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

March 16, 1999

MEMORANDUM TO: Francis Akstulewicz, Acting Chief
Generic Issues and Environmental Projects Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

FROM: Peter C. Wen, Project Manager *Peter C. Wen*
Generic Issues and Environmental Projects Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

SUBJECT: SUMMARY OF FEBRUARY 24, 1999, MEETING WITH
WESTINGHOUSE OWNERS GROUP (WOG) REGARDING
WCAP-14696, WOG CORE DAMAGE ASSESSMENT GUIDANCE

On February 24, 1999, a public meeting was held at the U.S. Nuclear Regulatory Commission's (NRC's) offices in Rockville, Maryland, between members of the WOG, Westinghouse, Wolf Creek Nuclear Operating Corporation (WCNOC), and NRC staff. Attachment 1 lists attendees at the meeting and Attachment 2 contains a copy of the material presented at the meeting.

The purpose of the meeting was to discuss the NRC staff's comments on the WOG Core Damage Assessment Guidance (Topical Report WCAP-14696, "Westinghouse Owners Group Core Damage Assessment Guidance,"), which is currently under staff review. This document, if approved by the staff, would replace the post-accident core damage assessment methodology that is currently in place at Westinghouse plants. To facilitate the discussion, a list of questions/comments was faxed to WOG and placed in the NRC public document room (Accession Number 9902190369) before the meeting.

During the meeting, WCNOC representatives presented an overview of the present role of core damage assessment in Wolf Creek's emergency plan and emergency response decision-making process. Their current methodology is based on the WOG post-accident core damage assessment guidance (CDAG) approved in 1984 and relies on results from the post-accident sample system (PASS). Because of significant delays in obtaining PASS results, current core damage assessments are not timely and therefore are not used as input to emergency action level (EAL) or protective action recommendation (PAR) decision-making. The WCNOC representative indicated that the revised CDAG, which relies only on fixed plant instrumentation, would offer results in a more timely fashion, and therefore could be integrated into the emergency plan decision-making process.

The WOG representative outlined the rationale and development philosophy for the revised CDAG, including consideration of the timeliness, accuracy, and availability of core damage assessment information. Considerable discussion centered on the information provided by the fixed plant instrumentation on which the CDAG relies during core damage events, and how this information is used in the CDAG.

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F. Akstulewicz

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March 16, 1999

On March 16, 1999, the staff received the response from the WOG (see Attachment 3,) documented the responses to all NRC comments/questions. The NRC staff is currently reviewing the WOG's response.

Attachments: As stated

cc w/atts: See next page

March 16, 1999

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cc w/atts: See next page

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NRC/WOG MEETING ON WOG CORE DAMAGE ASSESSMENT GUIDANCE
LIST OF ATTENDEES
February 24, 1999

<u>NAME</u>	<u>ORGANIZATION</u>
Bob Palla	NRR/DSSA/SCSB
Jim O'Brien	NRR/DIPM/EP&RPB
George Thomas	NRR/DSSA/SRXB
Mike Waterman	NRR/DE/EICB
Rick Hasselberg	NRC OPS CTR
Paul Boehnert	NRC ACRS
Peter Wen	NRR/DRIP/RGEB
Ken Vavrek	WOG-Project
Bob Lutz	Westinghouse
Dennis Boyd	ANO
Kurt Cozens	NEI
Bob Bryan	TVA
Ken Thrall	WCNOC
Dale Lemnons	WCNOC
David Claridge	WCNOC
Terry Carrett	WCNOC
William Kitchum	WCNOC
Ray Schneider	ABB/CEOG

Westinghouse Owners Group Core Damage Assessment (CDA) Guideline

Bob Lutz
Westinghouse / Westinghouse Owners Group

February 24, 1999



BACKGROUND

- Core Damage Assessment -
 - 10CFR50.47(b)(9)
 - "Adequate methods, systems and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use"
 - NUREG-0737 (II.B.3)
 - "... the capability to promptly quantify certain radionuclides that are indicators of core damage"



