. As a result of the call, Entergy agreed to provide supplemental information relative to three Final Safety Analysis Report (FSAR) Chapter 15 events and a summary of the analysis results for other Chapter 15 events. The information for the first of the three Chapter 15 events and the summary were provided in Reference 3.

Attachment 1 contains information relative to the second of the three Chapter 15 events discussed during the December 3, 2001 call. Attachment 2 contains the third of the three Chapter 15 events discussed during the December 3, 2001 call. With this letter all information requested by the staff, during the December 3, 2001 call, has been provided.

There are no technical changes proposed in this letter. The original no significant hazards considerations included in reference 1 is not affected by any information contained in this supplemental letter.

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Entergy requests that the effective date for this TS change be within 60 days of approval. Although this request is neither exigent nor emergency, your prompt review and approval prior to the start of RF 11 is requested. Entergy would like to implement this change during RF11 scheduled to start on March 22, 2002.

This letter contains no new commitments. If you have any questions or require additional information, please contact D. Bryan Miller at 504-739-6692.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 1, 2002.

Sincerely,

J.T. Herron
Vice President, Operations
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JTH/DBM/cbh

Attachments:
1. Steam Line Break Events
2. Control Element Assembly Ejection Event

cc: E.W. Merschoff, NRC Region IV
N. Kalyanam, NRC-NRR
J. Smith
N.S. Reynolds
NRC Resident Inspectors Office
Louisiana DEQ/Surveillance Division
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Attachment 1

To

W3F1-2002-0013

Steam Line Break Events

Steam Line Break Events

1. Event Description

The estimated frequency of a steam line break classifies it as a limiting fault.

Post-Trip Steam Line Break Event
The increase in steam flow resulting from a pipe break in the Main Steam System causes an increase in energy removal by the affected steam generator from the Reactor Coolant System (RCS). This results in a reduction of the reactor coolant temperature and pressure. With a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. The reactor trips which may occur due to a steam line break (assuming no loss of offsite power) are the Low Steam Generator Pressure, Core Protection Calculator (CPC) - High Variable Overpower, Low Steam Generator Water Level, High Linear Power Level, Low Primary System Pressure, or High Containment Pressure trip. Where a concurrent loss of offsite power is assumed, a reactor trip may also be caused by a CPC Reactor Coolant Pump (RCP) Speed trip or a low Departure from Nucleate Boiling Ratio (DNBR) trip.

For any reactor trip, the Control Element Assembly (CEA) of maximum worth is assumed to remain fully withdrawn. The low steam generator pressure signal also initiates a Main Steam Isolation Signal (MSIS) which closes the Main Steam Isolation Valves (MSIV) and Main Feedwater Isolation Valves. The cooldown and contraction of the primary coolant empty the pressurizer and initiate a Safety Injection Actuation Signal (SIAS). The emptying of the steam generator associated with the ruptured steam line terminates the cooldown. The boron injection due to the initiation of safety injection system causes a decrease in core reactivity. The failure of a High Pressure Safety Injection (HPSI) pump to start on SIAS has the most adverse effect (higher post-trip fission power due to less boron injection) with respect to core damage and radiological consequences. Therefore, one HPSI pump is conservatively assumed to fail. The operator may initiate plant cooldown by manual control of the steam generator atmospheric dump valves or the steam bypass valves associated with the intact steam generator anytime after reactor trip. In this analysis, it is conservatively assumed that operator action is delayed until 30 minutes after first indication of the event. The plant is then cooled to 350°F at which point shutdown cooling is initiated.

Modes 3 and 4 Post-Trip Steam Line Break
Steam line break events during Mode 3 and 4 operation are analyzed to demonstrate the adequacy of the shutdown margin as specified by Technical Specification 3.1.1.2 to prevent degradation in fuel performance as a result of post trip return to power. Cold leg temperatures above and below 500°F are considered in the following discussions.

The limiting steam line break is a large steam line break inside containment during Mode 3 operation with loss of offsite power in combination with a single failure and the minimum shutdown margin allowed from Technical Specification 3.1.1.2 plus the minimum worth of the most reactive rod. The largest possible steam line break size is the double ended rupture of a steam line upstream of the MSIV. In the Waterford 3 design, the blowdown area for the affected steam generator is 7.88 ft^2, the full cross-

sectional area of the steam line. The blowdown area of the intact steam generator is 3.14 ft^2, the area of the venturi.

For steam line breaks initiated at or below cold leg temperature equal to 500°F, the positive reactivity insertion due to cooldown is less than the subcriticality due to the shutdown margin from Technical Specification 3.1.1.2 plus the minimum worth of the most reactive rod without taking credit for safety injection boron. Since the total positive reactivity insertion without taking credit for safety injection boron is less than the initial subcriticality, there is no post trip return to power. Thus the adequacy of the Technical Specification 3.1.1.2 shutdown margin is demonstrated in this region.

