July 1, 2003

Mr. Hank A. Sepp, Jr.
Manager, Regulatory & Licensing
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT WCAP-15806-P AND WCAP-15807-NP, "WESTINGHOUSE CONTROL ROD EJECTION ACCIDENT ANALYSIS METHODOLOGY USING MULTI-DIMENSIONAL KINETICS" (TAC NO. MB4521)

Dear Mr. Sepp:

By letter dated February 25, 2002, Westinghouse Electric Company (Westinghouse) submitted licensing Topical Reports (TRs) WCAP-15806-P and WCAP-15807-NP, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics," to the NRC for review and approval. The objective of this TR is to present for licensing approval, an improved methodology for use in the Final Safety Analysis Report (FSAR) rod ejection accident analysis for pressurized water reactors. This application methodology is based on a three-dimensional core representation, using the NRC-approved core neutron kinetics code SPNOVA and the NRC-approved core thermal-hydraulic code VIPRE-01.

The staff has completed its review of the subject TR. The TR is acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the report and in the associated NRC safety evaluation (SE), which is enclosed. The enclosed SE defines the basis for acceptance of the TR. The staff has noted both the proposals for new fuel enthalpy criteria and the Westinghouse stated interim criteria, but will not endorse any new criteria at this time. The staff review of this TR covers only the justification of the improved rod ejection application methodology to adequately and conservatively represent the rod ejection event.

Pursuant to 10 CFR 2.790, we have determined that the enclosed SE does not contain proprietary information. However, we will delay placing the SE in the public document room for ten working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any information in the enclosure is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.790.

We do not intend to repeat our review of the matters described in the subject report, and found acceptable, when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. Our acceptance applies only to matters approved in the report.

In accordance with the guidance provided on the NRC website, we request that Westinghouse publish an accepted version within three months of receipt of this letter. The accepted version shall incorporate (1) this letter and the enclosed SE between the title page and the abstract, (2) all requests for additional information from the staff and all associated responses, and (3) a "-A" (designating "accepted") following the report identification symbol.

Should our criteria or regulations change so that our conclusions as to the acceptability of the report are invalidated, Westinghouse and/or the licensees referencing the TR will be expected to revise and resubmit their respective documentation, or submit justification for the continued applicability of the TR without revision of their respective documentation.

Sincerely,

/RA/

Herbert N. Berkow, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 700

Enclosure: Safety Evaluation

cc w/encl:
Mr. Gordon Bischoff, Project Manager
Westinghouse Owners Group
Westinghouse Electric Company
Mail Stop ECE 5-16
P.O. Box 355
Pittsburgh, PA 15230-0355

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Herbert N. Berkow, Director
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Project No. 700

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-15806-P AND WCAP-15807-NP,

"WESTINGHOUSE CONTROL ROD EJECTION ACCIDENT ANALYSIS METHODOLOGY

USING MULTI-DIMENSIONAL KINETICS"

WESTINGHOUSE ELECTRIC COMPANY

PROJECT NO. 700

1.0 INTRODUCTION

By letter dated February 25, 2002 (Reference 1), Westinghouse Electric Company (Westinghouse), submitted Topical Reports (TRs) WCAP-15806-P and WCAP-15807-NP, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics," to the NRC for staff review and approval. The purpose of the TR is to present for licensing approval an improved application methodology for the Final Safety Analysis Report (FSAR) rod ejection accident analysis, based upon a realistic three-dimensional core representation using the NRC-approved core neutron kinetics code SPNOVA (References 2 and 3) and the NRC-approved core thermal-hydraulic code VIPRE-01 (VIPRE) (References 4 and 5). This improved methodology can be used to replace the currently approved methodologies (References 6 and 7), which are primarily based on conservative one-dimensional (1-D) axial core neutron kinetics methods. The proposed methodology is for application to pressurized water reactors (PWRs) and the TR describes the three-dimensional (3-D) methodology for the analyses of the rod ejection accident. The two referenced codes, SPNOVA and VIPRE-01, are coupled to pass the necessary data for the nuclear, fluid and fuel temperature calculations.

The phenomena that are of importance in determining the consequences of a rod ejection accident, particularly in high burnup fuel cores, have been identified in the NRC-sponsored Phenomenon Identification and Ranking Tables (PIRT) exercise for this accident (Reference 8). The improved 3-D methodology presented in the Westinghouse TR is stated to be consistent with the identified phenomena.

As discussed in Section 2.2 of this safety evaluation (SE), ongoing reactivity insertion accident (RIA) experiments at the CABRI research facility (Reference 9) and other tests of prompt-critical events with irradiated fuel rods have indicated that the current NRC peak fuel enthalpy criterion of 280 calories per gram (cal/g) may not be conservative for high fuel burnups (> 50 GWd/tU). As part of the industry response to the recent experimental results, revised acceptance criteria have been developed by the Electric Power Research Institute (EPRI) through Working Group 2 of the EPRI Robust Fuel Program. This approach is described in an EPRI TR (Reference 10). The proposed limiting criteria for the allowable fuel enthalpy vary as a function of either fuel burnup or clad oxidation and have been submitted by industry through the Nuclear Energy Institute (NEI) requesting NRC review and endorsement (Reference 11). The proposed revised criteria would apply to both hot zero power (HZP) and hot full power (HFP) RIAs in both PWR and boiling water reactors (BWRs), and would affect the fuel failure threshold limit as well as the core coolability limit.