WOG Core Damage Assessment Methods

- Existing WOG Core Damage Assessment
 - Based on 1980 state-of-knowledge of severe accidents
 - Relies primarily on the results of analysis of radionuclide samples of plant fluids
- The revised CDA (WCAP-14696)
 - Based on 1996 state-of-knowledge of severe accidents
 - Relies entirely on fixed in-plant instrumentation



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Rationale for Change

- Revised WOG CDA was developed to recognize
 - Changes in knowledge of severe accidents
 - Progression
 - Indications
 - Fission Product Behavior
 - Current Emergency response decision making processes used by licensees
 - EALs
 - Offsite Dose Projections
 - Offsite Protective Actions



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ALTERNATIVES FOR OBTAINING CDA INFORMATION

- There are two fundamental methods to obtain information for making core damage assessments
 - Sampling (Post-Accident Sampling System)
 - Fixed In-Plant Instrumentation
- These fundamental methods may be supplemented by
 - Calculational Methods
 - Portable Instrumentation



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Measures of Effectiveness

- Revised WOG CDA is based on optimizing three measures of effectiveness of CDA
 - Timeliness of information
 - How representative is the information relative to current plant conditions?
 - Accuracy of information
 - How representative is the information relative to the actual plant conditions?
 - How representative is the information relative to that predicted for an accident?
 - Availability of information
 - Can the information be obtained during core damage accidents?



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TIMELINESS OF CDA INFORMATION

- There are two broad issues related to timeliness:
 - Time intervals for information
 - Delay time between request for sample and reporting results vs. instrument response times for fixed instrumentation
 - Manpower required to obtain information
 - Resources may be stretched during initial parts of an accident
- Timeliness issues are related to the recognition that during transient portions of a core damage accident core conditions change minute by minute



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ACCURACY OF CDA INFORMATION

- There are four broad issues related to accuracy:
 - Holdup of radionuclides and hydrogen in the RCS
 - Information from containment measurements may under predict core damage
 - Radionuclide removal processes in the containment
 - Information from containment measurements may under predict core damage
 - Plateout of chemical species
 - Sample information may under predict core damage; instrumentation may over predict core damage
 - Location (RCS and containment) of measurement / sample vs. actual conditions



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AVAILABILITY OF CDA INFORMATION

- There are two broad issues related to availability of information for CDA estimates
 - Impact of system failures during the accident
 - Ability to sample can be impacted by system availability
 - Ability to sample and analyze can be impacted by a.c. and d.c. power availability
 - Ability of instrumentation can be impacted by d.c. power availability
 - Impact of plant conditions during the accident
 - Ability to sample can be limited by pressures and fluid levels
 - Ability of instrumentation can be limited by qualification levels



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Development Philosophy

- For core damage estimates to be effective, they must be timely, accurate and available
- Core Damage Assessment should be as realistic as possible
 - Where uncertainties exist, the assessment should generally over predict the amount of core damage
- To meet these objectives, we must rely on fixed in-plant instrumentation



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Available Instrumentation

- Instrumentation that provides some information into the amount of core damage
 - Core Exit Thermocouples - indication of core overheating
 - Containment Radiation - indication of loss of fuel cladding and RCS barriers
 - Containment Hydrogen - indication of severe core overheating
 - Reactor Vessel Level - indication of inadequate core cooling
 - Neutron Monitors - indication of inadequate core cooling
 - Loop RTD - indication of core overheating

Underlined Instrumentation was chosen for numerical estimate



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Core Exit Thermocouples (CETs)

- Measures temperature of steam exiting a fuel assembly
- Approx. 50 fuel assemblies, uniformly distributed through the core, are fitted with CETs
- Limited useful measurement range to ~2000°F
- Indicated temperature lags clad / fuel temperature by 200°F to 600°F
- Errors can be diagnosed by comparison to adjacent CET indications