The results of the analyses show that the shutdown margin is sufficiently large to prevent a post trip return to power for any zero power steam line break.

2. Analysis

The reload analyses included the replacement of the Part Length Rods (PLRs) with full length CEAs and the reconfigured banks along with the removal of the 4 finger CEAs (new CEA configuration). The Physics Assessment Checklist (PAC) failed to meet the acceptance criteria [Reference 14 Attachment 2] for the Mode 3 and 4 overall reactivity change and required the Steam Line Break (SLB) post-trip Return to Power (RTP) scenarios [Reference 1 Section 15.1.3.1 and 15.1.3.2] to be reanalyzed for Cycle 12. The Hot Full Power (HFP) and Hot Zero Power (HZP) SLB scenarios passed the PAC assessment but were re-analyzed to provide inputs into the PAC automated process. The Pre-Trip SLB event [Reference 1 Section 15.1.3.3] and the SLB radiological consequences [Reference 1 Section 15.1.3.1.5] continued to be bounded by the previous analyses [Reference 14 Attachment 2] and were not reanalyzed for Cycle 12.

The post-trip SLB Final Safety Analysis Report (FSAR) analysis [Reference 1 Section 15.1.3.1] presented three SLB scenarios [Reference 3 Section 15.3.1]:

1. HFP double ended SLB with concurrent loss of offsite ac power (LOAC).
2. HFP double ended SLB without LOAC.
3. HZP double ended SLB with concurrent LOAC.

The Mode 3 and 4 (subcritical) post-trip SLB FSAR analysis [Reference 1 Section 15.1.3.2] presented two SLB scenarios:

1. Mode 3, double ended SLB with concurrent LOAC.
2. Mode 3, double ended SLB without LOAC.

The Waterford 3 Safety Evaluation Report (SER) [Reference 10 Section 5.1] listed the acceptance criteria for post-trip MSLB event as:

1. Site boundary doses do not exceed 10CFR100 guidelines.
2. Coolable core geometry is maintained.
3. RCS pressure remains below the safety limit.

The SER acceptance criteria are consistent with the Standard Review Plan (SRP) [Reference 4 Section 15.1.5] acceptance criteria.

The current FSAR post-trip SLB analyses were performed for Cycle 2 [Reference 1 Sections 15.1.3.1 and 15.1.3.2]. The Cycle 2 post-trip SLB information was transmitted to the NRC per Reference 6 and approved per Reference 10. The Cycle 12 post-trip SLB analysis was performed using the same NRC approved methodology and computer codes as Cycle 2.

In order to demonstrate acceptable consequences with respect to the acceptance criteria, the nuclear steam supply system (NSSS) response was simulated using the CESEC computer program [Reference 5]. The DNBR thermal margin in the reactor core was simulated using the HRISE computer program and the MacBeth Critical Heat Flux (CHF) correlation [Reference 7 and 8]. The reactivity feedback credit due to the local heatup of the inlet fluid in the hot channel was performed with the HERMITE/TORC code [Reference 9 and 13]. The physics inputs are validated using the HERMITE and ROCS computer codes [Reference 9, 11 and 12].

The major input differences between the Cycle 2 analysis and the Cycle 12 analysis are listed in Table 1.

Table 1. Cycle 2 to Cycle 12 Comparison

PARAMETER	CYCLE 2	CYCLE 12
HZP core inlet temperature (°F)	547	551
Subcritical pressurizer pressure (psia)	2300	1070/1800
HFP Power (Mwt) HZP Power (Mwt)	3478.2 1	3481 34.81
HFP RCS Flow (lbm/hr) HZP RCS Flow (lbm/hr) Subcritical RCS Flow (lbm/hr)	132.2×10^6 129.6×10^6 131.1×10^6	148×10^6 148×10^6 148×10^6
HFP Pressurizer Level (ft^3) HZP Pressurizer Level (ft^3) Subcritical Pressurizer Level (ft^3)	900 9.2 460	844.6 501.3 501.3
Steam Generator (SG) Plugged Tubes	50 per SG	1000 per SG
Pressurizer Level Control System (charging)	Automatic	Manual (off)
HFP Fuel Gap Conductance (Btu/sec-ft^2-°F) HZP Fuel Gap Conductance (Btu/sec-ft^2-°F) Subcritical Fuel Gap Conductance (Btu/sec-ft^2-°F)	1.813 1.813 1.813	0.24 1.812 1.611
HFP Inverse Boron Worth (IBW) (ppm/%$\Delta\rho$) HZP IBW (ppm/%$\Delta\rho$)	-110 -100	Table 2 Table 2
HFP CEA Scram Worth (%$\Delta\rho$) HZP CEA Scram Worth (%$\Delta\rho$)	-7.8 -4.45	-6.65/-6.60 -3.95
HFP Doppler Cooldown Curve HZP Doppler Cooldown Curve	Figure 1 Figure 1	Figure 1 Figure 1
HFP Moderator Cooldown Curve HZP Moderator Cooldown Curve	Figure 2 Figure 2	Figure 2 Figure 2