In the interim, until any new limits are approved, Westinghouse will apply additional conservatism to the 3-D methodology for the peak fuel enthalpy limit for the HZP rod ejection transient. This interim conservative adjustment limits the peak fuel enthalpy increase to 100 cal/g, which is less than the anticipated future criteria. Since the current Westinghouse criteria use an absolute limit of 200 cal/g for the peak radially averaged fuel enthalpy, and the HZP nominal fuel enthalpy is about 17.5 cal/g, this is equivalent to reducing the current Westinghouse design limit by 82.5 cal/g (41.25 percent). For the full power or non-prompt-critical cases they will continue to use the current licensed peak fuel enthalpy criteria until new criteria are approved.

The staff has noted both the proposals for new fuel enthalpy criteria and the Westinghouse stated interim criteria, but will not endorse any new criteria at this time. The staff review of this TR covers only the justification of the improved rod ejection application methodology to adequately and conservatively represent the rod ejection event.

2.0 REGULATORY EVALUATION

2.1 Current Regulatory Requirements

The staff has reviewed the Westinghouse TR in accordance with the applicable regulations and guidelines contained in the following documents:

- 10 CFR Part, 50 Appendix A, General Design Criteria (GDC) for Nuclear Power Plants;
- 10 CFR Part 100, Reactor Site Criteria;
- U. S. AEC, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," Regulatory Guide (RG) 1.77, May 1974;
- U. S. NRC, Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, July 1981.

Section 4.2 of the SRP specifies two licensing criteria for reactivity insertion accident events. A fuel coolability limit was established to restrict the energy deposition in the fuel rod to preclude fuel melt, fragmentation and dispersal. This meets the requirements of GDC 28, "Reactivity Limits," as cited in RG 1.77, that the coolability limit for violent expulsion of fuel should be 280 cal/gm of UO_2. A fuel rod failure threshold was established to limit fission product release during postulated accidents to meet the specific requirements of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC 19, "Control Room." Cladding failure is assumed to occur when the heat flux is greater than or equal to the departure from nucleate boiling (DNB) ratio for zero power, low power and full power RIA events in PWRs.

The rod ejection accident is a design basis reactivity insertion event for PWRs. An analysis of the radiological consequences of the event is typically presented in Chapter 15 of the plant updated FSAR. Acceptable analysis methods and criteria for the event are described in

RG 1.77, and the NRC review process for the FSAR analysis is described in Section 15.4.8 of the SRP.

The allowable dose consequences for the event are as given in RG 1.77. The number of fuel failures for the dose evaluation is based on the number of fuel rods reaching DNB, as discussed in Section 15.4.8 of the SRP.

2.2 Recent Developments

As discussed previously, ongoing RIA experiments at the CABRI Research Facility and other tests of rapid reactivity insertion (prompt-critical) events with irradiated fuel rods have indicated that the current NRC peak fuel enthalpy criterion of 280 cal/g may not be conservative for high burnups (> 50 GWd/tU).

On October 4, 1993, the Office of Nuclear Reactor Regulation (NRR) sent a user-need memorandum to the Office of Nuclear Regulatory Research (RES) involving high-burnup fuel issues. The memorandum requested work in three areas: (1) fuel performance model changes, (2) fuel performance code updates, and (3) evaluation of the fuel failure thresholds. Subsequently, an "Agency Program Plan for High-Burnup Fuel" was prepared and sent to the Commission on July 6, 1998. An attachment to the Agency Program Plan describes the high burnup fuel research schedules and resources related to several issues, as well as assessments of their safety significance. This plan was discussed with the Advisory Committee for Reactor Safeguards (ACRS) in 1998.

One of the activities listed in these documents relates to fuel damage thresholds and criteria for various events. For RIAs, GDC 28 states that:

> "The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effect of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core."

Section 4.2 of the SRP describes specific acceptance criteria for RIA analyses to demonstrate that GDC 28 is satisfied. The SRP describes a coolable geometry as:

> "…[retaining a] rod-bundle geometry with adequate coolant channels to permit removal of decay heat."

For a severe RIA, such as rod ejection in a PWR or a rod drop in a BWR, the SRP provides a specific acceptance criterion to prevent the violent expulsion of fuel and ensure core coolability:

> "…To meet the guidelines of Regulatory Guide 1.77 as it relates to preventing wide-spread fragmentation and dispersal of the fuel and avoiding the generation of pressure pulses in the primary system of a PWR, a radially averaged enthalpy limit of 280 cal/g should be observed. This 280 cal/g limit should also be used in BWRs."

