Westinghouse Class 3

Westinghouse Assessment of Topical Report Validity for Reactivity Insertion Accidents with High Burnup Fuel

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A Background

In a letter to Westinghouse dated November 14, 1994 (Reference 1), the NRC requested that the fuel vendors review their previously approved topical reports to assess if these topical reports remain appropriate in light of the unexpectedly low failure threshold seen in the CABRI Reactivity Insertion Accidents (RIA) test results. This transmittal responds to the NRC request.

In Reference 2, the Industry Issues Task Force (ITF) provided to the NRC information detailing the safety significance assessment with respect to the potential reduction in failure threshold for high burnup fuel during postulated RIA. This report concluded that the only RIA of concern for PWRs was the Rod Ejection event, and that the probability of this event occurring was extremely small (10^{-4} to 10^{-6} per year). This report further concluded that even if a rod ejection were to occur, the radiological consequences of this event would be well within the NRC requirements for this event, even if it was conservatively assumed that high burnup fuel in the core would fail at extremely low levels of energy deposition.

B Introduction and Summary

The Westinghouse response to the NRC's request to assess the validity of previously approved topicals consists of two parts: the first part details analysis performed by Westinghouse; the second part discusses specific topical report assessments. It should be noted that the conclusions of the first part demonstrate that there are significant conservatisms inherent in the analysis of both the rod ejection event and its radiological consequences for Westinghouse reactors within the licensing basis of currently approved NRC methodology. These conservatisms in combination are more than sufficient to demonstrate the safety of high burnup fuel and current plant operations; and to show that fuel burnup limits are not compromised even if the new RIA test data are assumed to be valid and applicable. The second part concludes that information presented in currently approved Westinghouse topicals continues to remain applicable, even when the mechanisms postulated to result in the high burnup RIA failures are considered.

Part 1: Details of Safety Assessment

1.0 Purpose of Assessment

For PWRs, the only Final Safety Analysis Report (FSAR) transient that will result in any fuel approaching or exceeding the levels of energy deposition that are of concern for this issue is the rod ejection event as noted in Reference 2.

The purpose of this Safety Assessment is to demonstrate that there are significant conservatisms inherent in the analysis of the rod ejection event and its radiological consequences. Taken together, these conservatisms are more than sufficient to demonstrate the safety of Westinghouse reactors and high burnup fuel, even if it is assumed that the new RIA test data are valid and applicable. Furthermore, safety can be demonstrated within the licensing basis of current NRC-approved methodology.

2.0 Safety Assessment

2.1 Description of the Event

The RCCA Ejection accident is a postulated Condition IV event which is assumed to occur as a result of a passive, mechanical failure of the control rod drive mechanism pressure housing. The failure of the pressure housing would cause the full reactor coolant pressure to act across the drive system mechanism, which could cause the control rod and drive shaft to be ejected from the core. If the control rod is initially deeply inserted into the core, and the core is at a just-critical condition, the ejection would cause a rapid reactivity insertion which, together with the adverse core power distribution associated with the ejected rod, could cause localized fuel rod damage and the release of radioactive fission products into the reactor coolant system or containment. The transient is terminated by the Doppler reactivity feedback due to the increased fuel temperature and by a reactor trip actuated by the neutron flux protection signals. The accident is analyzed to show that if the event should occur, the transient is terminated before conditions are reached which could result in a significant impairment in the ability to cool the core, and that the radiological consequences are well within the requirements of 10 CFR Part 100.

2.2 Event Probability

Since this event can only be caused by a passive, mechanical failure, the RCCA Ejection accident is considered to be a very low-probability event, with an event frequency in the range of 10^{-4} to 10^{-6} per year (ANSI 51.1-1983). This frequency is supported by reactor operating history, in which not one event has taken place in over 2,400 reactor years of commercial PWR world-wide operations, including over 1,000 reactor years of domestic operations. In addition to the very low frequency of the event, in order to produce the most severe reactivity insertion effects as presented in a typical plant FSAR, the following conditions would also have to exist simultaneously:

- the reactor would have to be just-critical,
- the control rods would have to be inserted to the insertion limit,
- the ejected rod would have to be one of the limiting inserted rods,
- an adverse Xenon distribution would have to be present to maximize the ejected rod worth.

Thus, it can be seen that the probability of a rod ejection event which may result in a substantial reactivity insertion is even lower, in that the probability of a rod ejection from a low or hot-zero-power condition with all rods in is even more unlikely due to the reduced amount of time that a plant resides in this mode of operation. Indeed, for plants that normally operate in an "all rods out" configuration, a rod ejection event would most likely not result in an appreciable reactivity insertion.

2.3 Additional Conservatisms:

Furthermore, in addition to the low probability of the event and its precursor conditions as described above, significant conservatism [illegible]ically applied to the following factors which affect the results of the transient analysis of the event.

- the ejected rod worth,
- the ejected rod peaking factor,
- the initial peaking factor,
- the delayed neutron fraction,
- the Doppler and moderator feedback,
- the initial hot spot fuel temperature,
- post-DNB heat transfer,
- reactor trip point and time delay,
- trip reactivity insertion.

For current Westinghouse rod ejection analyses, the transient occurs on the order of magnitude of 0.1 sec for the zero initial power case; whereas, the recent test data that documented fuel failure was on the order of 0.01 sec.

In addition to the above, there are significant conservatisms in the Westinghouse licensing-basis analysis approach, in which the reactor core kinetics analysis is performed using a one-dimensional (axial) core kinetics model, and the hot-spot fuel heat transfer calculation is performed using a separate code, assuming a conservative, constant, post-ejection power peaking factor. Present core transient analysis scoping calculations using the Westinghouse SPNOVA[3] code have demonstrated that the current

Westinghouse licensing basis methodology overpredicts the calculated peak fuel enthalpy by a factor of 2-4.

