October 14, 1999

Mr. Otto L. Maynard
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KA 66839

SUBJECT: WOLF CREEK GENERATING STATION - CORE DAMAGE ASSESSMENT GUIDANCE - (TAC NO. MA4190)

Dear Mr. Maynard:

In the letter of November 10, 1998 (ET 98-0079), you submitted a proposed revision to the core damage assessment guidance (CDAG) for Wolf Creek Generating Station (WCGS). The revised CDAG was based on WCAP-14696, "Westinghouse Owner's Group Core Damage Assessment Guidance," that had been submitted to the staff by the Westinghouse Owners Group (WOG). You identified WCGS as the lead plant for the WOG for review of WCAP-14696 and requested approval of the use of the WCAP at WCGS.

In the letter of September 2, 1999, to the WOG, the staff approved the use of the core assessment damage guidance in WCAP-14696, Revision 1, for Westinghouse nuclear power plants. Based on this approval, you may use WCAP-14696, Revision 1, at WCGS. Enclosed is a copy of the September 2, 1999, letter and associated safety evaluation.

This closes out the staff's review of TAC No. MA4190. If you have any questions, please contact me at 301-415-1307 or through the internet at jnd@nrc.gov.

Sincerely,

J N Donohew

Jack N. Donohew, Senior Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosure: September 2, 1999, Letter and Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 14, 1999

Mr. Otto L. Maynard
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Jack N. Donohew

Jack N. Donohew, Senior Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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Enclosure: September 2, 1999, Letter
and Safety Evaluation

cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 2, 1999

Mr. Lou Liberatori, Chairman
Westinghouse Owners Group Steering Committee
Indian Point Unit 2
Broadway & Bleakley Ave.
Buchanan, NY 10511

SUBJECT: SAFETY EVALUATION RELATED TO TOPICAL REPORT WCAP-14696, REVISION 1, "WESTINGHOUSE OWNERS GROUP CORE DAMAGE ASSESSMENT GUIDANCE" (TAC NO. M97447)

Dear Mr. Liberatori:

By letter dated November 22, 1996, the Westinghouse Owners Group (WOG) submitted Topical Report WCAP-14696, "Westinghouse Owners Group Core Damage Assessment Guidance," for NRC review. There is no proprietary version. In the topical report, a revised methodology was described that would be used by licensee emergency response organization staff for estimating the extent of core damage that may have occurred during an accident at a Westinghouse nuclear power plant. The revised methodology is a revised calculational technique for estimating core damage which relies on real-time plant indications rather than samples of plant fluids.

The revised post-accident core damage assessment methodology in WCAP-14696 replaces the methodology approved by the staff in 1984. The 1984 methodology was revised for two major reasons: (1) the current methodology relies on radionuclide samples and does not effectively support emergency response decisionmaking due to the significant time delay in obtaining and analyzing these samples using the post-accident sampling system (PASS), and (2) the methodology does not reflect the latest understanding of fission product behavior, particularly the sequence-specific nature of fission product retention and hydrogen holdup in the reactor coolant system (RCS), and fission product deposition in the containment and sample lines. Also, as part of a separate request related to PASS, the WOG has requested that the time commitment for obtaining and analyzing a radionuclide sample be eliminated, thereby rendering this information potentially unavailable for use in assessing core damage. The proposed PASS relaxations, discussed in WCAP-14986-P, will be the subject of a separate staff review and letter.

The NRC staff review of the revised guidance was initiated in early 1999 following submittal of a plant-specific application of the guidance by a lead plant, Wolf Creek Generating Station. The staff met with representatives of the WOG and the licensee on February 24, 1999, to discuss a number of comments and questions related to the revised guidance. The WOG provided formal responses to these items in a letter dated March 16, 1999. Based on further review, the staff issued a request for additional information (RAI) on March 25, 1999. The WOG provided responses to the RAI by letter dated April 28, 1999, and a subsequent revision (Revision 1) of the topical report.

The enclosed safety evaluation approves the use of the revised guidance in WCAP-14696, Revision 1, for core damage assessments for Westinghouse nuclear power plants. The staff

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concludes that the revised core damage assessment guideline (CDAG) in Appendix A to WCAP-14696 provides the capability to assess the degree of core damage with a sufficient level of accuracy and timeliness to support emergency response decisionmaking. The revised guideline represents an improvement over the existing methodology. It is both simpler and more timely, and accounts for fission product and hydrogen retention/holdup in the RCS and fission product removal by containment sprays in an approximate manner. By making core damage information available earlier in an event, such that it can be used to refine dose assessments and confirm or extend initial protective action recommendations, implementation of the revised CDAG should increase the effectiveness of the emergency response organization. Based on its review, the staff finds the revised CDAG provided in WCAP-14696, Revision 1, to be an acceptable basis for meeting the NUREG-0737, II.B.3 requirement for a core damage assessment procedure.