Although the SRP limit is set at 280 cal/g for coolability, the reactor vendors have adopted their own lower design values to ensure coolability (approximately 225 cal/g) to assure conservatism.

The SRP also contains another lower limit of 170 cal/g that is used as a threshold for cladding failure to determine that the radiological releases from an RIA remain well below 10 CFR Part 100 limits.

Work on RIA testing is continuing through experiments at the CABRI facility in France, and at the Nuclear Safety Research Reactor (NSRR) in Japan. As a result of a fuel failure during a test at CABRI in 1993 (REP Na-1), and one in 1994 (HBO-1) at NSRR, the NRC recognized that high burnup fuel cladding might fail during an RIA at a lower enthalpy than the 280 cal/g limit currently specified in RG 1.77. However, generic analyses performed by all of the reactor vendors have indicated that the fuel enthalpy during RIAs will be much lower than the RG 1.77 limit, based on 3-D neutronic calculations. For high burnup fuel which no longer contains significant reactivity, the peak fuel enthalpy calculated using the 3-D models is expected to be lower than the value of 100 cal/g that was discussed and recommended by the RES in Research Information Letter No. 174 dated March 3, 1997, as a potential replacement for the RG 1.77 value.

The March 3, 1997 Research Information Letter and the attachment to the Agency Program Plan describe both the fuel cladding failure threshold and the core coolability criteria limits, and suggest that new interim criteria be established to reflect the experience from the CABRI and NSRR experiments. The suggested interim criteria include a cladding failure limit of 100 cal/g. In addition, the coolability limit would be left at 280 cal/g for a fuel burnup of less than 30 GWD/t, while higher burnup fuel would be subject to a "no cladding failure" limit. The plan also noted that the 280 cal/g value might be lowered to about 230 cal/g, but explained that this was not a high-burnup issue.

The staff has noted both the proposals for new fuel enthalpy criteria and the Westinghouse stated interim criteria, but will not endorse any new criteria without appropriate review and approval. The staff's review of this TR covers only the justification of the improved rod ejection application methodology to adequately and conservatively represent the rod ejection event.

Current reactor design practices result in reactor core designs and operation where the calculated peak enthalpy during RIAs remains well below the 280 cal/g (and the 230 cal/g) limit. These calculations were performed using conservative, one-dimensional neutronic methods. More realistic calculations of these scenarios by both the reactor vendors and NRC contractors show that the most reactive fuel (usually new fuel) will not suffer a fuel enthalpy rise during an RIA above the proposed 100 cal/g interim cladding failure limit. Although some of the fuel failures that occurred in CABRI and NSRR have released fuel material outside the cladding, none have shown the sort of expulsion of molten fuel that threaten the integrity of the reactor coolant pressure boundary. The currently identified future tests at these facilities would not address fuel fragmentation and dispersal effects at high temperatures. Therefore, they would not provide information relevant to the 280 cal/g (or the 230 cal/g) coolability limit. Instead, these experiments would only provide further experimental confirmation for the proposed 100 cal/g fuel failure threshold limit.

Current results of analyses to calculate off-site releases during RIAs do not challenge licensing limits, and the staff does not believe that changing the cladding failure threshold limit from 170 cal/g to 100 cal/g would significantly affect these results. If the analyses are done with realistic methods, no cladding failure is expected using either limit.

3.0 TECHNICAL EVALUATION

3.1 Accident Description and Limits

The rod ejection accident is described as the mechanical failure of a control rod mechanism pressure housing such that the reactor coolant system pressure would cause the ejection of a partially- or fully-inserted control rod and drive shaft to its fully withdrawn position. If the reactor is at or near critical, the consequences of this mechanical failure are a rapid reactivity insertion and core power increase together with an adverse core power distribution, possibly leading to localized fuel rod damage. The power increase is arrested primarily by the negative reactivity due to the Doppler feedback resulting from fuel heatup, and the transient is terminated by a reactor trip which is initiated shortly after the beginning of the transient.

Due to the extremely low probability of a rod cluster control assembly ejection, this accident is classified as a Condition IV (limiting fault) event as defined by the American Nuclear Society Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants (ANSI N18.2-1973) (Reference 12).

For a typical PWR, the position of all control rods is continuously indicated in the control room, and an alarm will occur if one rod deviates from its bank demand position by more than 5 percent of the indicated rod position. There are low and low-low insertion limit monitors with visual and audio signals. Operating instructions require boration at the low limit alarm, and emergency boration at the low-low limit alarm. These alarm functions ensure that the initial conditions prior to a control rod ejection accident will not be worse than the cases analyzed. Should a rod ejection event occur while the reactor is at or near critical, the transient will typically be terminated by one or more of the automatic features of the reactor protection system. The protection system features are typically described in the FSAR for each plant.

