ENCLOSURE II TO ET 19-0008

Westinghouse Responses to NRC Request for Additional Information Documented in ADAMS Accession No. ML18270A094 on the Core Design and Safety Analyses Methodology Transition Program [Non-Proprietary]

(7 PAGES)

Westinghouse Responses to NRC Request for Additional Information Documented in ADAMS Accession No. ML18270A094 on the Core Design and Safety Analyses Methodology Transition Program [Non-Proprietary]

(7 pages including cover page)

Request

Rod Cluster Control Assembly Ejection Accident

Please demonstrate that the CREA analysis results discussed in Section 2.5.6 of Enclosure 1 are consistent with General Design Criterion (GDC) 28, "Reactivity limits," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the Code of Federal Regulations (10 CFR) Part 50 for long term cooling, and with the assumptions employed with the radiological analyses supporting the implementation of the alternative source term in accordance with 10 CFR 50.67, "Accident source term." GDC 28 requires that the effects of postulated reactivity accidents neither damage the reactor coolant pressure boundary greater than limited local yielding, nor impair the ability to cool the core. Meanwhile, the radiological consequence analysis of the CREA assumes no more than 10-percent of fuel centerline melt at the hot spot.

Since the NRC position stated in Regulatory Guide 1.77, "Assumptions used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974 (ADAMS Accession No. ML003740279), indicating that acceptance criteria of 280 calories per gram (cal/gm) should be applied to the event, and Westinghouse's position that 200 cal/gm and 10-percent fuel centerline melt criteria contained in WCAP-7588, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," were established, the NRC staff determined that more restrictive acceptance criteria were needed to assure the core would remain in a coolable geometry, given the potential for high temperature cladding failure and pellet cladding interaction. This position is documented in an NRC staff internal memo dated April 3, 2015, titled, "Results of Periodic Review of Regulatory Guide 1.77" (ADAMS Accession No. ML15075A311). The position is supported by the NRC staff internal memo dated January 19, 2007, titled, "Technical and Regulatory Basis for the Reactivity Initiated Accident Interim Acceptance Criteria and Guidance" (ADAMS Accession No. ML070220400).

a) *Please justify why 200 cal/gm and 10-percent fuel melt are appropriate acceptance criteria given that much more is known now about fuel damage behavior than when the method was approved in 1975.*

b) *Please discuss how the calculations appropriately include the effects of nuclear fuel TCD in the evaluation against the acceptance criteria. The hot spot fuel melt limit of 10-percent relates to the prevention of fuel dispersal into the coolant, and is reflected in the assumptions employed in the radiological analyses.*

Response

As subsequently demonstrated by the responses to part a) and part b) of this request, the Rod Cluster Control Assembly (RCCA) ejection accident analysis results discussed in Section 2.5.6 of WCAP-17658-NP Revision 1 are consistent with General Design Criterion (GDC) 28, "Reactivity limits," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the Code of Federal Regulations (10 CFR) Part 50 for long term cooling, and with the assumptions employed with the radiological analyses supporting the implementation of the alternative source term.

a) *Please justify why 200 cal/gm and 10-percent fuel melt are appropriate acceptance criteria given that much more is known now about fuel damage behavior than when the method was approved in 1975.*

The WCAP-7588 Revision 1-A acceptance criteria are a conservative replacement for the Regulatory Guide 1.77 criteria, as subsequently discussed. Specifically, the 200 cal/gm criterion is the conservative coolability limit utilized by Westinghouse relative to the Regulatory Guide 1.77 limit of 280 cal/gm. Additionally, the 10% fuel melt criterion is a conservative limit that has been used by Westinghouse for input to the radiological dose calculations. The 10% fuel melt limit is not intended to be used as the coolability limit.