The NRC requests that the WOG publish an accepted version of the revised topical report within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed safety evaluation between the title page and the abstract, and add an -A (designating accepted) following the report identification number (i.e., WCAP-14696-A).

If the NRC's criteria or regulations change so that its conclusion in this letter, that the topical report is acceptable, is invalidated, WOG and/or the applicant referencing the topical report will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

Sincerely,

Stephen Dembek

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO WCAP-14696, "WESTINGHOUSE OWNERS GROUP CORE DAMAGE ASSESSMENT GUIDANCE," REVISION 1

WESTINGHOUSE OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION

By letter dated November 22, 1996, the Westinghouse Owners Group (WOG) submitted Topical Report WCAP-14696, "Westinghouse Owners Group Core Damage Assessment Guidance," for NRC review [Reference 1]. This report provides revised guidance for use by licensee emergency response organization staff in estimating the amount of core damage that might have occurred during an accident at a Westinghouse nuclear power plant. The requirement for such guidance was established by the Nuclear regulatory Commission (NRC) following the Three Mile Island (TMI) 2 accident. The revised guidance is intended to replace the post accident core damage assessment methodology (PACDAM) submitted by WOG and approved by the staff in 1984 [Reference 2].

The WOG proposed a revision of PACDAM for two major reasons: (1) the current methodology relies on radionuclide samples, and does not effectively support emergency response decision-making due to the significant time delay in obtaining and analyzing these samples using the post-accident sampling system (PASS), and (2) the methodology does not reflect the latest understanding of fission product behavior, particularly the sequence-specific nature of fission product retention and hydrogen holdup in the reactor coolant system (RCS), and fission product deposition in the containment, and sample lines. Also, as part of a separate request related to PASS, the WOG has requested that the time commitment for obtaining and analyzing a radionuclide sample be eliminated, thereby rendering this information potentially unavailable for use in assessing core damage. The proposed PASS relaxations, discussed in WCAP-14986-P, are the subject of a separate staff review.

The NRC staff review of the revised PACDAM guidance was initiated in early 1999 following the submittal of a plant-specific application of the guidance by a lead plant, Wolf Creek Generating Station. The staff met with representatives of the WOG and the licensee on February 24, 1999, to discuss a number of comments and questions related to the revised guidance. The WOG provided formal responses to these items in a letter dated March 16, 1999 [Reference 3]. Based on further review, the staff issued a request for additional information (RAI) on March 25, 1999. The WOG provided responses to the RAI by letter dated April 28, 1999 [Reference 4], and submitted a subsequent revision of the topical report (WCAP-14696, Revision 1). This safety evaluation addresses the acceptability of the revised guidance in WCAP-14696, Revision 1 (WCAP-14696) for core damage assessment at a Westinghouse nuclear power plant.

ENCLOSURE

2.0 DISCUSSION

As part of the TMI Action Plan Requirements (NUREG-0737), NRC required licensees to provide a procedure for relating radionuclide concentrations to a realistic estimate of core damage. The primary interest was in being able to differentiate between four major fuel conditions; no damage, cladding failures, fuel overheating, and core melt. The requirement for a core damage assessment procedure was considered to be an element of NUREG-0737 Item II.B.3, Criterion 2 (a), which states:

> *The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the 3-hour time frame established above, quantification of the following:*
>
> *(a) Certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes)*

The current "WOG Post Accident Core Damage Assessment Methodology" (PACDAM) was approved in 1984 based on satisfying this criterion [Reference 5].

The PACDAM relies primarily on radionuclide analysis to establish the extent of core damage. The core damage estimates derived from radionuclide information are confirmed using auxiliary indicators, including containment hydrogen concentration, core exit thermocouples (CETs), reactor vessel level instrumentation system (RVLIS), and containment radiation monitors. Because of the significant delays in obtaining and analyzing radionuclide samples, estimates of core damage based on PACDAM are not generally available to support emergency response decisionmaking, until about three hours after the decision to take these samples.