The physical limits of this accident are that any consequential damage to either the core or the reactor coolant system must not prevent long-term core cooling and that any off-site dose consequences must be within the guidelines of 10 CFR Part 100. More specific criteria are applied to ensure that there is no significant fuel dispersal in the coolant, gross fuel lattice distortion, or severe shock waves. Acceptable limiting criteria for this accident have been defined in RG 1.77. However, Westinghouse typically has applied the following conservative criteria:

1. The average fuel pellet enthalpy at the hot spot shall be below 200 cal/g (360 BTU/lbm) for irradiated or unirradiated fuel. (Note: RG 1.77 allows a higher limit of 280 cal/g.)

2. The peak reactor coolant pressure shall be less than that which would cause stresses to exceed the faulted condition stress limits. (Note: Westinghouse

plants meet the more stringent emergency condition stress limits as specified by RG 1.77.)

3. Fuel melting will be limited to less than the innermost 10 percent of the fuel pellet at the hot spot, even if the average fuel pellet enthalpy at the hot spot is below the limits of Criterion 1 above.

Some plants use enthalpy as alternate criteria for fuel failure (200 cal/g for clad failure and 250 cal/g for incipient centerline melting, respectively). These criteria have been used in plant FSARs which have been reviewed and accepted by the NRC.

Note that the EPRI TR (Reference 10) proposes a fuel rod failure threshold of 170 cal/g for unirradiated fuel, decreasing to approximately 125 cal/g at a rod average burnup of 62 GWd/MTU. The proposed limit for core coolability (centerline melt) ranges from 230 cal/g for unirradiated fuel to approximately 200 cal/g at a rod average burnup of 62 GWd/MTU.

The staff has noted the proposals for new fuel enthalpy criteria and the Westinghouse current conservative criteria, but will not endorse any new criteria at this time. The staff's review of this TR covers only the justification of the improved rod ejection application methodology to adequately and conservatively represent the rod ejection event.

3.2 Current Methodology

The current Westinghouse licensing basis analysis methodology for the FSAR rod ejection accident is based upon the application of conservative 1-D axial core neutron kinetics methods, and is described in the NRC-approved TR WCAP-7588, Rev. 1A (Reference 6) and in the NRC-approved TR CENPD-190 (Reference 7).

These currently approved application methods use a 1-D nuclear design model to calculate the ejected rod worth and associated peaking factor, and the Doppler and moderator feedback. In order to bound future reload cycles for an individual plant, the vendor may perform, at the request of the licensee, a more conservative bounding analysis utilizing key parameters which are not expected to be exceeded. These parameter values are then utilized with appropriate uncertainty allowances in a bounding transient analysis calculation of the core and fuel behavior. The key parameters are checked every cycle to ensure that the analysis remains bounding.

The current static analysis of the ejected rod is performed in 3-D using an adiabatic feedback model which maintains the Doppler and moderator feedback at the initial (pre-ejection) condition. This will generate a peaking factor which is larger than would be calculated with a transient calculation that includes more realistic feedback. The various control rod bank locations are evaluated to determine the location of the worst ejected control rod assemblies in the core at various fuel burnup levels.

The current effective delayed neutron fraction for the entire core is obtained by weighting the delayed neutron fraction for different fissionable isotopes by the fraction of fissions in each isotope and the power sharing in the core.

In the current static calculation, key safety parameters of a single rod ejection are evaluated at the beginning and end of each reload fuel cycle using the three-dimensional nodal code ANC (References 13, 14, 15, and 16). The evaluation considers both full power and zero power initial conditions with the control banks at their respective insertion limits. Spatial peaking factors, control bank worths and ejected rod worths are derived from multi-dimensional neutronic calculations. In the static calculation of the rod ejection event, the peaking factors are calculated with an adiabatic assumption. That is, the nuclear feedback, both Doppler and moderator, is established during the initial condition calculation and this feedback is not allowed to vary when the rod is ejected.

The current fuel temperature (Doppler) coefficient is defined as the change in reactivity per degree change in effective fuel temperature. It is primarily a measure of the Doppler broadening of U238 and Pu240 resonance absorption peaks. The fuel temperature coefficient is calculated by performing two-group multi-dimensional neutronic calculations. Moderator temperature is held constant and power level is varied. The spatial variation of fuel temperature is taken into account by calculating the effective fuel temperature as a function of local power density throughout the core.

The current transient calculation of the rod ejection is performed in two stages, first an average core channel calculation and then a hot rod calculation. The average core calculation is performed using the approved TWINKLE (Reference 17) code in the 1-D (axial) mode to determine the core average power generation with time including the various core feedback effects, i.e., Doppler temperature and moderator temperature reactivities. A Doppler weighting factor is applied to the Doppler feedback to compensate for the missing dimensions. Enthalpy and temperature transients at the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation using the approved FACTRAN (Reference 18) code. The power distribution calculated without feedback is conservatively assumed to persist throughout the transient. The DNB heat flux is not calculated; instead DNB is conservatively assumed to occur near the start of the transient. Cases at the beginning and end-of-cycle (EOC), at full and hot zero power initial conditions, are analyzed. Input values for the ejected rod worth, peaking factors, delayed neutron fraction and feedback coefficients are determined using the nuclear design methods as discussed in the preceding subsection. Appropriate uncertainty allowances are applied to the parameters used in the calculation. A more detailed discussion of the current method of analysis can be found in typical plant FSARs, and in the Westinghouse rod ejection TR (Reference 6).