As the NRC has stated, more restrictive acceptance criteria than the Regulatory Guide 1.77 criteria may be needed. However, the existing 1-D analysis methodology is an overly-conservative method relative to more advanced 2-D or 3-D methods. The analysis documented in Section 2.5.6 of WCAP-17658-NP Revision 1 utilized the more restrictive WCAP-7588 Revision 1-A acceptance criteria (200 cal/gm) along with the historical 1-D analysis method in order to maintain the overall conservatism of the analysis.

It is noted that utilizing this approach (i.e.. the 200 cal/gm limit in combination with the conservative 1-D analysis method) was discussed between Wolf Creek. Westinghouse. and the NRC during a clarification call associated with this request in order to ensure that there was understanding on how the evaluation of the RCCA ejection accident would be performed.

b) Please discuss how the calculations appropriately include the effects of nuclear fuel TCD in the evaluation against the acceptance criteria. The hot spot fuel melt limit of 10-percent relates to the prevention of fuel dispersal into the coolant, and is reflected in the assumptions employed in the radiological analyses.

Regarding the issue of fuel thermal conductivity degradation (TCD), WCAP-17658-NP Revision 1 referenced LTR-NRC-12-18 (ADAMS Accession No. ML12053A105) as the basis for the continued safe operation of plants analyzed with Westinghouse codes and methods. consistent with the basis used for other plants analyzed with Westinghouse codes and methods. The rationale for referencing LTR-NRC-12-18 was due to the fact that the Westinghouse Performance Analysis and Design Model (PAD5) submittal. WCAP-17642-P-A Revision 1. was still under review and had not yet been approved by the NRC at the time that the licensing report for the Methodology Transition, WCAP-17658-NP Revision 1, was submitted.

Following the submittal of the Methodology Transition license amendment request, WCAP-17642-P-A Revision 1 was approved for use and provided the technical basis for utilizing the PAD5 model to address fuel TCD. However, as documented within LTR-NRC-18-7 (ADAMS Accession No. ML18023B555), the schedule to perform all not-LOCA analyses required to transition to the PAD5 models and methods is [ ][a,c] Furthermore, [ ][a,c] Therefore, due to the timing of the approval of WCAP-17642-P-A Revision 1 relative to the timing of the Methodology Transition submittal, it was not feasible to fully incorporate the PAD5 models and methods into the not-LOCA analyses within the confines of the Methodology Transition submittal.

Nevertheless, in order to respond to this request and to demonstrate that the acceptance criteria (i.e., 200 cal/gm and 10% fuel melt) for the RCCA ejection accident will continue to be met once the effects of TCD are explicitly included within the licensing basis analysis, a demonstration analysis was performed. This demonstration analysis was based on the RCCA ejection analysis described in Section 2.5.6 of WCAP-17658-NP Revision 1, but with analysis inputs revised to account for the effects of TCD. The revised analysis inputs were based on the models and methods described in WCAP-17642-P-A Revision 1. Specifically, the input changes included [ ][a,c] for both the HFP and hot zero power (HZP) cases. The conservative ejected rod worth values used in the analysis described in WCAP-17658-NP Revision 1 continued to be used.

The burnup-dependent inputs for each of the cases in the demonstration analysis were based on [ ][a,c] In addition, as the fuel melting temperature is burnup-dependent, it was decreased from 4900°F at beginning of cycle (BOC) and 4800°F at end of cycle (EOC) to [ ][a,c] for all cases. [1]

[1] The Wolf Creek Generating Station (WCGS) Technical Specification Safety Limit 2.1.1.2 is 5080°F decreasing at 58°F per 10,000 MWD/MTU of burnup.

As expected, with the effects of TCD included, the peak fuel average and centerline temperatures increased. For the HZP cases. [ ]^{a,c} The results from the demonstration analysis are presented in Table 1.

In summary, the demonstration analysis of the RCCA ejection accident utilized the same inputs as those used for the analysis described in Section 2.5.6 of WCAP-17658-NP Revision 1, with the following exceptions:

[ ]^{a,c}

Table 1: Wolf Creek RCCA Ejection Results

[ ]^{a,c}

Based upon the results in Table 1, all acceptance criteria continue to be met. Therefore, it is concluded that the effects of TCD can be accommodated for the RCCA ejection accident for Wolf Creek.

