OG-99-015
March 16, 1999

Project Number 694

Mr. Peter C. Wen
Project Manager,
Generic Issues and Environmental Projects Branch
Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Westinghouse Owners Group
**Transmittal of Responses to NRC Comments from the February 24, 1999 Core Damage Assessment Meeting (MUHP-1302)**

References: 1) P. C. Wen, Memorandum to File, "Discussion Topics for February 24, 1999 Meeting with Westinghouse Owners Group Regarding WCAP-14696, 'Core Damage Assessment Guidance'," February 17, 1999

Dear Mr. Wen:

At the NRC and Westinghouse Owners Group (WOG) meeting on February 24, 1999, the NRC provided a list of comments/questions on WCAP-14696, "Core Damage Assessment Guidance" (ref. 1). These comments/questions and the associated WOG responses were discussed during the meeting. Attachment A documents the WOG responses to these comments/questions. Pending final resolution of these comments/questions, the WOG will revise WCAP-14696 as necessary. If you require further information, feel free to contact Mr. Ken Vavrek in the Westinghouse Owners Group Project Office at 412-374-4302.

Very truly yours,

Louis F. Liberatori Jr., Chairman
Westinghouse Owners Group

attachments

cc: WOG Steering Committee (1L, 1A)
WOG Primary Representatives (1L, 1A)
WOG Analysis Subcommittee Representatives (1L, 1A)
WOG Licensing Subcommittee Representatives (1L, 1A)
A. P. Drake, Westinghouse, ECE 5-16 (1L, 1A)
J. B. O'Brien, USNRC OWFN 9H15 (1L,1A)
R. J. Palla, Jr., USNRC OWFN 8H7 (1L,1A)

Attachment A
Westinghouse Owners Group Response to NRC Comments/Questions
on WCAP-14696, Core Damage Assessment Guidance

| Ref. | NRC Comment/Question | WOG Response |
|---|---|---|
| P4 | As discussed in Section 1.3, some utilities use the core damage assessment methodology (and/or the post-accident sample system) to select the source term to be used in offsite dose assessment. For those plants, the results of the core damage assessment guideline (percent core damage) would need to be translated into a source term estimate. However, the same fission product (FP) behavior that complicates core damage assessment (e.g., holdup in the reactor coolant system, containment, and sumps) would need to be accounted for in this translation. The methodology should discuss how the core damage assessment results would be used to determine source terms at these plants especially if information presently provided by the post-accident sample system (PASS) is no longer available. | The source term for offsite dose assessment can be estimated from a variety of sources, including core damage assessment, the containment radiation monitor, results of sample analyses, offsite or onsite radiological survey results, PRA analyses and/or licensing analyses. Appropriate adjustments should be made to core damage assessment results or other sources of information that are used to estimate a source term for offsite dose assessment, per current plant conditions.<br><br>Typically, plants use either a source term library developed from the licensing basis analyses and/or the containment high range radiation monitor. Options for using core damage assessment / sample analysis results were included in the report for completeness since the WOG did not do a complete survey of all plants. In particular, Wolf Creek (the lead plant for NRC review of this topical report) does not use the information from the core damage assessment to assign a source term for dose assessment.<br><br>Further, it is beyond the scope of the core damage assessment methodology to specify the appropriate adjustments that should be made for using core damage assessment results (or any other methods) to define the source term for offsite dose assessments. |
| P15, para3 | The document states that an indicated temperature of 1200 °F can be translated to a peak cladding temperature of 1400°F. Does this imply that an indicated temperature of 1500°F can be translated to a peak cladding temperature of 1700°F? What if the indicated temperature was 1900°F? Do factors such as system pressure, or extended periods of core uncovery result in a different indicated-to-actual temperature relationship? How will the operator or TSC Core Damage Assessment Team know when an instrument has failed? | The core exit thermocouples measure the temperature of the fluid exiting a fuel assembly. Investigations of the behavior of the thermocouples relative to fuel cladding and fuel pellet temperatures were conducted during the development of the generic Westinghouse Owners Group Emergency Response Guidelines (ERGs), which are the basis for the Emergency Operating Procedures (EOPs) at WOG plants. Specifically, in the ERGs (and EOPs) the core exit thermocouples are used in the symptom based approach to diagnose the onset of Inadequate Core Cooling (ICC) and the use of the FR-C.2 "Response to Degraded Core Cooling" and FR-C.1 "Response to Inadequate Core Cooling" procedures when the core exit thermocouples indicate temperatures of 700 °F and 1200°F respectively.<br><br>The investigations included the review of analytical and experimental results. Conservative licensing code analyses for a range of loss of coolant accidents, as reported in WCAP-9753, show that the core exit thermocouple temperatures can lag the fuel rod peak cladding temperatures by 200°F to 400°F. The discussion of results describes the conservatisms in the modeling of these analyses and concludes that the core exit thermocouples would be much closer to the actual peak cladding temperatures than is |