3.3 New Methodology

3.3.1 Nuclear Model

The most significant difference between the current NRC-approved analysis method and the new improved analysis method for this accident is the change from a 1-D to a 3-D core neutron kinetics and feedback model. This eliminates the need to apply Doppler weighting factors to the core kinetics calculation to simulate the increased Doppler feedback due to the skewed power distribution following the ejection of the rod. It also eliminates the conservative assumption of a constant no-feedback value of the ejected rod peaking factor in the hot rod calculation. The computer codes used with the 3-D revised analysis method have already been reviewed and approved by the NRC. The nuclear model is based on the NRC-approved Westinghouse SPNOVA code (References 2 and 3).

In the new methodology, a detailed 3-D transient nuclear model is used for the analysis. The use of a transient model gives a more accurate analysis of the actual transient. The major changes to the model are the input of the kinetics parameters. These include the delayed neutron fractions, the delayed neutron decay constants, the neutron velocities for each energy group, and the transient driver functions which initiate the transient and the reactor trip. These parameters enable the ongoing calculation of the peaking factors as the transient unfolds, instead of assuming a constant peaking factor value throughout the transient. The overall delayed neutron fraction can be further adjusted for conservatism by applying a fixed multiplier to the individual node-by-node values.

Westinghouse proposes to use conservative preconditions for both HFP and HZP analyses that include time-of-life effects, rod insertion effects, potential xenon distributions and allowable control rod positions.

Typical cycle-specific key parameters that Westinghouse analyzes include:

- Ejected rod worth
- Ejected rod peaking factors
- Delayed neutron fractions
- Doppler temperature coefficient
- Moderator temperature coefficient

Uncertainty allowances are applied in determining these key parameters to provide conservative limiting analysis values.

3.3.2 Thermal-Hydraulic Feedback Model

The new core thermal-hydraulic and fuel rod models, as well as the hot rod model, are based on the NRC-approved Westinghouse version of the VIPRE-01 code and associated methods (References 4 and 5).

The thermal-hydraulic model used in the reactor core kinetics calculation includes the time dependent effects of reactor coolant flow, heat transfer from the clad to the coolant, and direct heat generation in the coolant. The calculation is performed with the VIPRE code using a mesh structure consistent with the nuclear analysis mesh structure. Since the rod ejection event is a very rapid transient, most of the thermal energy is retained in the fuel rod. Thus, the coolant temperature increase is relatively small. Some of the fission energy is deposited directly into the coolant through the slowing down of the fission neutrons and the absorption of gamma rays accompanying the fission process. This can become significant for transients that result in very high peak nuclear power increases, and is taken into account in the calculation.

Since the principal feedback mechanism is due to the Doppler feedback, thermal hydraulic modeling assumptions are made for the node average which conservatively maximizes heat transfer from the clad to the coolant. This includes the assumption of full reactor coolant flow (no reactor coolant pumps (RCPs) out-of-service) in the HZP rod ejection case, and no initiation of post-DNB heat transfer.

For the HZP case, the major factor turning the transient around is the Doppler feedback, which is directly related to the rapid increase in fuel pellet temperature. Under prompt-critical conditions, the fuel rod transient is nearly adiabatic; that is, the details of the heat transfer from the fuel to the clad and from the clad to the coolant is not of high importance. For the much slower HFP case, both the Doppler and moderator feedback are important, particularly at the EOC with the most negative moderator temperature coefficient. Therefore, for the HFP case, the fuel rod internal heat transfer and the heat transfer to the clad and coolant are important and are accounted for in the model.

The average fuel rod model for the feedback calculation is performed with the VIPRE code using a multizone fuel pellet representation for the fuel rod in each neutronic/thermal-hydraulic core node. The fuel pellet-to-clad gap heat transfer is calculated using the dynamic gap conductance model in VIPRE that accounts for changes in the fuel dimensions and fill gas pressure with temperature. Design values of pellet radial power distributions, based on an assembly-average burnup, are input for each fuel assembly. The resonance effective temperature is generated at each spatial node from the radially varying temperatures using design values of the T_{eff} weighting function.

To conservatively bound the transient and cover the uncertainties in the actual T_{eff} calculation, an input multiplier on the Doppler feedback cross section adjustment is applied. This allows a uniform uncertainty allowance to be applied on the Doppler feedback adjustments. Similarly, the core parameters are adjusted to make the moderator temperature more positive to conservatively represent the moderator density feedback effect.

