Mr. W. J. Johnson, Manager
Nuclear Safety Department
Westinghouse Electric Corporation
Box 355
Pittsburgh, Pennsylvania 15230-0355

JAN 31 1989

Dear Mr. Johnson:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT, WCAP-9226-P/9227-NP, "REACTOR CORE RESPONSE TO EXCESSIVE SECONDARY STEAM RELEASES"

We have completed our review of the subject topical report submitted by the Westinghouse Electric Corporation. We find the report to be acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

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

In accordance with procedures established in NUREG-0390, it is requested that Westinghouse publish accepted versions of this report, proprietary and non-proprietary, within 3 months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating Accepted) following the report identification symbol.

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

Sincerely,

ORIGINAL SIGNED BY A. C. THADANI

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shok C. Thadani, Assistant Director
for Systems
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RD-8-W

Enclosure:
Safety Evaluation
WCAP-9226-P/9227-NP

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AT

ENCLOSURE

SAFETY EVALUATION OF THE WESTINGHOUSE ELECTRIC CORPORATION
TOPICAL REPORT WCAP-9226-P/9227-NP
"REACTOR CORE RESPONSE TO EXCESSIVE SECONDARY STEAM RELEASES"

1.0 INTRODUCTION

The Westinghouse topical report WCAP-9226-P/9227-NP, "Reactor Core Response to Excessive Secondary Steam Releases," dated January 1978, describes the Westinghouse methodology used for analysis of the core response to excessive secondary steam release events, such as breaks in high energy secondary steamlines, in Westinghouse designed PWRs. The report also provides an extensive study to demonstrate the sensitivity of the consequences of a steamline break accident, which is the worst case event in the excessive steam release category, to a variety of assumptions and defines the limiting case conditions that will be analyzed and included in an applicant's Safety Analysis Report.

An excessive secondary steam flow increases primary-to-secondary heat transfer and, therefore, causes overcooling conditions in the primary system. This overcooling causes positive reactivity insertion due to the negative moderator temperature coefficient and Doppler coefficient which, under adverse conditions, could cause a return to power leading to excessive heat generation with an accompanying departure from nucleate boiling (DNB) and violation of fuel integrity safety limit. The extent of the moderator reactivity feedback problem depends upon a number of parameters, including the number of coolant loops in the reactor system and the degree of mixing in the reactor vessel of the fluids from the affected and the intact loops. Lack of detailed modeling in the vessel could result in under-estimation of such reactivity feedback because of overprediction of the mixing, and hence in non-conservative results.

The purpose of this review is to evaluate the technical merit of the Westinghouse approach in the steamline break analysis and the results of sensitivity studies to determine the limiting steamline break accident. The review is not focused on the computer codes used in the analysis, nor is the review directed toward any application to a specific plant analysis.

The staff review of WCAP-9226-P/9227-NP was performed with technical assistance from the NRC technical consultants at Argonne National Laboratory (ANL) and International Technical Services, Inc.

2.0 STAFF EVALUATION

2.1 Overview of Analysis Methodology

The process of analyzing an excessive secondary steam release event uses a transient reactor system code, MARVEL (Ref. 1) or LOFTRAN (Ref. 2), to generate the primary system response to the secondary steam release. Both LOFTRAN and MARVEL utilize point kinetics models to describe the core nuclear power transients initiated due to cooldown following a steam release. Variations in reactivity due to moderator density feedback, Doppler feedback, boron injection and rod motion are simulated in the transient model. The reactivity coefficients used in LOFTRAN are obtained from the calculation of a three-dimensional static neutronic code, TURTLE (Ref. 3), assuming the end of life and stuck rod conditions. Since the system transient is performed using point kinetics, reactivity checks are made by incorporating spatial effects to verify that the LOFTRAN point kinetics model overpredicts the total change in reactivity, producing a more severe transient. From the LOFTRAN system response, statepoints are generated for the heat flux, reactor coolant system pressure, inlet temperature, core flow, boron concentration and reactivity at a given instant in the transient. These statepoints are then investigated using a subchannel thermal hydraulic code such as THINC IV (Ref. 4) to determine the minimum DNB ratio.