In WCAP-14696, the WOG has proposed a revised core damage assessment methodology that would provide core damage estimates earlier than the present methodology. The report includes a characterization of how PASS is currently used in Westinghouse plants, a detailed discussion of core damage accident characteristics and the use of instrument indications to diagnose and evaluate core damage, and a revised methodology for assessing core damage. The actual "core damage assessment guideline" (CDAG) is provided in Appendix A of the topical report. The CDAG setpoints and the background for the guideline are provided in Appendices B and C, respectively. As part of plant-specific implementation, licensees would develop a plant-specific version of the CDAG based on the information contained in Appendices A through C.

The CDAG relies on fixed plant instrumentation. The extent of core damage is estimated from containment radiation monitors and CETs. The core damage estimates are compared to indications from hydrogen monitors, RVLIS, hot leg resistance temperature detectors (RTDs) and the source range monitor (SRM) to confirm that the core status and extent of damage suggested by the containment radiation monitors and CETs are consistent with these other indicators. The CDAG considers fission product and hydrogen retention/holdup in the RCS, and the impact of containment sprays on the airborne fission product inventory. The reliance on radionuclide samples from PASS, which is the primary means of assessing core damage in

PACDAM, is eliminated in the CDAG. By relying on fixed plant instrumentation and eliminating the depenoence on PASS, the time required to develop a core damage estimate would be significantly less than with the current methodology (within minutes using the CDAG, versus about 3 hours using PACDAM/PASS). Earlier availability of core damage information in an event would enable licensees to use this information to supplement their current emergency response decisionmaking process, thereby improving the capability to protect the public health and safety. A summary description of the core damage assessment process is provided in the next section.

3.0 CORE DAMAGE ASSESSMENT PROCESS

In the CDAG provided in Appendix A to WCAP-14696, the status of the core is initially classified based on CET and containment radiation monitor indications, together with RCS pressure and containment spray system status. The status of the core is assigned to one of three categories -- no core damage, possible clad damage, possible fuel over-temperature damage. If clad or fuel over-temperature damage is indicated, the user is directed to corresponding steps (Step A or Step B of the CDAG, respectively) where more detailed assessments and comparisons are performed.

If clad damage is indicated, the percent of core damage is estimated separately based on containment radiation and CET indications. These estimates are compared with the expected response from the containment hydrogen monitor, RVLIS, hot leg RTD, and source range monitor, and compared with each other. If the expected response is not obtained (i.e., a difference in core damage estimates from containment radiation and CETs of less than 50 percent of the estimate using the containment radiation monitor), possible causes for the deviation are considered.

If fuel over-temperature damage is suspected, the process is similar except a more detailed evaluation of hydrogen concentration information is performed for over-temperature damage. The percent of core damage is estimated separately based on containment radiation and CET indications. These estimates are compared with the expected response from RVLIS, hot leg RTD, and source range monitor, and compared with each other (with the expectation that the difference in core damage estimates from containment radiation and CETs would be less than 50 percent of the estimate using the containment radiation monitor). An estimate of core damage is also obtained based on the containment hydrogen concentration, with consideration of RCS pressure and vessel reflood. That estimate is not expected to deviate from the estimates developed from the containment radiation and CET indications by more than 25 percent core damage. If the expected response is not obtained, possible causes for the deviation are considered.

Plant-specific curves are used to establish the containment radiation level that would be detected at the containment radiation monitors as a function of time for 100 percent clad damage and for 100 percent fuel over-temperature damage. Separate curves are developed for events with high and low RCS pressure and for events with and without sprays operating. In developing the plant-specific curves, adjustments are made to account for the effects of fission product retention in the RCS and fission product removal by spray on the containment radiation levels. The extent of clad or fuel over-temperature damage is determined by comparing the containment radiation monitor reading with the expected value from the corresponding curve.

In parallel, the extent of clad or fuel over-temperature damage is determined as the fraction of CETs that exceed the temperature setpoint value associated with either clad damage or fuel over-temperature damage. Temperature differentials between CETs and clad are accounted for in establishing setpoints. In cases where core damage is limited to clad damage, a different CET setpoint value is used to denote clad damage depending on whether the RCS is at high or low pressure.