3.3.3 Peak Enthalpy Calculations

The hot fuel rod thermal calculation is performed independent of the node average fuel temperature feedback calculation, with additional conservatism applied to the modeling and initial conditions in order to maximize the increase in fuel temperature and enthalpy. The key limit for the accident is the calculated radially-averaged peak fuel enthalpy (RAPFE), or the maximum change in fuel enthalpy. The hot fuel rod model uses the same fuel pellet and clad mesh description as for the average rod.

The hot fuel rod model is based on the NRC-approved model described in the Westinghouse VIPRE modeling TR (Reference 4), and is similar to the model used in the approved FACTRAN code (References 6 and 18). It represents the hottest fuel rod from any assembly in the core. The pellet-to-clad gap heat transfer is calculated using the dynamic gap model in VIPRE, which is comparable to the conservative, NRC-approved FACTRAN transient gap model. In both cases, the model has been calibrated against the design fuel rod temperatures generated by an acceptable fuel rod performance code such as the approved PAD program (Reference 19), using the method described above for the average rod model. Consistent with current plant licensing applications, the heat transfer to the coolant is calculated using the Dittus-Boelter correlation for single phase forced convection, the Thom correlation for nucleate boiling, and the Bishop-Sandberg-Tong correlation (Reference 20) for transition and film boiling beyond DNB. In order to maximize the temperature and enthalpy increase within the fuel pellets, the hot spot of the fuel rod is in post-DNB film boiling during the transient. The Baker-Just correlation (Reference 21) is used to account for heat generation in the cladding material due to the zirconium-water reaction.

A benefit of this new method compared to the current licensed analysis methodology based on 1-D kinetics is that more realistic time-dependent core average power, rod peak power, and axial power distributions are taken directly from the 3-D kinetics results for the hot rod transient calculation instead of using a more conservative constant peak power value. The calculation can be performed for the hot rod in the hottest assembly (the one with the peak fuel enthalpy), or for different fuel assemblies in the core at various levels of burnup.

3.4 Neutronic Parameters and Core Initial Conditions

Key neutronic parameters, such as Doppler and moderator feedback, are determined in order to maximize the rod ejection effect. For example, the total Doppler feedback coefficient is at a maximum at the beginning of a cycle, and at a minimum at the end of a cycle. The moderator temperature coefficient becomes more negative with decreasing boron concentrations, and with increasing temperatures.

The effect of the ejected rod worth is dependent on the arrangement of fuel assemblies within the core, the control rod pattern, the axial power distribution due to burnup and xenon effects, and the allowed control rod insertion limits. If the control rods are partially inserted, the ejected rod worth increases for power distributions which skew the power to the top of the core. The core power distribution is naturally skewed slightly to the bottom of the core at full power due to the temperature feedback. Thus a burnup skew builds into the core with cycle depletion, and is at a maximum at the end of the cycle. For this reason, the EOC condition provides the most limiting axial power distributions. The ejected rod peaking factor will also increase as the ejected rod worth increases. Thus if the ejected rod worth is conservatively increased, this also conservatively increases the ejected rod peaking factor without applying a separate additional conservatism on the peaking factors.

There are two other key core operating parameters, besides the time of life and depletion model, that have a significant effect on the ejected rod worth and peaking factors, and that can be adjusted as part of the initial conditions for the analysis. These are the xenon distribution and the control rod bank positions.

The axial xenon distribution can have a significant impact on the ejected rod worth and the ejected rod peaking factor for partially-inserted rod banks. Xenon distributions that force the power distribution more to the top of the core are more limiting since they increase the axial peaking factor and increase the worth of the rod that is being ejected.

At HFP, there is a nominal operating range in which the reactor is expected to operate. This band of operation is typically defined by axial offset limits. Those limits can be either a band around the equilibrium value, or absolute limits. Since HFP operation is the expected norm, a limiting axial xenon distribution is used in the precondition for the rod ejection evaluation. This is a highly unlikely situation since it would result in the operator having no room to control the reactor, but it provides a conservative bound.

At HZP, there are no limitations on the axial power distribution. HFP equilibrium xenon, and zero xenon conditions result in very similar ejected rod worths. Top-skewed xenon distributions decrease the ejected rod worth. Mildly bottom-skewed xenon distributions slightly increase the ejected rod worth. Therefore, an artificially skewed-to-the-bottom xenon distribution is chosen which increases the ejected rod worth beyond the zero xenon case to conservatively account for adverse power distributions.

The control bank insertion also has a significant role in the ejected rod worth. The ejected rod worth generally increases with increased control bank insertion for the same axial power shape. Thus, deeper insertions increase the ejected rod worth. Technical specification limits on control rod insertion, and the control rod insertion limit alarms, ensure that it is highly unlikely that the control rods will be inserted beyond the specified limits. Thus, the assumption that the control banks are at their insertion limit is a conservative initial condition for the rod ejection accident. In order to perform a more bounding analysis where a higher bounding ejected rod worth is desired, a deeper insertion can be utilized, and/or the control rod cross sections can be adjusted.