2.4 Fuel Dispersal Considerations

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy, have been carried out as part of the SPERT project by the Idaho Nuclear Corporation[4]. Extensive tests of UO_2 zirconium clad fuel rods representative of those in Pressurized Water Reactor type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm. These results differ significantly from the TREAT[5] results, which indicated a failure threshold of 280 cal/gm. The clad failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure, (large fuel dispersal, large pressure rise) even for irradiated rods, did not occur below 300 cal/gm.[6] Based on these results, Westinghouse has internally set the threshold limit at 200 cal/gm*. Therefore, for the ejected rod event, the analysis value must remain below 200 cal/gm*. This also supports the NRCs conclusion, as specified in Reference 12, that "for the high burnup fuel, the potential for a damaging pressure pulse is small in comparison to the dispersion of lower burnup fuel at high enthalpy". This is even more true for failures at low levels of energy deposition, since there is very little energy present in the dispersed particles to create a pressure pulse.

In view of the above experimental results, Westinghouse uses specific criteria[6] to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are:

- average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel,*
- average clad temperature at the hot spot below the temperature at which clad embrittlement may be expected (2,700 °F)**,

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* As specified in Reference 6, the limit for average fuel pellet enthalpy at the hot spot is set at 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel. For conservative bounding calculations, 200 cal/gm is used for both fuel types (i.e., irradiated and unirradiated).

** This is not required by the NRC and has been removed as a criteria for some plants.

- peak reactor coolant pressure less than that which would cause stresses to exceed the faulted condition stress limits, and

- fuel melting will be limited to less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion (1) above.

Demonstration that the above criteria are satisfied will continue to ensure that there are no long term coolability issues, even if it is assumed that the new RIA test data are valid and applicable.

Various reactivity impacts of fragmented fuel were discussed in Reference 7, which concluded that fuel fragmentation could lead to a positive reactivity component in LWR's. A more detailed evaluation of the Reference 7 data indicates that fuel fragmentation and dispersal would have a large net negative reactivity component in PWR's thus reducing reactivity insertion consequences.

Factors influencing reactivity following fragmentation of fuel include the effect of fuel pellet or fragment size, fuel temperature, and potential void formation in the moderator. Void formation induces a large negative reactivity component by reducing moderation. The smaller size of the fragment compared to the original pellet also decreases reactivity since resonance self-shielding is reduced. The reduced temperature of the fragment compared to the original pellet decreases doppler feedback, however, and can increase reactivity if long term constant power conditions are assumed.

The evaluation in Reference 7 made several unrealistic assumptions in this regard and concluded that under certain conditions the net impact of these mechanisms could be a net positive reactivity associated with fuel fragmentation. The evaluation assumed that the heat generation rate in the initially intact fuel rod was sufficient to lead to fuel melting, but took no credit for formation of additional voids in the moderator when the rod was assumed to fragment. The calculations performed to determine the temperature reactivity effect for the original pellet and the fragments also assumed that both were in thermal equilibrium with constant heat generation rates. Westinghouse calculations based on rod ejection transient heat generation rates on intact fuel rods and postulated pellet fragments confirm that the large temperature differential assumed in Reference 7 does not occur until after control rods have entered the core.

During a rod ejection transient, the peak fuel temperature occurs within about 2 seconds following the rod ejection as shown in Figure 4.3, Reference 6. By this time, control rods are dropping into the core, adding large amounts of negative reactivity. If fragmentation is assumed to occur prior to the insertion of control rods, additional void formation in the moderator and smaller size of the fragments will very quickly add negative reactivity. Within this short time span, Westinghouse analysis confirms that the positive reactivity associated with the temperature differential between intact and fragmented fuel is insufficient to overcome the negative reactivity effect of reduced size fragments. This does not credit the substantially larger, negative reactivity associated with moderator voiding. Any positive reactivity resulting from further cooling of the fragments after this short time would be offset by the negative reactivity from control rod insertion. In the longer term, control rods will maintain the core in a subcritical state.

2.5 Evaluation of the Effect on High Burnup Fuel/Methodology

This safety assessment was performed by determining the peak fuel enthalpy that would be experienced by high burnup fuel as a result of a control rod ejection transient. The analysis was performed by first determining the number of fuel assemblies which experience high post-ejection power peaking factors and high burnups, and combining this with the calculated peak fuel enthalpy as a function of power peaking factor.

The ejected rod analysis is performed using a combination of three-dimensional static nuclear methods[8] and one-dimensional nuclear kinetics methods[9] which employ very conservative methodology. These Westinghouse design codes[8][9] have been licensed by the NRC. The preconditions for the transient kinetics calculations are based on maximum ejected rod worths and hot channel factors as calculated by the ANC[8] code. Other key parameters used in the analysis include: reactivity feedback weighing factors, moderator and doppler coefficient, delayed neutron fraction, and trip reactivity insertion.

The calculation of the rod ejection transient is performed in two stages: first, an average channel core calculation and then, a hot region calculation. The average core calculation is performed using the TWINKLE[9] code to determine the average power generation with time including the various total core feedback effects (i.e., Doppler reactivity and moderator density reactivity). Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core power generation by the hot channel factor, as obtained from ANC[8], and performing a fuel rod transient heat-transfer calculation using the FACTRAN[10] code.

2.5.1 Analysis

The goals of the neutronic analysis performed by Westinghouse is to first determine if there is a correlation between post rod ejection heat flux hot channel factors, F_Q, and enthalpy rise and the corresponding assembly burnups. Second, the assembly population distribution as a function of post rod ejection F_Q is determined for several severe rod ejection simulations.