Auxiliary indicators are used to confirm the initial classification of core status, and the extent of core damage as indicated by CETs and containment radiation monitors. For core damage that is limited to clad damage the expected response of the auxiliary indicators is: no significant hydrogen detected, reactor vessel water level in a range where only limited clad heatup may occur, hot leg RTDs less than a value corresponding to extensive clad heatup, and source range monitor count rate corresponding to core uncovery. For core damage that involves fuel over-temperature damage the expected response of the auxiliary indicators is: reactor vessel water level below an elevation corresponding to extensive clad heatup, hot leg RTDs greater than a value associated with extensive clad heatup, source range monitor count rate corresponding to core uncovery, and significant hydrogen detected.

4.0 EVALUATION

The scope of the staff's review included: (1) the rationale for revising the core damage assessment methodology, and the acceptability of eliminating dependence on PASS, (2) the appropriateness of control room indications used to assess core damage, (3) the approach for relating instrumentation readings to core damage estimates, and (4) the consistency of fission product/hydrogen assumptions and setpoint values with the current understanding of severe accident progression and fission product behavior. The staff's assessment is provided below.

The current core damage assessment methodology does not account for fission product and hydrogen retention/holdup in the RCS, and fission product deposition in the containment, and sample lines. Information provided in support of the revised guidance indicates that these mechanisms can be significant in certain sequences. Because they are neither precluded by PASS design nor accounted for in the current methodology, these mechanisms can bias the radionuclide and hydrogen concentration samples obtained from PASS, and lead to under-estimates of the extent of core damage. The revised methodology includes an explicit, albeit approximate, accounting of these mechanisms. In addition, by eliminating the current reliance of the core damage estimate on obtaining and analyzing a radionuclide sample, the timeliness of the core damage estimate is substantially improved. On these bases, the staff considers modifications to the current methodology to be warranted.

The CDAG relies on the indications from the safety-related containment radiation monitors and CETs, together with the RCS pressure and containment spray system status, to arrive at an initial estimate of core damage. In subsequent steps these indications are further evaluated to arrive at an estimate of core damage, and several additional indicators are used to independently confirm the core damage estimate, including the safety-related containment hydrogen monitor, RVLIS, RTDs, and SRM. The staff concludes that the use of fixed plant instruments in the manner described in the CDAG provides an acceptable alternative to radiochemistry analysis of a radionuclide sample to obtain an approximate estimate of the extent of core damage during the transient phase of an accident. The array of instrument parameters on which the CDAG is based provide the most direct indication of the onset and extent of core damage. These instruments include the key indicators that would be used by NRC to evaluate the extent of core damage during an operational event (Response Technical Manual , NUREG/BR-0150), as well as several additional indicators such as RTD and SRM readings to confirm core status. There is reasonable assurance that these instruments, specifically, the CETs, containment radiation monitors, and hydrogen monitors, will be available, since they are identified as Category 1 instruments in Regulatory Guide 1.97, and as such, are environmentally qualified, redundant, and powered from batteries in the event of a loss of AC power[1]. The process and priorities established in the CDAG for using these instruments (i.e., the primary reliance on containment radiation monitors and CETs, with confirmation from other less direct indicators) is commensurate with the value of the various instruments in estimating core damage.

The approach for converting instrument readings into core damage estimates is consistent with the current understanding of clad and fuel damage characteristics, and accounts for fission product and hydrogen retention/holdup in an approximate fashion. Specifically, containment radiation monitor readings are compared to plant-specific radiation levels for 100 percent clad damage or fuel over-temperature damage, CET readings are compared to values typically associated with clad damage and fuel over-temperature damage, and containment hydrogen concentration is compared to amount expected in containment for 100 percent over-temperature damage. (CET readings that exceed the setpoints or the operating limits of the thermocouples are interpreted as core damage in that region of the core.) The CDAG includes a step where the core damage estimates derived separately from different indicators (containment radiation, CET, and containment hydrogen concentration readings) are compared and reconciled, thereby improving the confidence in the core damage estimate.

The CDAG provides recommended assumptions regarding fission product release from the fuel and retention of fission products within the RCS. These assumptions would be used by licensees in developing plant-specific curves relating containment radiation readings to core damage. For events limited to clad damage (Step A of the guideline), the CDAG assumes the NUREG-1465 gap release source term is released to the RCS. For events involving fuel over-temperature damage (Step B of the guideline), the CDAG assumes the NUREG-1465 source term for gap release plus early in-vessel release is released to the RCS. The in-vessel releases are adjusted to account for retention of fission products within the RCS. For events limited to clad damage, the CDAG assumes 50 percent of noble gases and 2 percent of all other fission products released to the RCS are subsequently released to the containment in high pressure sequences, and assumes 100 percent of noble gases and 50 percent of all other fission products released to the RCS are subsequently released to containment in low pressure sequences. For events involving fuel over-temperature damage, the CDAG assumes 50 percent of noble gases and 5 percent of all other fission products released to the RCS are subsequently released to the containment in high pressure sequences, and assumes no further reduction in low pressure sequences.