3.5 Reload Safety Evaluation (RSE)

The currently approved Westinghouse RSE methodology (Reference 22) uses a bounding analysis approach which is characterized by utilizing key parameters determined from a static analysis to determine whether a detailed transient case should be analyzed for the current cycle. The key parameters for the rod ejection transient that vary from cycle to cycle, assuming no change in plant operating characteristics or fuel type, are:

- Ejected rod worth
- Ejected rod peaking factors
- Delayed neutron fractions
- Doppler temperature coefficient
- Moderator temperature coefficient

The reference bounding safety evaluation calculation may be performed with more conservative values for these key parameters through the use of more conservative allowances. If a reload safety evaluation, using the cycle-specific static values, is less limiting than the reference bounding analysis of record, then a cycle-specific transient analysis does not need to be performed.

If the plant operating characteristics (power, temperature, pressure, flow, design peaking factors, etc.) or the fuel type or characteristics (clad diameter and thickness, pellet diameter, grid) should change, this is identified in the RSE methodology, and more variables are evaluated to determine if the analysis must be repeated.

These parameters are all associated with the reactor core, and the use of the 3-D methodology requires the generation of a detailed 3-D core model. Thus, these parameters are implicitly handled by the nuclear model.

The rod ejection transient is a very rapid transient, and as such there is no recirculation loop impact on the course of the transient. Therefore, variations in the primary and secondary system characteristics have no impact on the course of the transient.

The control rod ejection time is chosen to be fast enough to be of no consequence to the actual transient, so this time is insensitive to the control rod cluster geometry and rodlet absorber composition. The modeling of the ex-core detectors and the determination of the time of reactor trip are dependent on the type of plant, ex-core detector geometry and the protection system setpoints and allowances. The position of the tripped rods versus time is also dependent on the control rod cluster geometry and rodlet composition. The trip behavior thus is plant specific, and is modeled as such; and, as noted in the Westinghouse TR, the control rod trip has only a secondary impact on the limiting parameters calculated for the transient.

The currently approved RSE methodology in WCAP-9272-A continues to be applicable to analyses with the approved 3-D kinetics/thermal-hydraulics methodology (SPNOVA/VIPRE) and its application is consistent with both the current RG 1.77 criteria and any new proposed criteria. Therefore, the use of this methodology is acceptable to the staff.

The basic 3-D rod ejection methodology, as defined in the Westinghouse submittal, is generally applicable to all Westinghouse and Combustion Engineering nuclear steam supply systems (NSSS) pressurized water reactors. Application of this methodology to non-Westinghouse NSSS plants would, however, require additional justification, including submittal of sample calculations, uncertainty analyses, and applicable benchmarking results.

3.6 Sample Application Analyses

The plant selected for the sample application calculations is a Westinghouse 3-loop core with a 17x17 fuel assembly and an 8-cluster lead control bank (Bank D). This core design has typically been one of the most limiting for the HZP rod ejection transient.

3.6.1 Hot Zero Power

Analyses were performed to determine the worst ejected rod cluster and the appropriate starting conditions for the transients. Additional calculations were performed for this study to demonstrate the sensitivity of various parameters.

The HZP analyses were performed at the EOC conditions. Several sample conservative analyses were conducted:

- Statistical reload analysis: A typical single reload cycle analysis with key parameter conservative allowances added to the results statistically. (Labeled Base Case)

- Deterministic reload analysis: A typical single reload cycle analysis with conservative allowances included for each key parameter separately. (Labeled All Allowances)

- Bounding analysis: A typical multi-cycle bounding analysis which increases the key parameters prior to including the key parameter conservative allowances to create an analysis that is expected to bound most future reload cycles. (Labeled Bounding Case)

The key parameters used in the evaluations and the results are provided in Table 3.3 of the Westinghouse submittal. The key parameter sensitivities are presented in Table 3.4 of the submittal. Results of the analysis showed that the application of all the uncertainties in the key parameters together increases the core average peak power considerably, and that the peak fuel delta enthalpy also increases compared to the conservative base case without those uncertainties. The individual perturbations, when combined as the square root of the sum of the squares, provide the same impact as the transient calculation which included all of them together. This also confirms that the individual effects were independent.

The bounding analysis is a severe case with the adiabatic ejected rod worth being slightly less than $2.00 in reactivity. The Doppler multiplier has been adjusted lower, and the feedback effect is further conservatively bounded. This bounding reference calculation produces results which are far more limiting than is typically assumed. The results are summarized in Figures 3.7 and 3.8 and in tabulated form in the submittal.

The staff finds the analyses and results presented in WCAP-15806-P to be acceptable, since the overall modeling is improved and all key parameters are represented conservatively.