There exist numerous factors which may influence the core power distribution and power peaking during a rod ejection event. These may include the following effects:

- Number of loops
- Type of loading pattern
- Power level
- Fuel type
- Presence of axial blanket fuel
- Burnable absorber type
- Initial radial power distribution
- Control rod pattern
- Control rod insertion limits
- Combinations of the above

Three-dimensional static neutronic studies are performed to demonstrate that the rod ejection event results in significant peaking factor increases for only a very localized area of the core. Hence, only a small population of high burnup fuel experiences high peaking factors and enthalpies.

The rod ejection accident is typically evaluated as part of the reload design at four reactor conditions, hot-zero-power (HZP) and hot-full-power (HFP), at BOL and EOL. The calculations are performed using the PHOENIX-P/ANC design system(8)(11) with either a simple two-dimensional calculation synthesized with an assumed axial peaking factor or a full three-dimensional calculation.

A scoping analysis was first performed to determine the effect of the control rod ejection accident on higher burnup fuel. The results of this scoping analysis were used as input to the Safety Assessment provided to the NRC in Reference 2. The transient was analyzed by calculating the peak values of energy deposition (cal/gm) that could be reached during the transient in various fuel assemblies in the core. These calculations were performed using the current Westinghouse licensing basis rod ejection methodology(6). The analysis was performed for the end-of-cycle hot-zero-power case, since this case results in the most severe nuclear power transient as well as the highest fuel burnups. For a specific ejected rod worth, the core average nuclear power transient was calculated once, but the transient fuel temperature calculation was performed several times with different transient peaking factors to represent the various fuel assemblies.

Initially, to evaluate post rod ejection peaking factor behavior for Westinghouse plants, a series of static rod ejection calculations were performed for four different plant reload fuel designs. These were 4-loop plants with relatively long cycle lengths ([ ] a, c) and relatively high maximum fuel burnups.

In order to confirm the results of the scoping analysis, a more detailed study was also performed. This involved twenty different high discharge burnup reload and hypothetical designs (lead rod burnup from [ ][a, c]). Two, three and four-loop designs were analyzed, utilizing various Westinghouse control rod patterns. Low leakage patterns, some including axial blankets were evaluated, and in one instance a high leakage core was investigated. Additionally, a ten foot core and 14 foot core were also analyzed.

Detailed full-core, three-dimensional PHOENIX-P/ANC static modeling of rod ejections was performed for the selected cycle of each plant. The HZP ANC models were conservatively preconditioned with control rods at typical zero power insertion limits. All calculations were performed using the standard reload procedure, with established uncertainties applied to all analysis results.

Following generation of the rod ejection cases, the peak quarter assembly F_Q was converted to a cal/gm value based on one-dimensional kinetics analysis. Population studies for peak quarter assembly cal/gm versus maximum quarter assembly pin power were then made, to ascertain the distribution of energy deposition with pin burnup. In all, [ ][a, c] were analyzed. Of these, [ ][a, c], and were studied in detail.

2.5.2 Results

The scoping evaluation of the post rod ejection peaking factors as a function of assemblywise burnup indicated no direct correlation existed. The results indicated that a control rod ejecting from a low burnup fuel assembly can drive surrounding higher burnup fuel assemblies to high F_Q. Conversely, a control rod ejecting from a high burnup fuel assembly can also drive surrounding low burnup fuel assemblies to high F_Q.

It was concluded that the rod ejection event results in a very localized increase in peaking factors. The impact of the rod ejection on neighboring assemblies is related to the reactivity worth of the ejected rod. The results of the scoping analyses demonstrate that only a small percentage of the fuel assemblies can be driven to very high F_Q. These results appear to be independent of the fuel burnup distribution. Since current fuel management strategies typically utilize 1/3 or less of the core inventory as high burnup (thrice-burned), the small number of high burnup fuel assemblies that would experience high F_Q is even fewer.

The results of the scoping study showed that even with the significant conservatisms inherent in this analysis, only a small percentage of fuel assemblies could be driven to enthalpies greater than 30 cal/gm. Furthermore, since the fraction of high burnup fuel is generally limited to less than 1/3 of the core inventory, even fewer high burnup rods experience energy depositions of this magnitude. For a very high worth (bounding case - 900 pcm) control rod ejection*, the number of high burnup rods achieving maximum local energy depositions in excess of approximately 30 cal/gm was conservatively estimated to be no more than 20% of the core. For a more typical reload worth case (600 pcm), the number of high burnup rods achieving similar enthalpies was approximately 7% of the core.

This is supported by several hundred case studies performed for the detailed study. Figure 1 shows a scatter diagram of the worst core design scenario analyzed in the [ ] a, c. In this case; which utilized a projected 24 month fuel cycle, very high discharge rod burnups and a core loading pattern that met all Technical Specification limits; only [ ] a, c, would achieve 30 cal/gm. This case had a [ ] a, c rod ejection worth, and these results appear consistent with the original scoping study. A typical rod ejection case (Figure 2), at [ ] a, c, shows that [ ] a, c would exceed 30 cal/gm.

The peak fuel enthalpy increase will be significantly reduced (by a factor of 2-4) from the above results if a three-dimensional transient methodology is employed.

2.6 Radiological Consequences

The acceptance criteria identified in NUREG-0800 for the radiological consequences of a rod ejection accident are 75 rem thyroid and 6.25 rem whole body (25% of 10 CFR 100 limits). Fuel rods that are evaluated to experience DNB in the transient are assumed to fail and contribute to the dose. If failure of high burnup fuel occurred as a result of the accident, the affected fuel rods would also contribute to doses; however, radiological consequences will increase only when DNB had not occurred for the failed high burnup fuel rod, since a rod can only fail once.

2.6.1 Justification that Defined Dose Limits are not Exceeded

The radiological impact analysis for the rod ejection accident performed in the FSAR typically assumes

* [ ] a, c

10% failed fuel in the reactor core. Although it is predicted that the additional fraction of the core that might experience damage in the event of a rod ejection accident remains small (refer to Section 2.5.2 in Part 1), the effect of increased core damage on calculated doses has been evaluated(2) assuming that 100% of the rods in the core are damaged (versus the 10% normally assumed failure) with the consequent release of the fission products in the fuel clad gap.