The MARVEL code was originally developed by Westinghouse for reactor system transient analysis of a 2-loop plant. The LOFTRAN code was developed for analysis of the 3- and 4-loop plants. Both MARVEL and LOFTRAN contain models for the phenomena important in the steamline break events, such as point kinetic models, steam generator heat transfer models, safety injection system models, various plant operating control systems, break flow model and the reactor vessel mixing models. The LOFTRAN code has been accepted by NRC for analysis of various transients and accidents including the steamline break accident.

THINC is a three-dimensional thermal hydraulic code which calculates the local coolant density, mass velocity, enthalpy, vapor void, static pressure and resulting DNBR distribution in a PWR core. THINC and its latest version THINC IV have been accepted by NRC for licensing calculations.

TURTLE is a three-dimensional static neutronics code which has been accepted by NRC for licensing calculation. However, it is indicated in the report that a modified version of the TURTLE code has been used in the analysis. Though the purpose of this review is to determine the acceptability of the analysis methodology for an excessive steam release event, we will require that, if the modified version of TURTLE is used in licensing calculation, it should be submitted for NRC staff review and approval.

2.2 Reactor Vessel Mixing

In using the LOFTRAN (or MARVEL) code for analysis of the system response to excessive secondary steam release events, the reactor vessel is nodalized as a split core with split bypass, downcomer, lower head and upper plenum regions. Only the upper head remains as one volume. The mixing between loops, which occurs in the inlet and outlet plena of the reactor vessel without any net flow between loops, is not simulated mechanically. Instead, provision is made to allow for mixing in the reactor vessel inlet and outlet plena through input mixing factors which allow the user to input any desired degree of mixing between loops. Such mixing affects the coolant thermodynamic properties, and therefore the moderator reactivity feedback.

Modeling the fluid mixing in the reactor vessel is an important task regarding excessive steam release analysis because of its effect on the reactivity feedback. Westinghouse took a two-pronged approach to modeling the reactivity feedback.

(1) To more accurately model the mixing between the affected and intact loop fluids, experimental data were taken at the Indian Point One-Seventh scale and other fluid mixing test facilities (Westinghouse proprietary information). These fluid mixing test produced flow contours indicating constant fractions of flow from the cold loop and therefore the fraction of cold loop flow for any region of the core. These flow maps serve as the basis for determining the mixing coefficients used in LOFTRAN (or MARVEL) for system thermal hydraulic calculation, and also provide a basis for the core fluid inlet temperature distribution in the TURTLE three-dimensional core physics calculation.

(2) The LOFTRAN system thermal hydraulics and point kinetics calculations are coupled with the 3-D TURTLE core physics computation via an iterative computational procedure. In the iterative process, the core power computed by the LOFTRAN (or MARVEL) code is checked at many points throughout the transient by running the 3-D core physics code which uses as its input the flows, fluid temperatures, etc. computed by LOFTRAN (or MARVEL) and adjusted by the temperature distribution map from the mixing test data. If the core powers calculated by the two codes differ, reactivity coefficients are adjusted in the LOFTRAN (or MARVEL) code's point kinetics routines to yield agreement with the stand-alone physics code. Iteration is carried out until the results of the two codes converge to a predetermined bound. To accelerate the convergence between the LOFTRAN and the physics codes, Westinghouse developed a method, based upon experience developed from use of the iterative procedure, of using the LOFTRAN point kinetics computations together with weighting factors which

place more relative importance on the cooler section of the core and therefore produces more severe reactivity feedback.

The reactor vessel mixing model is described in the "LOFTRAN Code Description" (Ref. 2), and the fluid mixing tests are described in a response to the staff question (Ref. 5). In addition, the NRC staff and its technical consultants had performed an audit (Ref. 6) of the proprietary fluid mixing data used by Westinghouse to support the mixing models and core fluid inlet temperature maps on which the determination of overall acceptability of the WCAP-9226 methodology depends. These proprietary experimental fluid mixing data was found to support the mixing and weighting factors used by Westinghouse in LOFTRAN and MARVEL. This confirms a previous NRC conclusion stated in its safety evaluation report for the LOFTRAN code that for steamline analysis, the input mixing coefficients based on the fraction of mixing measured in the Indian Point One-Seventh Reactor Vessel Model Test are conservative because they lead to a colder inlet temperature in the affected loop and thus to a more severe reactivity excursion. During the NRC audit, Westinghouse further assured that the fluid inlet temperature maps used by Westinghouse in the 3-D core physics computations reasonably represent the experimental data mentioned above, and that these temperature maps conservatively model the temperature of the fluid being convected into the region of the assumed stuck rod to be colder than would be indicated by the experimental data. Therefore, we conclude that the mixing of fluids from the affected and intact loops is reasonably represented, and that the computation of core power is conservative.