PARAMETER	CYCLE 2	CYCLE 12
3-D Reactivity Feedback Credits	Approved method	Same method
Core Peaking Factors	Refer to discussion below	

The inverse boron worth (IBW), CEA scram worth, doppler and moderator cooldown curves, 3-D reactivity feedback, and core peaking factors are affected by the new CEA configuration.

The Cycle 2 analysis maximum cold leg temperature was increased for Cycle 12 to add additional conservatism. Cold leg temperature is nominally maintained at 545 °F.

The Cycle 12 analysis performed a parametric on pressurizer pressure. The subcritical post-trip SLB minimum initial pressurizer pressure produced slightly more adverse consequences (minimally). A pressurizer pressure of 1070 psia or 1800 psia was used depending upon the event initial temperature. These pressures are conservative and produce the limiting consequences.

The difference between the Cycle 2 and the Cycle 12 HFP values are due to a slight increase in RCP heat load. The Cycle 12 HZP power level is 1% of the full power value. The initial HZP power level does not have a primary impact on the event consequences because the event is dominated by the SG blowdown and the negative reactivity available (scram and boron worth). The Cycle 12 SLB power levels are appropriate and conservative for the initial conditions.

The Cycle 12 analysis RCS flow is based upon the Technical Specification (TS) [Reference 2 TS 3.2.5] minimum flow requirement applicable for Mode 1. The Cycle 12 post-trip SLB analyses were performed to cover the operating Modes 1-4. Each of the SLB Mode 1-4 analyses were performed assuming a LOAC which causes a loss of power to the RCPs and a corresponding RCS flow coastdown. The RCS flow coastdown bounds the range of potential RCP configurations that could be present for the lower mode events. Thus, the analyses remain bounding.

The post-trip SLB break consequences are relatively insensitive to the initial pressurizer level because the SLB event rapidly empties the pressurizer and the RCS pressure is dominated by the behavior of drawing a void in the upper head. The Cycle 12 analysis used the pressurizer levels that correspond to the setpoints of the pressurizer level control system. These pressurizer levels are acceptable for initial conditions.

The number of SG tubes assumed to be plugged was increased to 1000 per SG for this analysis in Cycle 12. This produces more severe consequences because of a larger initial primary to secondary temperature difference. Thus, the higher tube plugging limit is conservative and bounding.

The Cycle 12 analysis does not credit boron injection (negative reactivity) via the charging system. This produces conservative and bounding consequences.

The Cycle 12 HFP analysis uses a small fuel gap conductance (HGAP) value because this maximizes the initial fuel temperature. This causes the greatest decrease in fuel temperature and resultant insertion of positive reactivity once the event is initiated and the reactor is tripped. The largest value of fuel gap conductance is 1.611 Btu/sec-ft^2-°F. The HZP SLB uses a larger value than the HFP scenarios and the subcritical uses the largest value. The large fuel rod gap conductance maximizes the fuel heat that can be removed across the gap and minimizes any lag that could occur during the cooldown. The HFP, HZP, and subcritical SLB analyses chose the HGAP values that produced conservative and bounding consequences. This differed between scenarios due to the initial conditions.

The Cycle 12 IBWs were used as a function of moderator temperature. The IBW differs for All Rods In (ARI) and ARI less the single highest worth CEA (N-1) configuration and are a function of moderator temperature. The IBWs are validated in the PAC as being bounded.

The Cycle 12 analysis iterated on CEA scram worth to determine the minimum worth required to meet the post-trip SLB acceptance criteria. The smaller scram worths are conservative and bounding. The CEA scram worths are validated in the PAC as being bounded.

The Doppler cooldown curves presented in Figure 1 are compared by examining the slope of each curve. The SLB transients are initiated from a quasi-equilibrium condition such that the reactivity addition (temperature cooldown) for each scenario would be based upon the slope of each curve at each temperature region. The Doppler cooldown curves are validated in the PAC as being bounded.