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Fuel Rod Clad Failure

- Fuel rod clad failures can occur at 1400°F for large LOCA based on clad strain correlations
- At higher RCS pressures, higher clad temperatures are required for equivalent strain
 - at 2200 psig, a clad temperature of 2000°F is required
- An RCS pressure of 1050 psig was chosen to represent high and low RCS pressure
- RCS Pressure > 1050 psig -- 1600°F CET
- RCS Pressure < 1050 psig -- 1200°F CET



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Fuel Rod Overheating

- Most of the noble gas and volatile (iodine and cesium) is released from the fuel pellet at temperatures just above 2400°F
- Thus we are interested in the amount of fuel that has exceeded about 2400°F
- At these temperatures, the cladding temperature is higher than the pellet temperature due to Zr-water reactions
- Thus, significant f.p. releases can be inferred from CET indications near 2000°F



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Containment Radiation Levels

- Measures gross activity in the containment
 - noble gases, iodines, cesiums and other volatiles
- Containment radioactivity levels are affected by
 - the amount released from the core / RCS
 - normal coolant activity
 - clad failures (small fraction of core inventory)
 - fuel overtemperature
 - the amount retained in the RCS
 - the amount washed-out by containment spray



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Impact of Fuel Damage on Containment Radiation

- No fuel damage can be indicated by low containment radiation levels
- Fuel rod clad damage and core overheating can be indicated by higher containment radiation levels
 - Approx 60% clad failure occurs before overheating
 - 1% Clad damage can be differentiated from normal coolant activity
 - Between about 50% and 100% clad damage, fuel overheating cannot be positively identified
- SGTR and ISLOCA may not result in release to containment



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Impact of LOCAs vs. Non-LOCAs on Containment Radiation

- For LOCAs, the retention in the RCS is
 - 50% of volatiles and 0% of the noble gases
- For non-LOCAs, the retention in the RCS is
 - < 98% of the volatiles and < 50% of the noble gases
- An RCS pressure 1600 psig was used to discriminate LOCAs and non-LOCAs
- Opening of the RCS (hot leg creep failure, PORVs, etc) results in LOCA-type releases to containment



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Impact of Containment Spray on Containment Radiation

- Containment sprays can quickly reduce volatile radionuclide concentrations by as much as a factor of 100; noble gases are unaffected by spray
- Natural processes also reduce volatile and noble gas radionuclide concentrations, but much more slowly than sprays



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Containment Hydrogen Monitor

- Measures hydrogen concentration in the containment
- Containment hydrogen levels are affected by
 - the amount hydrogen generated in the core during boil-down
 - additional hydrogen generated during recovery
 - the amount of hydrogen retained in the RCS



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Impact of LOCAs vs. Non-LOCAs on Containment Hydrogen

- For large and medium LOCAs, the hydrogen generation is limited by the rate of water boiloff
 - generally on the order of 25% zirc-water reaction
- For small LOCAs and non-LOCAs, the lower boil-off rate results in higher hydrogen generation
 - generally on the order of 50% zirc-water reaction
- An RCS pressure of 1050 μ sig was used to distinguish between the two classes of accidents



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Other Impacts on Containment Hydrogen

- Recovery (addition of water to the core after the initial boil-down) can produce as much as an additional 25% zirc-water reaction
- As much as 50% of the hydrogen can be held-up in the RCS for non-LOCA accidents
 - For ease of calculation, the same 1050 psig was used to account for RCS holdup



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Summary

- Primary assessment of clad damage and fuel overheating is made using
 - Core exit thermocouple indications
 - and
 - Containment radiation levels
- For fuel overheating episodes, containment hydrogen is also used to validate the estimate
- Reactor vessel level, source range neutron monitors and hot leg RTDs are used as secondary indicators of core damage



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Summary (Continued)

- Compared to the existing CDA methodology, this revision is more accurate and timely
- Methodology is estimated to be accurate to plus or minus 50% for the amount of damage
- This level of accuracy is considered adequate for input to EP decisions
 - For example, the recommendation would not change for a 25% or a 50% fuel overtemperature estimate



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Conclusion

- This methodology is more reliable than the present methodology which relies on samples of radioactive fluids from plant systems
- We believe that the effectiveness of the emergency plan is improved using the revised CDA



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Post-Script

- The temptation to critique and over-analyze the values used as setpoints in this methodology should be avoided in light of the accuracy required from the methodology and its end use.