2.3 Modeling Conservatisms

In addition to the conservatisms introduced in the fluid mixing modeling between the affected and the intact loops, other sources of conservatism are present in the overall WCAP-9226 methodology through selection of input parameters and plant conditions.

Spatial reactivity coefficients were computed for a conservative scenario which represents the end of life (EOL) having the most negative moderator temperature coefficient, and the most reactive rod cluster control assembly stuck in its fully withdrawn position. These assumptions maximize the reactivity insertion throughout the steamline break accident and produce conservative results. Reactivity checks with the 3-dimensional physics code are also made to verify that the point kinetics model overpredicts the total change in reactivity for conservatism.

Other primary modeling conservatisms the Westinghouse SLB analysis include: (1) use of the Moody critical flow model for steam blowdown calculation resulting in higher steam blowdown rate to maximize the forcing function of the steamline break, and (2) use of a conservatively low effective tube bundle height in the steam generator to maximize the time the water level remains above the tube bundle, and maximize the heat transfer to the secondary side and the severity of the core response throughout the transient. Each of these models represents a conservative approach to its respective phenomenon.

2.4 <u>Sensitivity Studies</u>

The response of a PWR system to an excessive secondary release is dictated by the steam release rate. The consequence is also affected by the features of the reactor protection systems such as steamline isolation, feedwater isolation, core boration from safety injection and reactor trip. Chapter 3 of WCAP-9226 provides the sensitivity studies performed by Westinghouse to demonstrate that the main steamline break accident analyses in the applicants' safety analysis reports represent the bounding case for the excessive secondary steam release event. The sensitivity studies performed with LOFTRAN using a standard 2785 MWt 3-loop Westinghouse designed plant as the basic model. Westinghouse contended that the trends of these sensitivity studies performed for the three loop plants are applicable to the four loop plants. We agree that this is a reasonable conclusion.

Two cases were analyzed by Westinghouse: (1) the Base Case to show what the largest hypothetical steamline break transient might realistically look like at the worst time in fuel life, EOL, and (2) the Reference Case which is a standard Westinghouse SAR analysis for the 3-loop plants. In each case, the basic model incorporated a Series 51 steam generator which has a flow restrictor in the steam line downstream of the steam generator and can, therefore, have a double-ended steamline rupture area of 4.6 ft^2, whereas the more recent D series or the F series steam generators contain a 1.4 ft^2 flow restrictor as an integral part of the steam generator outlet nozzle, limiting the break flow to that amount. Thus this particular basic model would compute the worst break flow. Comparison of the Base Case with the Reference Case identified the extent of differences which SAR conservatism and additional calculational uncertainty factors render to the worst break at worst time in fuel life. The Reference Case was then used as the basis for the rest of the sensitivity studies.

Sensitivity studies were performed with respect to the single independent failure assumption (as in the availability of safeguards train), initial power operating modes, instrument errors, location of the pressurizer, feedwater assumptions, moderator density feedback, boron coefficient, shutdown margin, power feedback, RCS flow, moisture carryover/steam generator performance, upper head injection, initial steam generator water mass, plant feature variation (such as 3-loop plant versus 4-loop plant and steam generator types), N-1 loop operation, and steamline break sizes.

The results documented in Chapter 3 of WCAP-9226 indicate that those parameters to which the MDNBR was found to be sensitive are treated in a conservative fashion in the Reference Case.

2.5 ANL Audit Calculations

In 1983 and 1984, the staff technical consultants at the ANL performed audit calculations (Refs. 7 and 8) of certain steamline break computations contained in the North Anna and Marble Hills FSARs, which used the methodology described in WCAP-9226. These audit calculations were performed using RELAP5/MOD1.5 point kinetics and a set of weighting and mixing factors which were identical to those used by Westinghouse in a fully converged set of thermal-hydraulic computations.