The primary release path for the rod ejection accident is from postulated containment leakage. The doses are calculated by conservatively assuming that the activity in the reactor coolant and the activity in the failed fuel rod gap is released to the containment building atmosphere. The typical dose analysis contains several assumptions which represent significant conservatisms. Thus, there is substantial margin which, when removed, will accommodate the increase in source term due to the assumption that all rods are damaged without exceeding dose acceptance limits.

The most limiting doses for this accident are the thyroid doses which have an acceptance criterion of 75 rem (NUREG 0800). The conservative analytical assumptions may be adjusted as described below to reduce the overall thyroid doses for the rod ejection accident:

- Dose Conversion Factor (DCF): Currently reported doses use either the DCFs based on International Commission on Radiation Protection (ICRP) Publication 2 or the ones provided in RG 1.109. The use of ICRP Publication 30 DCFs alone would reduce thyroid doses by a factor of approximately 1.4.

- Plateout: A majority of plant analyses do not currently take credit for plateout. Plateout of half of the elemental iodine released from the gap can reasonably be assumed, resulting in a factor of approximately 2 reduction in thyroid doses alone due to containment leakage.

- Retention in Coolant: A majority of plant analyses do not currently take credit for retention of iodine in the coolant. In accordance with RG 1.4, 91% of the iodine is assumed to be elemental iodine. Plants which take credit for this coolant retention typically assume retention of between 50% and 90% of the iodine. Conservatively, retention of 50% of the iodine released from the fuel would be reasonable. This retention of iodine in the coolant alone would reduce thyroid doses due to containment leakage by a factor of 2.

- Gap Fraction: Consistent with RG 1.77, the fission product gap fraction has been assumed to be 10%. A gap fraction of 2% is a better, but still conservative, estimate of the fission product activity available for release, since the iodine gap fraction peaks at approximately 30 GWD/MTU and decreases in the higher burnup, lower temperature fuel. Use of a 2% gap fraction alone reduces doses by a factor of 5.

The overall reduction in doses that would be achieved by removing the above conservatisms is a factor of 28. Plants which currently take credit for plateout and retention in the coolant have much lower calculated doses than those plants which do not. In summary, by taking into account the revised assumptions as itemized above, the thyroid doses from the rod ejection accident would be well within the 75 rem acceptance limit, even assuming that all fuel rods in the core are damaged.

2.6.2 Meeting the Licensing Basis Dose Limit with Licensing Basis Methodology

[ ] a,c

3.0 Conclusions

The safety significance of hypothetical lower fuel failure limits for high burnup fuel has been assessed. It has been demonstrated that there are significant conservatisms inherent in the analysis of both the rod ejection event and its radiological consequences for Westinghouse reactors within the licensing basis of current NRC-approved methodology. This has been demonstrated in the results of the more detailed study where [ ] a, c would achieve 30 cal/gm for a bounding scenario. These results, even when conservatively combined with the DNB failures, would result in the currently calculated doses increasing by a factor of [ ] a, c. This increase can then be reduced by a factor of [ ]$^{a, c}$ as noted above, using available conservatism within current NRC-approved methodology. These conservatisms in combination are more than sufficient to document the safety of high burnup fuel and that the safety of current plant operations and fuel burnup limits is not compromised by the new RIA test data even if it is assumed that the data is valid and applicable.

4.0 References

1. Letter from R. C. Jones (RSB-NRC) to N. J. Liparulo (Westinghouse), <No Subject>, November 14, 1994.

2. Letter from Marion, A. (NEI) to R. Jones (NRC), "NEI Resopnse to NRC Staff's Request for Information on Reactivity Insertion Accidents," December 28, 1994.

3. Beard, C. L., Chao, Y. A., Huang, P., and Penkrot, J. A., "SPNOVA - A Multidimensional Static and Transient Computer Program for PWR Core Analysis," WCAP-12394-A, June 1991.

4. Taxelius, T. G. (Ed.), "Annual Report - SPERT Project, October 1968 - September 1969," Idaho Nuclear Corporation, IN-1370, June 1970.

5. Liimatainen, R. C. and Testa, F. J., "Studies in TREAT of Zircaloy-2 Clad, UO_2 - Core Simulated Fuel Elements," ANL-7225, January - June 1966, p. 177, November 1966.

6. Risher, D. H., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods." WCAP-7588, Revision 1-A, January 1975.

7. Rajamaki and Wasastjerna, "On the Reactivity Effects of Nuclear Fuel Fragmentation with Reference to the Chernobyl Accident," Nuclear Science and Engineering, 101, 41-47 (1989).

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9. Risher, D. H., and Barry, R. F., "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A, January 1975.

10. Hargrove, H. G., "FACTRAN - A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

11. Liu, Y. S., et al., "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10966-NP-A, September 1986.

12. Letter from J. M. Taylor (EDO-NRC) to NRC Commissioners, "Reactivity Transients and Fuel Damage Criteria For High Burnup Fuel," November 9, 1994.

Figure 1

Cal/GM Versus Burnup - Hypothetical Cycle - EOL HZP

a,c

Figure 2

Cal/GM Versus Burnup - Typical Cycle - EOL HZP

a,c

Part 2: Westinghouse Topical Report Assessments

WCAP-7588-Rev 1-A - "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurizer Water Reactors Using Spatial Kinetics Methods"

SER received, January 1975

Description of Topical Report

This topical report describes the Westinghouse methodology for performing the analysis of the rod ejection accident. The report presents the accident analysis licensing requirements and limits, the reactor protection, a sensitivity study to determine the sensitivity of the accident to variations in input parameters and analysis assumptions, the detailed analysis methodology, and typical results for a wide range of input peaking factors and ejected rod worths. The report demonstrates that the methodology is conservative compared to full three-dimensional transient analysis methods. The report also addresses the RCS overpressure and rods-in-DNB aspects of the event. The report presents curves of the maximum hot spot peaking factor that can be allowed for each ejected rod worth, without exceeding the stated limits; however, these curves are not used in licensing evaluations for individual plants.