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[1]There are about 50 CETs in a typical Westinghouse reactor, with each thermocouple representing about 2 percent of the core. There are a minimum of two containment area radiation - high range monitors, installed at widely separated locations. Continuous indication of containment hydrogen concentration is provided by redundant, safety-related, hydrogen monitors.

Additional credit for fission product retention in high pressure sequences is appropriate since the NUREG-1465 source terms were chosen to be representative of sequences with low pressure in the RCS at the time of core degradation. High pressure sequences allow for longer residence time of aerosols released from the core, increased retention of aerosols within the RCS, and lower releases into containment than provided for in the NUREG-1465 source terms. A substantial quantity of fission products (as well as hydrogen) is also held-up within the RCS as compressed vapor. Credit for fission product retention is reasonable for estimating clad damage in low pressure sequences since no credit for RCS holdup is assumed in the NUREG-1465 gap release source term. In contrast, the NUREG-1465 early in-vessel release source term accounts for in-vessel fission product retention. Thus, no further reduction in fission product releases are assumed in the CDAG in estimating fuel over-temperature damage in low pressure sequences.

The CDAG also assumes all airborne fission products except noble gases are reduced by a factor of 100 for sequences with sprays operating. No reduction of containment fission product inventory is assumed when sprays are not operating. Existing studies (e.g., "A Simplified Model of Aerosol Removal by Containment Sprays," NUREG/CR-5966) show that typical pressurized water reactor containment spray systems are capable of rapidly reducing the concentration of airborne activity by about two orders of magnitude within about 30 minutes. Thus, the staff considers this assumption reasonable.

In response to a staff request regarding the bases for the recommended fission product assumptions, WOG provided results from modular accident analysis program (MAAP) calculations for several high pressure and low pressure core melt scenarios, with and without sprays. This information shows the fraction of the fission products predicted to reside in the core, RCS, and containment at various times, including the fraction of fission products that are airborne and deposited within the containment. The staff considered this information, as well as the results from source term code package calculations, the NUREG-1150 expert elicitation process for source term issues, and more recent MELCOR calculations in judging the reasonableness of the holdup/retention assumptions. It should be noted that fission product releases and distributions vary considerably from sequence to sequence, since release and deposition mechanisms are dependent on sequence specifics, such as core heatup rate, temperature distributions within the core and RCS, and operation of engineered safety features. Even for the same sequence, significant differences in predicted fission product behavior are not uncommon due to differences in models and assumptions. Although better understood than at the time of the TMI accident, fission product behavior is complex and the uncertainties in fission product calculations remain large. As a result, the airborne fission product inventory "seen" by the containment radiation monitors can, realistically, only be related to a level of core damage in an approximate sense. Based on the staff's review, the recommended assumptions in the WCAP provide a reasonable characterization of fission product retention in the RCS for high and low pressure sequences, within the uncertainties inherent in these parameters. While the selected values do not bound all conceivable scenarios, they would generally lead to conservative estimates (i.e., over-estimates) of core damage.

The approach taken in the CDAG for translating hydrogen concentration readings into core damage involves comparison of the measured containment hydrogen concentration with the maximum concentration of hydrogen expected for an in-vessel core damage event. The concentration of hydrogen that is expected in containment is established for four different categories based on a characterization of results from MAAP calculations. These categories reflect whether the RCS is at high or low pressure (which impacts the quantity of hydrogen produced as well as retained in the RCS), and whether water has been injected to the core (which impacts the quantity of hydrogen produced). The hydrogen setpoint values are established for ice condenser containments as well as large dry containments, with the note that containment hydrogen concentration would only be a reliable measure of fuel over-temperature for accident sequences in which the hydrogen igniters were not in operation.

Consideration of water injection and RCS pressure as distinguishing characteristics impacting containment hydrogen concentration is appropriate since these two factors are the single-most important factors impacting hydrogen production and release. The staff considered the assumptions recommended in the CDAG for establishing the plant-specific hydrogen concentration setpoint values for each of these categories. Similar to the situation regarding fission product behavior, hydrogen production and release is sequence specific, and predicted hydrogen concentrations can differ significantly from code to code. Nevertheless, for purposes of relating the measured hydrogen concentration to an approximate level of core damage, the assumptions provided in the CDAG provide a reasonable characterization of the extent of hydrogen produced and retained in the RCS for each of the four categories considered.