Sensitivity and parametric studies were performed for various combinations of the following parameters: (1) 3- and 4-loop plants, (2) full and zero powers, (3) with and without offsite power, (4) with and without conservatism, (5) full and minimum safeguards, and (6) various break sizes. ANL concluded that the Westinghouse methodology contained many important conservatisms including the following:

1. Assumed 10 seconds of full main feedwater flow plus auxiliary feedwater (AFW) flow at the lowest expected temperatures, and then reduced to only the AFW flow which was diverted to the affected steam generator;

2. Used high upper head injection flowrate to allow continued depressurization;

3. Allowed only pure steam to exit the break and thus maximized the cooldown of the primary system;

4. Assumed minimum capacity for boric acid injection by allowing only the safety injection system pumps (not charging pumps) to deliver boron;

5. Lack of calculation of the primary side heat structures in LOFTRAN resulted in more rapid primary cooldown;

6. Used conservative reactivity feedback corresponding to the EOL, i.e., the largest moderator density coefficient, a low boron coefficient and low shutdown margin; and

7. Ignoring decay heat caused faster primary cooldown and therefore a greater return to power.

Results of the ANL audit computations indicate that Westinghouse results were conservative for typical Westinghouse 3-and four-loop plants.

2.6 Use of The W-3 Correlation

For the ANS Condition II events, such as an inadvertent valve opening, the design basis is to ensure that there will be at least a 95 percent probability at 95 percent confidence level that the hot rod will not experience departure from nucleate boiling. This is analyzed with a subchannel core thermal-hydraulic code (THINC IV) along with the critical heat flux (CHF) correlation, W-3. For an ANS Condition IV accident, such as main steamline break, fuel failure may occur, but the maximum amount of failure is limited by the radiological release set forth in 10 CFR 100. The fuel failure criteria for a steamline break is the DNBR limit of the CHF correlation which corresponds to the 95/95 limit.

The W-3 CHF correlation has previously been approved for application over a pressure range from 1000 to 2300 psia with a minimum DNBR limit of 1.3. However, in the steamline break accident, the primary system pressure may drop to as low as 500 psia which is outside the range of applicability of W-3. In respond to a staff question (Ref. 9), Westinghouse provided additional information to justify the use of the W-3 correlation in the lower pressure range. This information included a statistical analysis based on experimental data in the 500 to 1000 psia range, which adequately supports the conclusion that there is a 95 percent probability with 95 percent confidence level that DNB would not occur if the DNBR limit with the W-3 correlation is 1.45 in that pressure range. Thus we conclude that the W-3 correlation is an acceptable correlation for use in the SLB analysis with the appropriate DNBR limits of 1.45 for pressures from 500 to 1000 psia and 1.3 from 1000 to 2300 psia.

3.0 SUMMARY

The staff has reviewed the Westinghouse topical report WCAP-9226 regarding the methodology used for analysis of excessive secondary steam release events, and the sensitivity studies used to identify the limiting cases used in the SARs. We find that both the analysis methodology and the sensitivity studies are acceptable. This acceptability is subject to the following restrictions:

1. Only those codes which have been accepted by the NRC should be used for licensing application; and

2. For the pressure between 500 and 1000 psia, the 95/95 DNBR limit for the W-3 correlation is 1.45.

4.0 REFERENCES

1. Krise, R. C and S. Miranda, "MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System," WCAP-8843, November 1977.

2. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.

3. Barry, R. F. and S. Altomare, "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7213-A (P) and WCAP-7758-A (NP), February 1975.

4. Chelemer, H., et al., "THINC-IV - An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.

5. Letter from E. P. Rahe, Jr. (Westinghouse) to H. L. Thompson (USNRC), Response to Request for Additional Information on WCAP-9226 (P) and WCAP-9227 (NP), 'Reactor Core Response to Excessive Secondary Steam Release,'" April 21, 1986.

6. Letter from W. J. Johnson (Westinghouse) to M. W. Hodges (USNRC), "Review of Mixing Assumptions for Steamline Rupture WCAP-9226, Revision 1," NS-NRC-88-3312, February 29, 1988.

7. "RELAP5/MOD1.5 Analysis of a Main Steam Line Break - North Anna Power Station Unit 2," compiled and edited by G. B. Peeler, ANL/LWR/NRC 83-9, July 1983.

8. T. A. McDonald, "RELAP5/MOD1.5 Analysis of Steam Line Break Transients for a 4-Loop Westinghouse Plant," ANL/LWR/NRC 84-1, January 1984.

9. Letter from E. P. Rahe, Jr. (Westinghouse) to H. N. Berkow (USNRC), "Westinghouse Response to NRC Additional Request on WCAP-9226-P/WCAP-9227-NP," NS-NRC-86-3116, March 25, 1986.