Impact Assessment of RIA Issue

The analysis limit used in the topical report for demonstration of continued short and long-term core coolability is a peak fuel enthalpy of 200 cal/gm. This limit is conservative compared to the NRC limit of 280 cal/gm specified in Regulatory Guide 1.77 (Ref. 1). The core coolability limit is not affected by the high burnup fuel RIA issue. In order to determine the number of fuel failures for the offsite dose evaluation, the criterion used was the number of fuel rods in DNB, which is consistent with the requirements of NRC SRP 4.3 (Reference 2). This criterion is more limiting that the 170 cal/gm limit specified in Reference 2 for use on BWRs. The RIA issue does not affect the reactor transient calculation, the results of the sensitivity study, or the RCS pressure or rods-in DNB transient results presented in the report.

Justification for Continued Applicability

Continued applicability of topical report, WCAP-7588 Rev 1-A is justified with respect to the RIA issue. This topical report is currently used as a reference for the licensed Westinghouse Rod Ejection Methodology, which is applied to the analysis of individual plants. The high burnup RIA issue does not affect the methodology, the transient analysis results, or the RCS overpressure or rods-in-DNB evaluation. Therefore, topical report WCAP-7588 Rev. 1-A is not affected by the high burnup RIA issue. In addition, the analysis results presented in Part 1 of this report demonstrate that the analysis of the rod ejection transient using the methods outlined in this topical will still result in radiological dose limits being met, even if high burnup fuel is assumed to fail at low levels of energy deposition.

References

1. USNRC Regulatory Guide 1.77, "Assumptions Used For Evaluating A Control Rod Ejection Accident for Pressurized Water Reactors," May, 1974.

2. USNRC Standard Review Plan, Section 4.2, "Fuel System Design," NUREG-0800, Rev. 2 July, 1981.

WCAP-8963-P-A, "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis"

SER received May 1978.

Description of Topical Report

This topical report describes the basis for the Westinghouse fuel rod internal pressure criterion. This criterion is stated as follows:

> "The internal pressure of the lead fuel rod in the reactor will be limited to a value below that which could cause (1) the diametral gap to increase due to outward cladding creep during steady-state operation and (2) extensive DNB propagation to occur."

The above criterion allows for a small percentage of the fuel rods in the core to operate at an internal pressure in excess of the primary system pressure. The topical report concluded that the revised fuel rod internal pressure design basis is acceptable from a safety standpoint.

Impact Assessment of RIA Issue

The RIA issue addresses the concern that fuel at high burnup could experience failure at peak enthalpy levels below those for which DNB is expected to occur. These failures have been postulated to be due to a pellet-clad interaction type of mechanism. Based on the results of international ramp test programs performed to assess PCI, fuel rod internal pressure has not been identified as a significant factor affecting margin to pellet-clad interaction.

Justification for Continued Applicability

Continued applicability of this topical report with respect to the RIA issue is justified, since the topical report adequately addresses the effect of operation above primary system pressure on Condition III and IV transient events. The primary concern for Condition III/IV events for operation above system pressure is related to clad behavior following a DNB event, while the RIA issue is concerned with additional fuel failure at conditions prior to when DNB would occur, possibly due to a PCI type mechanism. Rod internal pressure is not a significant factor in pellet-clad interaction related fuel failure; therefore, the rod internal pressure limit topical is not affected by the RIA issue of concern.

WCAP-9272-A, "Westinghouse Reload Safety Evaluation Methodology"

SER received May 28, 1985

Description of Topical Report

This topical report describes the reload safety evaluation methodology considered by Westinghouse. The Westinghouse reload safety evaluation methodology consists of (1) a systematic evaluation to determine whether the reload parameters are bounded by the values used in the reference safety analysis, and (2) a determination of the effects on the reference safety analysis when a reload parameter is not bounded to ensure that specified design bases are met. When the above steps identify either a potential unreviewed safety question or the need for a change in the plant Technical Specification, this is identified to the license holder (utility). The utility then makes the final decision regarding the need for prior NRC approval as provided in 10 CFR 50.59. Included in this topical is a description of the Westinghouse reload process and the supporting nuclear, thermal and hydraulic, and safety evaluation methodologies.

Impact Assessment of RIA issue

The purpose of this impact assessment is to address the RIA issue associated with fuel coolability as described in Section 5.3.17 of this topical. It is stated that the consequence of this accident is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel damage. The accident is classified as a Condition IV event and the limiting criteria for this accident are given in the NRC Regulatory Guide 1.77. It is noted that Westinghouse historically has applied the following, more conservative criteria in evaluating this accident:

1) Average fuel pellet enthalpy at the hot spot is 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel*. This limit is conservative compared to the NRC limit of 280 cal/gm specified in the NRC Regulatory Guide 1.77.

* As specified in Reference 6, the limit for average fuel pellet enthalpy at the hot spot is set as 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel. For conservative bounding calculations, 200 cal/gm is used for both fuel types (i.e., irradiated and unirradiated).

2) Average clad temperature at the hot spot is below the temperature at which clad embrittlement may be expected (assumed to be at 2,700 °F for this short duration event)*

3) Peak reactor coolant temperature is less than that which would cause stresses to exceed the faulted condition stress limits.

4) Fuel melting is limited to less than 10% of the fuel volume of the hot spot, even if the average fuel pellet enthalpy is below the limits of criterion (1) above.