As part of the review the staff identified three areas requiring further justification. These include: (1) omission of PASS radionuclide samples as a means of confirming core damage estimates, (2) consistency of clad damage criteria with best estimate burst strain correlations, and (3) the effect of additional fission product removal mechanisms beyond those considered in CDAG, specifically, deposition in containment by natural processes. The resolution of these issues is summarized below:

The basic question regarding the omission of PASS concerns whether certain elements of the current methodology should be retained such that a radionuclide sample could be used to confirm the core damage estimate derived from fixed plant instrumentation. The staff concludes that the methodology can serve its intended purpose of classifying the extent of core damage without relying on PASS. Accordingly, PASS need not be incorporated as an element of CDAG.

For purposes of estimating whether clad damage has occurred, the CDAG originally recommended a CET value of 1600°F if RCS pressure exceeds 1600 psig and a CET of 1200°F if RCS pressure is below 1600 psig. The staff questioned the consistency of these temperature/pressure values with best-estimate burst strain correlations. In response, the WOG provided additional information regarding best-estimate versus conservative calculations, and results of further evaluations assuming higher burnup fuel. Based on further evaluation, the WOG reduced the failure criteria (for the high RCS pressure) from 1600°F to 1400°F to account for lower failure temperatures with higher fuel burnup. The staff concludes that the revised pressure/temperature setpoint values provide a reasonable measure for assessing the extent of clad damage.

Finally, the CDAG considers fission product retention in the RCS and removal by containment sprays, but does not specifically account for fission product deposition in containment by natural processes. In response to a staff request, the WOG provided the results of additional MAAP calculations showing the fraction of fission products in containment that are airborne and deposited. Based on a review of this information as well as NRC-sponsored work related to aerosol removal (NUREG/CR-6189), the staff concludes that the impact of natural processes is within the margin of error of the CDAG (i.e., within the sequence to sequence variability in releases from RCS), and need not be explicitly considered.

In reviewing WCAP-14696, the staff also considered the guidance contained in a "Post-Accident Sampling Guide for Preparation of Procedure to Estimate Core Damage" [Reference 6]. This document was developed by the staff and forwarded to licensees in the 1982-1983 time-frame. The document (1) discussed factors that should be considered in estimating core damage, and (2) suggested categories of fuel damage, an example of a process for estimating core damage from radionuclide information, sample locations and their role, and use of auxiliary indications to confirm core damage estimates derived from radionuclide measurements.

The Sampling Guide suggested four broad categories of core damage (no damage, cladding failure, fuel overheating, and core melt), with the degrees of core damage in each category further delineated. The CDAG enables the degree of core damage to be classified into the first three of these categories, but does not attempt to distinguish between extensive fuel overheating and core melt. Recognizing the large uncertainties in accident progression as well as the role of the core damage assessment methodology in emergency response decisionmaking, such refinement is not necessary to support the decisionmaking process, and implies a greater level of knowledge than is justified. As illustrated by the TMI experience, the extent of core damage and melting was not fully understood until years after the accident. Finally, with the shift from reliance on PASS to reliance on fixed plant instrumentation, the specific provisions set forth in the Sampling Guide regarding sampling from the RCS, containment sump, and containment atmosphere are no longer relevant to the revised methodology.

5.0 CONCLUSION

The staff concludes that the revised CDAG provides the capability to assess the degree of core damage with a sufficient level of accuracy and timeliness to support emergency response decisionmaking at a Westinghouse nuclear power plant. The revised guideline represents an improvement over the existing methodology. It is both simpler and more timely, and accounts for fission product and hydrogen retention/holdup in the RCS and fission product removal by containment sprays in an approximate manner. By making core damage information available earlier in an event, such that it can be used to refine dose assessments and confirm or extend initial protective action recommendations, implementation of the revised CDAG should increase the effectiveness of the emergency response organization. Based on its review as discussed above, the staff finds the revised CDAG provided in WCAP-14696, Revision 1, to be an acceptable basis for meeting the NUREG-0737, II.B.3 requirement for a core damage assessment procedure.

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Date: September 2, 1999

6.0 REFERENCES

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