The Moderator cooldown curves presented in Figure 2 are compared by examining the slope of each curve. The SLB transients are initiated from a quasi-equilibrium condition such that the reactivity addition (temperature cooldown) for each scenario would be based upon the slope of each curve at each temperature region. The Moderator cooldown curves are validated in the PAC as being bounded.

During the SLB return to power, 3-D reactivity feedback credits are used to account for the local heatup of the inlet fluid in the hot channel which occurs near the location of the stuck CEA. This credit is based on the 3-D coupled neutronic-thermal-hydraulic calculations [Reference 9 and 13]. The Cycle 2 and Cycle 12 reactivity credits differ but are based upon the same method used in Cycle 2 [Reference 6] and continues to be appropriate for Cycle 12.

The core power distribution used in the HFP and HZP SLB scenarios are highly peaked due to both the core being in the N-1 configuration following the reactor trip and the cold RCS coolant resulting in highly bottom peaked core power distributions. The initially subcritical SLB scenarios maintained subcriticality throughout the event and without having significant increases in core power production, the consideration of the core power distribution was unnecessary. The core peaking factors are validated in the PAC as being bounded.

Table 2. Inverse Boron Worths

Tmod (°F)	IBW N-1 ppm/%Δρ	IBW ARI ppm/%Δρ
545	-103.13	-103.74
500	-97.71	-98.28
450	-93.15	-93.71
300	-84.1	-84.6
200	-80.32	-80.79
68	-77.5	-77.86