As previously stated, [ ]^{a,c} Thus, the intent is to not incorporate the PAD5 models and methods into the licensing basis for the RCCA ejection accident at this time, as the remaining not-LOCA analyses have not yet been updated (with the exception of the Steamline Break (SLB) analysis, which is discussed in the subsequent response).

It is recognized that the demonstration analysis performed for this evaluation [ ]^{a,c} Therefore, it has already been demonstrated that the analysis bounds the upcoming fuel cycle. Furthermore, to ensure that the results continue to remain valid for future cycles, an additional check will be performed as part of the reload process to confirm that the [ ]^{a,c} remain bounding on a cycle-specific basis until Wolf Creek fully implements the PAD5 models and methods into its licensing basis for the not-LOCA analyses. Once the PAD5 models and methods are fully incorporated into the Wolf Creek licensing basis, the PAD5 analysis will become the analysis of record and the [ ]^{a,c} will become the limits checked as part of the standard reload process.

Request

Steamline Break Accident

Please demonstrate how the MSLB analyses contained in Section 2.2.5 of Enclosure 1 are consistent with the requirements of 10 CFR Part 50, Appendix A, GDC 27, "Combined reactivity control systems capability," and GDC 28. GDC 27 requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes so that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained. GDC 28 requires that the effects of postulated reactivity accidents neither damage the reactor coolant pressure boundary greater than limited local yielding, nor impair the ability to cool the core. Per the Westinghouse analytic methods proposed for implementation, the MSLB is analyzed to demonstrate that fuel damage criteria are satisfied, including departure from nucleate boiling limits and fuel melt limits.

In order for the NRC staff to determine whether the MSLB analyses assure compliance with the requirements of GDC 27 and 28, please discuss how the MSLB results and acceptance criteria appropriately include the effects of TCD, including:

a) The hot zero power and hot full power departure from nucleate boiling ratio, and

b) The hot full power peak linear heat rate.

Response

As subsequently demonstrated by the responses to part a) and part b) of this request, the SLB analyses contained in Section 2.2.5 of WCAP-17658-NP Revision 1 are consistent with the requirements of 10 CFR Part 50, Appendix A, GDC 27, "Combined reactivity control systems capability," and GDC 28.

a) The hot zero power and hot full power departure from nucleate boiling ratio, and

The response to part a) is combined with the response to part b).

b) The hot full power peak linear heat rate.

Per [

]^{a,c} there are no changes to the departure from nucleate boiling ratio (DNBR) results for the HZP SLB and HFP SLB accidents due to the effects of TCD.

While the transient analysis of the system response for the HFP SLB accident is not impacted by the effects of TCD, the power-to-melt limit (i.e., peak fuel linear heat generation rate that would cause fuel melting, expressed in terms of [ ]^{a,c}) is affected by TCD. Therefore, the power-to-melt limit used in the analysis of the HFP SLB transient was recalculated to explicitly consider the effects of TCD and defined as a [

]^{a,c} The transient analysis statepoints were then used to recalculate the corresponding peak linear heat generation rate observed during the HFP SLB event, based on upon the upcoming fuel cycle (Cycle 24), for comparison to the revised power-to-melt limit. The results for the limiting HFP SLB case are summarized in Table 2. The peak fuel linear heat generation rate does not exceed the value that would cause fuel melting.

For the HFP SLB, the transient analysis of the system response is consistent with that described in Section 2.2.5.2 of WCAP-17658-NP Revision 1. The calculation of the peak fuel linear heat generation rate is based upon the upcoming fuel cycle (Cycle 24). As the peak fuel linear heat generation rate can change on a cycle-specific basis, an additional check will be performed as part of the reload process to confirm that the [ ]^{a,c} power-to-melt limit with the effects of TCD considered continues to be met for the HFP SLB accident until Wolf Creek fully implements the PAD5 models and methods into its licensing basis for the not-LOCA analyses. Once the PAD5 models and methods are fully incorporated into the Wolf Creek licensing basis, the PAD5 analysis will become the analysis of record and the [ ]^{a,c} power-to-melt limit will be checked as part of the standard reload process.