CDAM, PASS and E-Plan Discussion



**Wolf Creek Nuclear Operating
Corporation**

February 24, 1999

CDAM Summary Overview

- WCAP-14696 submitted 11/22/96 for proposed revision of CDAM
- WCNOC submittal made 11/10/98 for plant specific reduction in PASS based on WCAP-14696
- Proposed revision of CDAM utilizes fixed in-plant instrumentation (e.g. CET, CHARM, Cont. H₂) instead of samples to determine results



PASS Summary Overview

- WCAP-14986 submitted 10/26/98 for proposed reduction of PASS requirements
- WCNOG submittal made 11/10/98 for plant specific reduction in PASS based on WCAP-14986
- NRC review requested by 5/15/99 to allow appropriate design and installation efforts to be completed to resolve Y2K issues



PASS Summary Overview

- Wolf Creek proposes elimination of regulatory requirements for PASS samples, with the exception of the Boron analysis which will have an 8 hour time limit for sample and analysis
 - Boron sample would be taken by grab sample and analyzed
 - Grab sample capability would be retained for any other analysis desired
- Wolf Creek currently complies with NUREG-0737 and Reg. Guide 1.97 w/ the exception of RCS dissolved H₂ and dissolved gases. Exception taken 9/98 w/NRC approval.



Key Objectives of Decision Making

- Protection of Public Health and Safety
- Timely
- Appropriate



Key Elements of Decision Making

- EALs
- CDA
- PARs



Emergency Action Levels - EALs

- First phase of diagnosis
- Primarily event driven
- Numerous plant parameters monitored for actions
 - EAL Triggers
 - EOP Actions and / or transitions



Core Damage Assessment - CDA

- Performed in Technical Support Center
- There is no Emergency Plan Procedure use of a detailed Core Damage Assessment
- Inadequate core cooling indicators that are used are
 - Core temperatures
 - High / alarming radiation monitors
 - RCS inventory
 - Injection flows
 - High activities (coolant / containment atmospheric)



Protective Action Recommendations - PARs

- WCNOC

- WCNOC develops PARs based primarily on plant condition and trends (event based decision)
- Dose projections may be considered when issuing PARs



Protective Action Implementation - KDHE

- Kansas Department of Health & Environment Responsibility

- Performs independent offsite dose assessment
- Monitors discussion / decision process of PARs
- Short term State Protective Action Guides (PAGs), like the PARs and EALs are event driven
- Long term PAGs are based on environmental monitoring



Emergency Dose Assessment Program - (EDCP)

- **Emergency Dose Assessment Program (EDCP) uses several types of inputs:**
 - Meteorological Data
 - Concentration
 - Effluent Radiation Monitor readings
 - Field Team measured dose rates and sample data



Emergency Dose Assessment Program - (EDCP)

- **Emergency Dose Assessment Program (EDCP) also uses these inputs:**
 - Flows
 - Ventilation flow rates
 - Containment Pressure changes
 - Containment Design Leakage rates
 - Containment High Range Area Monitor readings
 - Isotopic data



Summary

- Current CDA uses CHARMS and PASS
 - not timely for input to EAL and PAR decisions
- Proposed revised CDA uses only in-plant instrumentation
 - may be timely for input to EAL and PAR decisions
- Should increase effectiveness of E-Plan decision making process



Conclusions

- PASS sample results are not timely for EALs and PARs
- Final decision on PARs are not sensitive to isotopic knowledge
- With revised CDA, there will be no change in the ability to declare EALs or issue Protective Action Recommendations



Distribution: Mtg. Summary w/ WOG Re CDAG Issues Dated March 16, 1999

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