Figure 1. Doppler Cooldown Curve

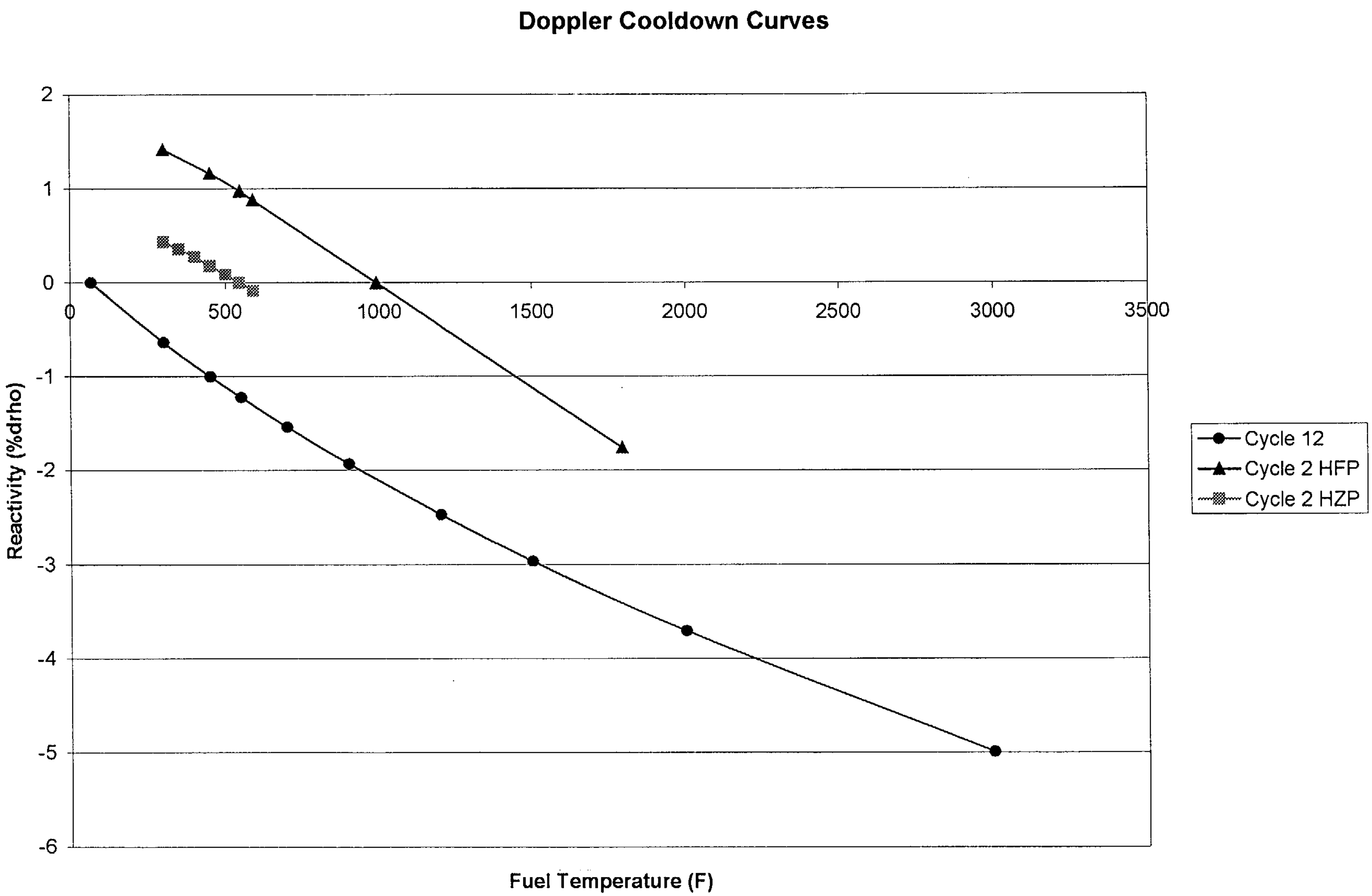
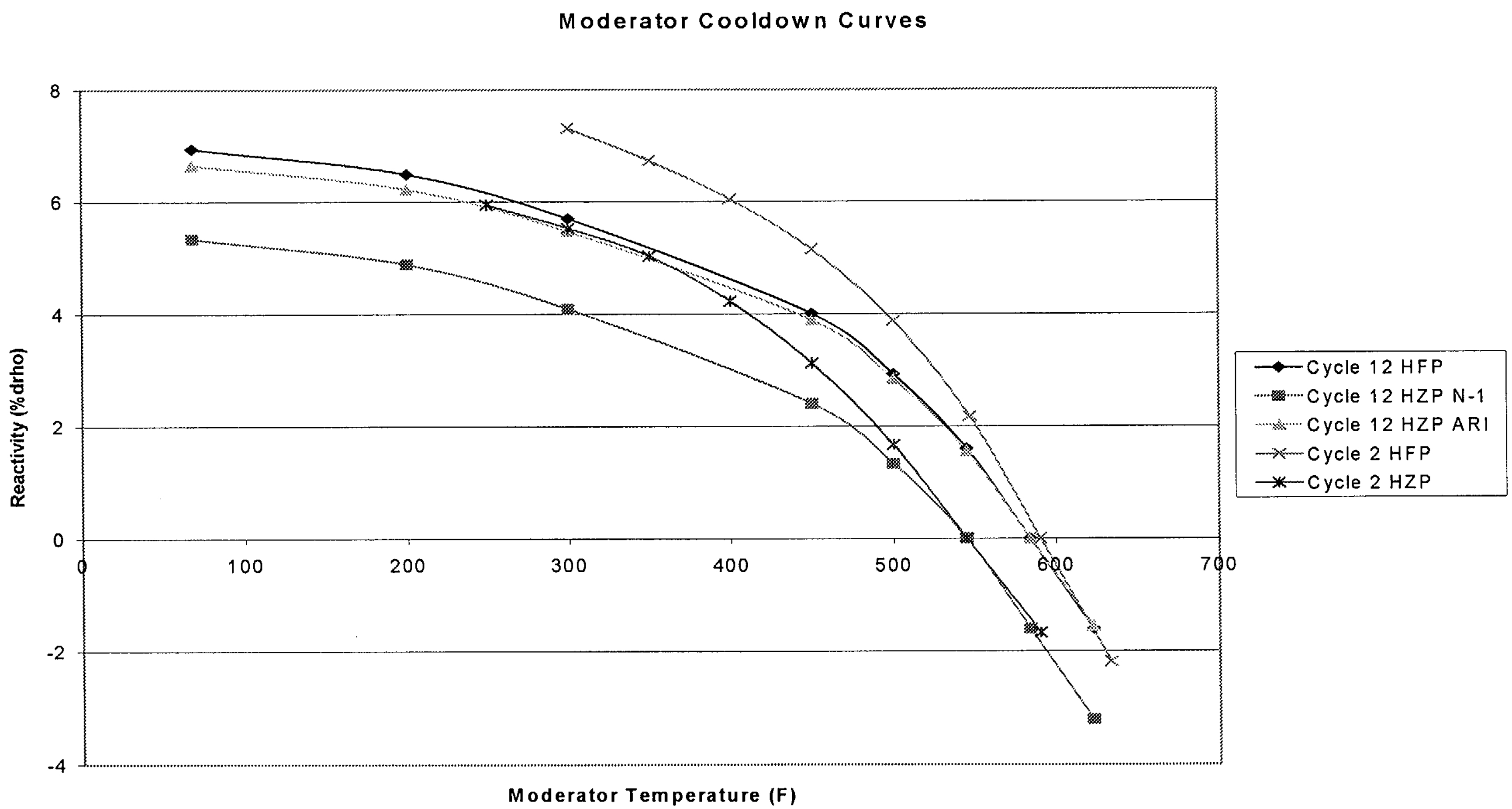


Figure 2. Moderator Cooldown Curve



3. Results

The SLB results remain similar to those presented in the Waterford 3 FSAR [Reference 1].