In addition to the above, conservative core-related analysis assumptions were made. These included: conservative values of Doppler Power Coefficient and Moderator Temperature Coefficient, minimum values of delayed neutron fraction, maximum initial fuel temperature and maximum hot channel factor.

The high burnup radiological consequence of the RIA issue is not addressed in this topical; however, reference is made to the allowable dose consequence given in Regulatory Guide 1.77.

Justification for Continued Applicability of WCAP-9272-A

Continued applicability of this topical report with respect to the RIA issue is justified, since the high burnup RIA issue does not affect this methodology. The fuel coolability issue (rod ejection accident) is addressed in a conservative manner by using a lower fuel pellet enthalpy limit compared to NRC guidelines. Furthermore, fuel coolability will not be impacted due to the RIA high burnup issue as documented in Part 1. The conservative analysis methodology described in this topical continues to remain applicable and can be continued to be used for Westinghouse reload designs considering the effects of the RIA issue.

* This is not required by the NRC and has been removed as a criteria for some plants.

WCAP-9500-A, "Reference Core Report 17x17 Optimized Fuel Assembly"

SER received May 1981

Description of Topical Report

This topical report serves as a reference core design report for the optimized fuel assembly design consisting of a 17x17 array of fuel rods having a reduced fuel rod diameter. The 17x17 optimized fuel assembly employs a zircaloy spacer grid at all grid elevations except that the top and bottom grids are inconel. These design changes result in an improved water-to-uranium ratio and reduced parasitic absorption, which aid in neutron economy and allow for more efficient use of the fuel. The methodology described in this topical applies not only to 3 and 4 loop 17x17 plants, but generically to plants having other standard fuel rod arrays (i.e., 14x14, 15x15, 16x16 and 17x17; 2-, 3- and 4-loops).

Impact Assessment of RIA Issue

The calculation of the RCCA ejection transient is performed using licensed Westinghouse rod ejection methodology. Input parameters for the rod ejection analysis are conservatively selected on the basis of values calculated for an optimized core. The more important parameters used in the analysis include: ejected rod worths and hot channel factors, reactivity feedback weighing factors, moderator and doppler coefficient, delayed neutron-fraction, and trip reactivity insertion.

The following conservative assumptions are presented in the topical to address the radiological consequences of the postulated rod ejection accident:

- Prior to the accident the plant is assumed to be operating at full power,
- 100 percent of the noble gases and iodines in the cladding gaps of the fuel rods experiencing cladding damage was assumed released to the reactor coolant,
- 50 percent of the iodine and 100 percent of the noble gases in the fuel that melts were assumed released to the coolant,
- The fraction of fuel melting was conservatively assumed to be 0.25 percent of the core,

- Instantaneous mixing occurs in the containment of all the noble gases and 50 percent of iodine activity is released from the coolant,

- No credit is assumed for iodine removal in the containment due to containment sprays, and

- The containment leaks for the first 24 hours at its design leak rate as specified in the technical specification of 0.10 percent/day. Thereafter, the containment leak rate is 0.05 percent per day.

Justification for Continued Applicability of WCAP-9500-A

The analyses in WCAP-9500-A indicated that the described fuel and cladding limits for the rod ejection accident were not exceeded using conservative factors in the analysis. The analyses demonstrated that the fission product release resulted in radiological doses that were well within NRC acceptance criteria. These results continue to remain valid (within currently approved NRC licensed basis) in light of the results of the Westinghouse analysis presented in Part 1 of this assessment. Therefore, the topical report continues to remain applicable.

WCAP-10125-P-A, "Extended Burnup Evaluation of Westinghouse Fuel"

SER received October 1985

Description of Topical Report

This topical report evaluates the effect of extended burnup on the design and operation of Westinghouse fuel. The report provides a comprehensive review of extended burnup fuel design, including fuel rod and fuel assembly mechanical design, nuclear design, and non-LOCA and LOCA safety analyses. Each area of potential concern for extended burnup is examined to show that applicable design criteria, performance models, and design and safety analysis methods are sufficient for design and operation to a target lead rod average burnup of [ ] $^{a, c}$. The Westinghouse extended burnup operating experience and fuel performance data base is presented as justification for the adequacy of Westinghouse design and performance models, and ongoing irradiation programs designed to provide additional confirmation of extended burnup fuel performance were reviewed.

Impact Assessment of RIA Issue

The Westinghouse fuel rod and fuel assembly design criteria presented in WCAP-10125-P-A address Condition I and II operations, and are not affected by the new RIA test data. Similarly, the nuclear design parameters which are assessed, including peaking factors, rod worth, and reactivity coefficients are not affected by the RIA issue. Per the NRC review of PWR FSAR Chapter 15 accident analyses in Reference 1, the only PWR transient which is impacted is the rod ejection event. Other non-LOCA events as well as the LOCA event are not affected by the RIA issue, either because they are not capable of producing a sufficiently high energy deposition to be of concern or because the events are sufficiently slow such that the new RIA test data is not applicable.

Design limits pertinent to the rod ejection event are discussed in the non-LOCA safety analyses section. Fuel coolability is addressed by requiring that the average fuel pellet enthalpy at the hot spot be below 225 cal/gm for non-irradiated fuel and below 200 cal/gm for irradiated fuel*. These limits are more conservative than the limit of 280 cal/gm specified by the NRC in Reg. Guide 1.77. Per Reg. Guide 1.77, it is also assumed that all rods which are predicted to experience DNB will fail and these are accounted for in the radiological safety analyses.

* As specified in Reference 6, the limit for average fuel pellet enthalpy at the hot spot is set as 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel. For conservative bounding calculations, 200 cal/gm is used for both fuel types (i.e., irradiated and unirradiated).