In addition to the HFP SLB accident, the SLB at power with coincident rod withdrawal (SLB w/RWAP) case was explicitly evaluated. While this case is not presented within the Wolf Creek Updated Safety Analysis Report (USAR), it is analyzed to address the concerns outlined in NRC Information Notice IE-79-22 related to non-safety grade equipment being subjected to an adverse environment from high-energy line breaks inside or outside containment.

The scenario is that a high-energy steamline break could fail the automatic rod control system cabling and/or equipment such that the RCCA banks begin to withdraw from the core simultaneous with the break. Such a rod withdrawal, combined with the core power increase caused by the steamline break, may lead to a rapid power excursion and a potentially adverse core condition. RCCA bank withdrawal during the steamline break could only occur with the rod control system in the automatic mode. At zero power, the rod control system is in the manual mode, and therefore could not inadvertently withdraw rods due to equipment or cabling being exposed to an adverse condition. As a result, the coincident RCCA withdrawal can only be postulated for a steamline break from an at-power initial condition.

[

]^{a,c}

The input parameters, assumptions, and acceptance criteria for the transient analysis of the SLB w/RWAP accident are the same as those for the HFP SLB accident (Section 2.2.5.2.1.2 of WCAP-17658-NP Revision 1), with the following additions or changes:

- [

  ]^{a,c}
- Cases were analyzed for both minimum (0 percent) and maximum (10 percent) steam generator tube plugging (SGTP) conditions.
- Cases were analyzed assuming both minimum and maximum reactivity feedback coefficients, corresponding to BOC and EOC core conditions. [

  ]^{a,c}
- Rod withdrawal was conservatively modeled to begin at the time of the break, with no credit taken for any delay. A constant [ ]^{a,c} maximum differential rod worth was assumed for control bank D at hot full power (D-bank is the only one that is permitted to be inserted into the core at full power). The rods withdraw until reactor trip is actuated, or until the rods would be fully out of the core.

Based upon the transient analysis results, the following limiting cases were selected in order to demonstrate that the DNBR and peak fuel linear heat generation rate criteria continue to be met:

A. 0% SGTP, maximum reactivity feedback. [ ]^{a,c}

B. 10% SGTP, minimum reactivity feedback. [ ]^{a,c}

C. 10% SGTP, minimum reactivity feedback. [ ]^{a,c}

[ ]^{a,c} The results for the limiting cases are summarized in Table 3.

The results of the analysis of the SLB w/RWAP accident demonstrate that the DNB design basis continues to be met with the effects of TCD explicitly included in the analysis. Additionally, the peak fuel linear heat generation rate does not exceed the value that would cause fuel centerline melting.

The analysis of the SLB w/RWAP accident assumed a maximum differential rod worth of [ ]^{a,c} to bound the value for the upcoming fuel cycle (Cycle 24). Also, the calculation of the peak fuel linear heat generation rate is based upon the upcoming fuel cycle (Cycle 24). As the maximum differential rod worth and peak fuel linear heat generation rate can change on a cycle-specific basis, additional checks will be performed as part of the reload process to confirm that the maximum differential rod worth limit remains bounding and the [ ]^{a,c} power-to-melt limit with the effects of TCD considered continues to be met for the SLB w/RWAP accident until Wolf Creek fully implements the PAD5 models and methods into its licensing basis for the not-LOCA analyses. Once the PAD5 models and methods are fully incorporated into the Wolf Creek licensing basis, the PAD5 analysis will become the analysis of record and these limits will be checked as part of the standard reload process.

Table 2: Wolf Creek HFP SLB Results

[ ] a,c

Table 3: Wolf Creek SLB w/RWAP Results

[ ] a,c