The intent of the post-trip SLB analysis was to determine the minimum scram worth required to not exceed the Specified Acceptable Fuel Design Limits (SAFDLs). Table 3 presents the post-trip SLB results and demonstrates the DNBR and PLHR SAFDLs are not exceeded.

Table 3. Post-Trip SLB Consequences

Event	Scram Worth (%Δρ)	Time (sec)	Fission Power (Mwt)	Decay Power (Mwt)	DNBR	LHR	Maximum Post-Trip Reactivity (%Δρ)
HFP	-6.65	72.8	191.1	148.25	5.36	20.42	-0.00327
HFP w/ LOAC	-6.60	142.3	175.62	130.45	1.30	16.0	-0.00056924
HZP w/ LOAC	-3.95	231.4	143.18	6.767	1.34	13.4	0.00012556

The intent of the Mode 3 and 4 SLB analyses were to demonstrate that the subcritical SLB scenarios remained subcritical by 0.01 %Δρ. This assures that no appreciable post-trip SLB return to power has occurred. Table 4 presents the results that demonstrate acceptable consequences.

Table 4. Mode 3 and 4 Post-Trip SLB Consequences

Event	Initial Temperature (°F)	Initial Subcriticality (%Δρ)	Maximum Post-Trip Reactivity (%Δρ)
SLB	400	-2.2*	-0.0949
SLB	450	-3.75*	-0.924
SLB	551	-5.3*	-0.825
SLB w/ LOAC	400	-2.2*	-0.0936
SLB w/ LOAC	450	-3.75*	-0.933
SLB w/ LOAC	551	-5.3*	-0.537

* The initial subcriticality is based upon TS 3.1.1.2 [Reference 2] which refers to the Core Operating Limits Report (COLR) [Reference 15]. The COLR Figure 1 provides the temperature dependent shutdown margin requirements. The plant procedures ensure the shutdown margin requirements are met by maintaining a conservative RCS boron concentration. The RCS boron concentration accounts for the worst stuck rod. Thus, the values used in Table 4 are the COLR Figure 1 values plus the worst stuck rod value because per TS 3.1.1.2 all CEAs are confirmed to be fully inserted therefore there is no stuck CEA. As a result of the PAC assessment, the COLR Figure 1 shutdown margin requirement will be increased for Cycle 12 to ensure that subcriticality is maintained for these events.

Since, the SLB event is a cooldown event that depressurizes the RCS, the acceptance criterion for the RCS pressure to remain below the safety limit was never approached.

In addition, since the SAFDLs were never exceeded a coolable core geometry was maintained and the radiological consequences remain bounded by previous analyses.

4. Conclusion

The Cycle 12 post-trip SLB event was performed using NRC approved methods and computer codes [Reference 3 and 5-13]. The analysis included the replacement of the Part Length Rods with full length CEAs and the reconfigured banks along with the removal of the 4 finger CEAs. The analysis results remain within the NRC approved Safety Evaluation Report [Reference 10 Subsection 5.1] acceptance criteria which corresponds to the Standard Review Plan [Reference 4 Section 15.1.5] acceptance criteria.

The post-trip SLB event acceptance criteria are:

1. Site boundary doses do not exceed 10CFR100 guidelines.
2. Coolable core geometry is maintained.
3. RCS pressure remains below the safety limit.

Thus, the Cycle 12 post-trip SLB event consequences are acceptable.

5. References

1. Waterford Steam Electric Station, Unit No. 3, Final Safety Analysis Report (FSAR), Facility Operating License Number NPF-38, Docket No. 50-382, through Revision 11.
2. Technical Specifications, Waterford Steam Electric Station, Unit No. 3, Docket No. 50.382, Appendix "A" to License No. 38, through Amendment 174.
3. NUREG-0787, Safety Evaluation Report related to the operation of Waterford Steam Electric Station, Unit No. 3, Docket No. 50-382, July 1981.
4. NUREG-0800, Standard Review Plan, Revision 2, July 1981.
5. CENPD-107, CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System, December 1981.
6. W3P86-3328, "Waterford 3 SES Docket No. 50-382 Reload Cycle 2 Reports," October 1, 1986.
7. R. V. MacBeth, "An Appraisal of Forced Convection Burnout Data," Proc. Inst. Mech. Engrs., Vol. 180, Pt. 3C, PP 37-50, 1965 - 1966.
8. D. H. Lee, "An Experimental Investigation of Forced Convection Burnout in High Pressure Water - Part IV, Large Diameter Tubes at About 1600 psia," A.E.E.W. Report R479, 1986.
9. CENPD-188-A, "HERMITE Space-Time Kinetics", July 1975.
10. NRC Safety Evaluation Report dated January 16, 1987, "Reload Analysis Report for Cycle 2 at Waterford 3."