A generic safety assessment of the RIA issue with respect to the rod ejection accident, based on foreign test results which indicated a potential reduction in failure threshold for high burnup fuel, is documented in Part 1 of this assessment. The assessment of the safety significance of hypothetical lower fuel failure limits for high burnup fuel concludes that there are significant conservatisms inherent in the analysis for both the rod ejection event and its radiological consequences for Westinghouse PWRs. These conservatisms in combination are more than sufficient to demonstrate that the safety of plant operations with high burnup fuel, within currently approved NRC licensing bases, is acceptable. Therefore, there is no significant safety impact.

Justification for Continued Applicability

Continued applicability of the extended burnup methodology provided in WCAP-10125-P-A is justified with respect to the RIA issue, since the criteria associated with the design limits pertaining to the rod ejection event are more conservative than the limits specified by the NRC in Reg. Guide 1.77. The assessment of the potential reduction of the fuel failure enthalpy limit for RIA events for the PWR rod ejection analysis, provided in Part 1 of this assessment, provides justification that all applicable safety limits continue to be met within currently approved NRC licensing bases for the RIA event. These conclusions remain valid even for the extended burnup limit of [ ] a, c as licensed through the methodology presented in Reference 2 and as documented in Reference 3.

References:

1. Memorandum, J. M. Taylor to NRC Commissioners, "Reactivity Transients and Fuel Damage Criteria for High Burnup fuel," November 9, 1994.

2. Davidson, S. L. (Ed.), et al., "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-A, October 1994.

3. Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), "Westinghouse Interpretation of Staff's Position on Extended Burnup," NTD-NRC-94-4275, August 29, 1994.

WCAP-10444-P-A, "VANTAGE 5 Fuel Assembly Reference Core Report"

SER received July 1985

Description of Topical Report

This topical presents generic information relative to a combination of improved fuel design features introduced by Westinghouse. The features incorporated in this improved design include: (1) Axial Blankets for improved neutron utilization, (2) Integral Fuel Burnable Absorbers for predictable power distributions and moderator temperature coefficient control, (3) Intermediate Flow Mixer grids for increased thermal/hydraulic margins, (4) Reconstitutable top and bottom nozzles to facilitate fuel rod removal/replacement and for fuel assembly reconstitution and (5) extended burnup for neutron economy. The topical report provides a licensing basis for evaluating the VANTAGE 5 fuel assembly design and serves as the basis for applications incorporating the above VANTAGE 5 design features.

Impact Assessment of RIA Issue

The effects and consequences of the rod ejection analysis are provided in Section 5.4.8.2 of this topical. The calculation of the RCCA ejection transient is performed using licensed Westinghouse rod ejection methodology. Cases are described for the worst ejected rod worth results at different times in life. For all cases, the radiological doses would be expected to be well within NRC acceptance criteria.

Justification for Continued Applicability of WCAP-10444-P-A

Even on a pessimistic basis, continued applicability of WCAP-10444-P-A is justified, with respect to the RIA issue, since the conservatism in this topical indicates that the VANTAGE 5 fuel rod and the clad design limits are not exceeded. The analyses in the topical for the rod ejection accident demonstrated that the upper limits of fission product release result in radiological dose limits that would be expected to be well within NRC acceptance criteria. The results of the Westinghouse analysis performed for this assessment, as presented in Part 1, demonstrate that there are significant conservatisms in the analysis such that the results continue to remain applicable within the licensing basis. Therefore, the topical report continues to remain applicable.

WCAP-10851-P-A, "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations"

SER received May 1988.

Description of Topical Report

This topical report justifies the implementation of updated fuel performance models for cladding creep and growth, fuel swelling and densification and fission gas release using the PAD code as the principal Westinghouse design tool. The PAD computer program iteratively calculates the interrelated effects of temperature, pressure, cladding elastic and plastic behavior, fission gas release, and fuel densification and swelling as a function of time and linear power. Other fuel performance models incorporated into the PAD code have been approved by NRC in prior submittals (References 1, 2, 3 and 4).

Impact Assessment of RIA Issue

The fuel and cladding performance models incorporated in the PAD code have been derived on the basis of empirical fuel and cladding performance data which have been obtained for the whole range of steady state operating conditions up to and exceeding the current licensed target fuel rod average burnup limit. These data are an adequate basis to confirm the continued acceptability of the PAD performance models for their intended applications up to the current NRC-approved burnup limit. The issues associated with the RIA event limits reevaluation have no effect on this topical report.

Justification for Continued Applicability

Continued applicability of the topical WCAP-10851-P-A with respect to the RIA issue is justified, since the PAD code is used primarily for evaluation of steady state fuel performance parameters, though Condition I and Condition II transient fuel duty is considered as part of standard fuel rod design analysis. PAD is also used to generate steady state initial conditions for input to safety analysis calculations. Other than the generation of steady state initial conditions, the PAD code is not used in the analysis of RIA events, and therefore is not affected by this issue. Furthermore, since this code is already benchmarked to data obtained from high burnup fuel (i.e., the impact of high burnup effects, e.g., pellet rim effect, are already implicity included), it can be concluded that this code continues to remain applicable for high burnup fuel.

References:

1. Supplemental information on fuel design transmitted from R. Salvatori, Westinghouse NES, to D. Knuth, AEC, as attachments to letters NS-SL-518 (12/22/73); NS-SL-521 (12/29/72), NS-SL-524 (12/29/72) and NS-SL-543 (1/12/73), (Westinghouse proprietary); and supplemental information on fuel design transmitted from R. Salvatori, Westinghouse NES, To D. Knuth, AEC, as attachments to letters NS-SL-527 (1/2/73) and NS-SL-544 (1/12/73).

2. Hellman, J. M., (Ed.), Fuel Densification Experimental Results and Model for Reactor Application," WCAP-8218-P-A, March, 1975 (Proprietary) and WCAP-8219-A, March 1975 (Non-Proprietary).