3.6.2 <u>Hot Full Power</u>

Both beginning-of-cycle (BOC) and EOC cases were analyzed at HFP. Results of the analysis show that the ejected rod reactivity worths are almost the same, but with different ejected rod core locations and with different parameters associated with the time in life. In both cases, the key parameter conservative uncertainties were included: the Doppler feedback and delayed neutron fractions were reduced by a conservative multiplier, the moderator temperature coefficient was made more positive by increasing the soluble boron concentration, and with a xenon distribution giving a limiting positive axial offset with the control rods deeply inserted.

Both HZP and HFP cases showed the characteristic rapid increase in power until the Doppler feedback balances the reactivity insertion, followed by a decrease to the new equilibrium power. The control rod trip then initiates the shutdown. The HFP transient results indicated a different profile than the HZP transient for the following reasons:

- The HFP ejected rod worth is much less than the delayed neutron fraction, and the transient is not a prompt event.

- The reactor is already operating at power so, in general, the fuel temperatures are already significant and pellet clad contact has occurred. Thus, the heat transfer is very good between the pellet and the coolant.

- Also, since the reactor is at power, there is no delay time, as seen in the HZP cases, for the flux level to increase into the significant range.

The summary of the parameters and results are provided in Tables 3.5 and 3.6 of the Westinghouse submittal. Profiles of the core average power, peak fuel enthalpy and minimum DNB ratio are shown in Figures 3.9 through 3.11 of the same submittal.

The staff finds the analyses and results presented in WCAP-15806-P to be acceptable, since the overall modeling is improved and all key parameters are represented conservatively.

3.7 Staff Audit

To resolve questions raised during the staff's review of the Westinghouse TR, the staff conducted a technical audit at the Westinghouse Nuclear Center in Monroeville, Pennsylvania on March 17-18, 2003. Selected areas identified in the staff's review were examined in detail during the audit and Westinghouse provided a formal response (Reference 23) to the staff's questions. Topics covered in the audit included:

- SE restrictions on previously approved TRs
- Uncertainty analyses
- Benchmarking
- Key parameters
- Cross section modification during rod ejection
- Input controls
- Code coupling limitations

The audit was accomplished by a combination of interviews with the Westinghouse 3-D Rod Ejection Team members, reviews of the sample case calculation notes, and reviews of the individual code Users Manuals. Setup and execution of a special test case requested by the staff were also performed during the audit. This test case examined the effect of moderator temperature feedback by driving the moderator temperature coefficient to zero.

The staff also examined the derivation and solution in SPNOVA of the 3-D neutron kinetics equations and the effect of the pulse width variation. The coupling of the SPNOVA and VIPRE codes was reviewed, including neutronic and thermal-hydraulic channel mapping, file manipulation, time step synchronization, and limitations on representation of feedback mechanisms. The statistical approach used was clarified, and justification was provided for the use of the square root of the sum of the squares combination of independent variables.

The staff requests for clarification and additional information were satisfied during this audit, and it was confirmed that the methodology, analyses and results presented in the Westinghouse TR are consistent with RG 1.77 and the applicable SRP sections and that all key parameters can be modeled conservatively. The staff finds the analyses and results presented in WCAP-15806-P and WCAP-15807-NP to be acceptable, since the overall modeling is improved and all key parameters are represented conservatively.

4.0 CONCLUSION

The staff reviewed the application methodology, analyses, and results presented in WCAP-15806-P and WCAP-15807-NP and determined that the analyses and results are performed in accordance with the guidance and limitations provided in RG 1.77 and the applicable sections of the SRP. In addition, the new 3-D time dependent application methodology is a considerable improvement over the current methodology, leading to an more realistic calculation of the rod ejection event. The staff accepts this methodology and concludes that it is acceptable for referencing in licensing applications. The staff finds the analyses and results presented in

WCAP-15806-P to be acceptable, since the overall modeling is improved and all key parameters are represented conservatively.

The staff also noted both the proposals for new fuel enthalpy acceptance criteria and the Westinghouse stated interim criteria, but will not endorse any new criteria at this time. The staff approval of this TR covers only the justification for the use of the improved methodology to adequately and conservatively represent the rod ejection event.

The staff has concluded that although the current RG 1.77 limits may not be conservative for cladding failure at higher fuel burnups, the more realistic analyses performed by the NSSS vendors, which have been confirmed by NRC-sponsored calculations, provide reasonable assurance that the effects of postulated RIAs in operating plants with fuel burnups up to the currently approved 62 GWD/MTU will neither (1) result in damage to the reactor coolant pressure boundary nor, (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core as specified in current regulatory requirements (GDC 28).

On the basis of the above review and justification, the staff concludes that the proposed change to the Westinghouse control rod ejection accident analysis methodology is acceptable.

The basic 3-D rod ejection methodology, as defined in the Westinghouse submittal, is generally applicable to all Westinghouse and Combustion Engineering NSSS pressurized water reactors. Application of this methodology to non-Westinghouse NSSS plants would, however, require additional justification, including submittal of sample calculations, uncertainty analyses, and applicable benchmarking results.

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