11. CENPD-266-P-A, "The ROCS and DIT Computer Codes for Nuclear Design," April 1983.

12. CENPD-275-P-A, "C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorber," May 1988.

13. CENPD-161-P, "TORC Code: A Computer Code for Determining the Thermal Margin of a Reactor Core," July 1975.

14. W3F1-2002-004, "Waterford Steam Electric Station, Unit 3; Docket No. 50-382; Supplement to Amendment Request NPF-38-234; Replacement of Part-Length Control Element Assemblies," January 17, 2002.

15. Core Operating Limits Report for Cycle 11 Revision 0, September 22, 2000.

Attachment 2

To

W3F1-2002-0013

Control Element Assembly Ejection Event

Control Element Assembly Ejection Event

1. Event Description

The estimated frequency of a Control Element Assembly (CEA) Ejection event classifies it as a limiting fault. The CEA Ejection is postulated to result from a complete circumferential break of the control element drive mechanism (CEDM) housing or of the CEDM nozzle on the reactor vessel head, and is assumed to result in the CEA being ejected to a fully withdrawn position. The CEDM housing and the CEDM nozzle are an extension of the reactor coolant system boundary and designed and manufactured to Section III of the ASME Boiler and Pressure Vessel code. Hence, the occurrence of such a failure is considered highly unlikely.

A typical CEA ejection transient behaves in the following manner: After ejection of a CEA from the Hot Full Power (HFP) or the Hot Zero Power (HZP) (critical) initial condition, the core power rises rapidly for a brief period. The rise is terminated by the Doppler effect. Reactor shutdown is initiated by either the High Power Level trip or Core Protection Calculator - Variable Overpower trip, terminating the power transient. The core is protected against severe fuel damage by the CEA insertions permitted at various power levels by the Power Dependent Insertion Limit (PDIL) Technical Specification and by the High Power trip.

2. Analysis

The reload analyses included the replacement of the Part Length Rods (PLRs) with full length CEAs and the reconfigured banks along with the removal of the 4 finger CEAs (new CEA configuration). The Physics Assessment Checklist (PAC) failed to meet the acceptance criteria [Reference 5 Attachment 2] and required the CEA Ejection fuel failure scenarios [Reference 1 Section 15.4.3.2] to be reanalyzed for Cycle 12. As provided in Section 15.4.8.II of the Standard Review Plan (SRP) [Reference 3], the acceptance criteria for the CEA Ejection event are based on meeting the requirements of General Design Criterion (GDC) 28 as it relates to the effects of postulated reactivity accidents neither resulting in damage to the reactor coolant pressure boundary greater than limited local yielding, nor causing sufficient damage to impair significantly the capacity to cool the core. These criteria are consistent with those addressed in the Waterford 3 Safety Evaluation Report (Sections 15.2.4.6 and 15.4.5 of Reference 2). The specific criteria in the SRP for evaluating the control rod ejection accident are:

a. Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod.

b. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in the ASME Code.

c. The fission product inventory in the fuel rods calculated to experience a departure from nucleate boiling DNB condition is an input to the radiological evaluation. The radiological criteria used in the evaluation of control rod ejection accidents (PWRs) are given in Appendix B of Regulatory Guide 1.77 [Reference 6]

The evaluation of CEA Ejection is based on the methodology described in Reference 4 which was reviewed and approved by the NRC (Section 15.2.4.6 of Reference 2). Initial evaluations were performed as part of the PAC to address the criteria above, and demonstrated that the results of the fuel enthalpy analysis and Reactor Coolant System (RCS) overpressure evaluation presented in the FSAR (Tables 15.4-27 and 28 of Reference 1) remain valid for Cycle 12 operation. The PAC explicitly accounts for partial length rod (PLR) replacement and the associated new CEA configurations. The impact of PLR replacements and the CEA configurations is considered in the CEA Ejection analysis through the core physics parameters employed in the analysis.

The PAC also demonstrated that the results of the DNB evaluation presented in the Final Safety Analysis Report (FSAR) [Reference 1] for zero power initial conditions (Table 15.4-29 of Reference 1) remain valid for Cycle 12 operation. Additionally, as described below, the PAC was supplemented with a Cycle 12-specific evaluation for full power initial conditions to demonstrate that the fraction of rods that experience DNB is less than the 9.12% reported in the FSAR (Table 15.4-29 of Reference 1).