3. Miller, J. V., (Ed.), "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8720, October 1976 (Proprietary) and WCAP-8785, October 1976 (Non-Proprietary).

4. Leech, W. J., Davis, D. D., and Benzvi, M. S., "Revised PAD Code Thermal Safety Model," WCAP-8720, Addenda 2, October 1982 (Proprietary).

WCAP-12488-A, "Westinghouse Fuel Criteria Evaluation Process"

SER received July 1994

Description of Topical Report

This topical describes a process and the criteria by which new or modified fuel designs will be evaluated by Westinghouse. Providing that no changes to the Technical Specifications are required because of the fuel design change, the change may then be implemented without prior NRC review and approval, if it meets the criteria specified within this topical. During the review process of this topical, a new section was added providing those criteria for making adjustments to the fuel performance and the material property models based on new data without NRC review and approval. The objective of the above approach is to expedite the NRC review process and reduce the staff and industry resources needed for the review of new fuel designs.

Impact Assessment of RIA Issue

The fuel coolability issue is addressed in the Fuel Coolability Design Criteria (Rod Ejection) section of the topical. The design basis as stated is that violent expulsion of fuel material as a result of an RIA will be avoided in Westinghouse cores and that core coolability will be maintained. The Westinghouse design limit for the average fuel pellet enthalpy is 200 cal/gm for irradiated and 225 cal/gm for unirradiated fuel*. The Westinghouse design limit is more conservative than the 280 cal/gm limit specified in Regulatory Guide 1.77 and has been previously approved in the review of WCAP-7588 Rev 1-A and WCAP-10125-P-A. This limit is not impacted by the new RIA data.

Justification for Continued Applicability of WCAP-12488-A

Core coolability as documented in Part 1 of this assessment will be maintained and RIA events will not be affected by Westinghouse fuel design changes.

* As specified in Reference 6, the limit for average fuel pellet enthalpy at the hot spot is set as 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel. For conservative bounding calculations, 200 cal/gm is used for both fuel types (i.e., irradiated and unirradiated).

WCAP-12610, "VANTAGE + Fuel Assembly Reference Core Report"
SER received July 1, 1991

WCAP-12610, Appendices F and G
SER received October 9, 1991

WCAP-12610, Appendix B, Addendum 1
SER received September 15,1994

Description of Topical Report

This topical report serves as a Reference Core Report for an improved fuel assembly design referred to as the VANTAGE + fuel assembly. In support of providing fuel performance improvements, a new zirconium based fuel rod clad and guide thimble tube alloy, known as ZIRLO™ was introduced. It was demonstrated that this alloy achieved a significant improvement in clad and guide thimble corrosion resistance and dimensional stability under irradiation. This report presents the information required to support the licensing basis for implementation of the VANTAGE + fuel assembly in Westinghouse fuel reload regions for lead rod burnups up to [ ] a, c when licensed by the NRC. Although this topical addresses all the licensing aspects of a Westinghouse core design up to a burnup level of [ ] a, c, the NRC SER approval for this topical report was only up to a burnup limit of [ ] a, c.

Impact Assessment of RIA Issue

In WCAP-12610, Appendix A, the thermophysical properties of ZIRLO™ and Zircaloy-4 clad were shown to be essentially identical [ ] a, c. In order to determine the impact of this slight change, the RCCA ejection event was analyzed at hot full power and hot zero power conditions to demonstrate that any consequential damage to the core or the reactor coolant system will not prevent long-term core cooling and that off-site doses would be within the guidelines of 10 CFR 100.

Justification for Continued Applicability of WCAP-12610

Due to the superior corrosion properties of ZIRLO™ (i.e., reduced oxidation and hydrogen pickup), it will exhibit superior performance to RIA threshold limits. Furthermore, accident analyses performed in WCAP-12610 demonstrated that the ZIRLO™ clad fuel resulted in a small reduction in both the fraction of fuel melted at the hot spot as well as the peak fuel stored energy when compared to the results for Zircaloy-4. Therefore, this topical report continues to remain applicable. These conclusions remain valid even for the extended burnup limit of [ ] [a, c] as licensed through the methodology presented in Reference 1 and as documented in Reference 2.

References:

1. Davidson, S. L. (Ed.), et al., "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-A, October 1994.

2. Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), "Westinghouse Interpretation of Staff's Position on Extended Burnup," NTD-NRC-94-4275, August 29, 1994.

WCAP-13589-A, "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel"

SER received February 1995

Description of Topical Report

This topical report reevaluates the densification power spike factor and the clad flattening design criterion following observations that significant axial pellet gaps do not occur in current Westinghouse fuel. Prior topical reports related to the densification power spike factor and clad flattening analysis methods in References 1 and 2, respectively, have been superseded by WCAP-13589-A.

Impact Assessment on RIA Issue

The RIA issue has no affect on fuel densification behavior, and therefore, this topical and References 1 and 2 are not affected by this issue.

Justification for Continued Applicability

Continued applicability of the elimination of clad flattening and the densification spike factor is justified with respect to the RIA issue. The Westinghouse evaluation of fuel densification effects in WCAP-13589-A was based on fuel performance data obtained at rod average burnup levels which span the full range of current fuel discharge burnup levels, though it is noted that the potential formation of axial fuel column gaps due to fuel densification is primarily an issue at beginning of fuel life. The RIA issue has no affect on fuel densification, and therefore has no affect on the conclusions reached in this topical.

References:

1. Hellman, J. M., (Ed.), Fuel Densification Experimental Results and Model for Reactor Application," WCAP-8218-P-A, March, 1975 (Proprietary) and WCAP-8219-A, March 1975 (Non-Proprietary).

2. George, R. A., Lee, Y. C., and Eng, G. H., "Revised Clad Flattening Model," July 1974.