The fuel failure rate is determined by calculating the Departure from Nucleate Boiling Ratio (DNBR) value corresponding to selected pre- and post-ejected radial peaking factors and ejected rod worths, and applying the data to a pin census to determine the specific number of failed fuel pins for the given conditions. Once the DNBR values have been calculated based on the selected pre- and post-ejected radial peaking factors, axial power distributions, and ejected rod worths, a deterministic fuel failure model is applied to the data. The fuel failure model assumes that every fuel pin in the pin census with a DNBR value calculated below the SAFDL results in cladding failure.

The fuel failure rate is calculated based on initial conditions and physics parameters ensuring that the range of plant full power operation is bounded. The analysis utilized physics parameters such as Doppler feedback, axial power shapes, ejected rod worths and excore detector decalibration factors that bound Cycle 12 operation. Table 1 compares values assumed in the Cycle 12 full power analysis with those used in the limiting FSAR (Cycle 1 full power beginning of cycle case) analysis.

Table 1: Cycle 1 to Cycle 12 Comparison

PARAMETER	CYCLE 1	CYCLE 12
Delayed Neutron Fraction	0.007234	0.005126
Ejected Worth, %$\Delta\rho$	0.1639	0.1180
Moderator Temperature Coefficient, 10^{-4} %$\Delta\rho$/°F	+0.5	-0.2
CEA Worth Inserted on Reactor Trip, %$\Delta\rho$	-6.4	-2.0
Variable Overpower Trip, % of Full Power	130	126
Initial Three-Dimensional Fuel Pin Peaking Factor	3.50	1.98
Core Flow, 10^6 lbm/hr	143	149
Core Inlet Temperature, °F	534	545

The delayed neutron fraction was conservatively chosen to bound the minimum value calculated for Cycle 12. Similarly, the ejected worth conservatively bounds maximum values calculated for Cycle 12. These values are validated in the PAC as being bounded

The Cycle 12 moderator temperature coefficient is the least negative value that will be allowed by the Waterford 3 Cycle 12 COLR (Section 3.1.1.3).

The Cycle 12 scram worth was chosen to bound both full and zero power operating conditions. The Cycle 12 Variable Overpower reactor trip setpoint conservatively accounts for power measurement uncertainty, and for excore detector decalibration associated with the CEA Ejection event.

The initial three-dimensional fuel pin peaking factor for Cycle 12 is established based on a conservative axial power shape combined with a radial peaking factor that conservatively bounds the maximum value calculated for the Cycle.

The core flow and inlet temperature were chosen consistent with the initial three-dimensional fuel pin peaking factor to establish the minimum initial DNBR margin that will be allowed by the Waterford 3 Cycle 12 COLR (Section 3.2.4). Starting from a Power Operating Limit ensures that the event consequences remain bounding.

3. Results

The Cycle 12 CEA Ejection analysis demonstrated that the maximum fuel failure rate remains below approximately 8.4%, which is less than the current FSAR results of 9.12% fuel failure that was reviewed and approved in the Waterford 3 SER (Reference 2). In addition, the PAC demonstrated that the results of the fuel enthalpy and RCS overpressure analyses presented in the current FSAR remain valid for Cycle 12 operation. Thus, the Cycle 12 CEA Ejection analysis remains bounded by the Cycle 1 analysis.

4. Conclusion

The Cycle 12 CEA Ejection analysis was performed using NRC approved methods and computer codes [Reference 1, 2 and 4]. The analysis included the replacement of the Part Length Rods with full length CEAs and the reconfigured banks along with the removal of the 4 finger CEAs. The analysis results remained within the NRC approved Safety Evaluation Report [Reference 2 Subsection 15.2.4.6 and 15.4.5] acceptance criteria which corresponds to the Standard Review Plan [Reference 3 Section 15.4.8.II] acceptance criteria.

The specific acceptance criteria for evaluating the control rod ejection accident are:

a. Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod.

b. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in the ASME Code.

c. The fission product inventory in the fuel rods calculated to experience a DNB condition is an input to the radiological evaluation. The radiological criteria used in the evaluation of control rod ejection accidents (PWRs) are given in Appendix B of Regulatory Guide 1.77.

The evaluation performed for Waterford 3 utilizing Cycle 12 specific data produce a maximum number of fuel pins in DNB that is bounded by the value presented in the Waterford 3 FSAR. Therefore, the DNB calculations and the radiological dose release for the CEA Ejection event presented in the Waterford 3 FSAR remain valid for Cycle 12 operation.

5. References

1. Waterford Steam Electric Station, Unit No. 3, Final Safety Analysis Report (FSAR), Facility Operating License Number NPF-38, Docket No. 50-382, through Revision 11.
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