| Ref. | NRC Comment/Question | WOG Response |
|---|---|---|
| | | shown by the analysis results. Analyses using the best estimate MAAP4 code, which models the core as a lumped node (does not model separate cladding and fuel pellet nodes), show that the core exit thermocouples indicate 50°F to 100°F lower than the peak fuel node temperatures. Finally, experiments performed by INEL and reported in NUREG/CR 3396, "Detection of Inadequate Core Cooling With Core Exit Thermocouples: LOFT PWR Experience" conclude that the core exit thermocouples may be several hundred degrees lower than the peak temperatures in the core. Further all of the evidence supports the conclusion that the thermocouple indications were biased by the core / cladding temperatures in the upper portion of the core.<br><br>Based on this information, it is concluded that the core exit thermocouples provide an adequate measure of core temperatures to estimate temperatures at which potential cladding damage (i.e., above about 1400°F) and core overtemperature (above about 2400°F) may be occurring. Additionally, the difference between the core exit thermocouple indication and the actual clad temperatures varies with core conditions and, for the purposes of the core damage assessment is assumed to be a constant 200°F value. Finally, since there are a large number of core exit thermocouples that are nearly uniformly distributed radially throughout the core, the failure of individual thermocouples should be easily detected based on comparison with indications from adjacent thermocouple locations. This validation of instrumentation indications by comparison to other information is already part of EOP and SAMG training and therefore is not unique to core damage assessment. |
| P71, para5 | The document states that "the NUREG-0737 requirements for a core damage assessment based on samples of plant fluids does not have a valid basis using the current understanding of fission product behavior and the progression of core damage accidents." In an ideal situation, this may be true. But in reality many of the on line instruments may not be reliable and can not be the only source to obtain the status of the core. The grab samples are critical even though the results may not be available within a short time. | Based on information presented in the CDA topical report, the use of fixed in-plant instrumentation provides a more reliable (timely, accurate and available) indication of the amount of core damage, compared to analysis of samples of plant fluids. This is based on the evolution of knowledge on the progression of core damage accidents and the behavior of fission products and has led to the change in CDA methodology from one relying on analysis of samples to one relying on fixed in-plant instrumentation. The NUREG-0737 requirements for numerous samples of plant fluids are not based on current knowledge of fission product behavior; the analysis results from these samples are not timely, may not be accurate and may not be available, even in the long term after a core damage accident. Thus, the sampling requirements are not consistent with the overall intent of the post-TMI requirements, which is to provide methods to ascertain the plant conditions.<br><br>Typically, plants (including Wolf Creek, the lead plant for the NRC review of this topical report) develop Emergency Action Levels (EALs) and Protective Action |

| Ref | NRC Comment/Question | WOG Response |
|---|---|---|
| | | Recommendations (PARs) based on pre-defined plant indicators to provide a pre-determined level of actions necessary to protect the public. This diverse set of indicators will already have signaled the onset of core damage and the Emergency Response Organization will issue the required EALs and PARs long before a core damage estimate is completed, even using the revised methodology presented in this topical report. |
| P33, para4 | Describe how the PASS systems meet design specifications of NUREG-0737 and Regulatory Guide 1.97 in view of the deficiencies identified. | Generically, it is believed that the current PASS systems meet the detailed requirements of NUREG-0737 and Regulatory Guide 1.97. For example, the requirements within NUREG-0737, Item II.B.3 (11) require numerous design considerations which should be considered (e.g. provisions for purging sample lines, for reducing plateout, for appropriate disposal of samples). The PASS installation at WOG utility plants, including Wolf Creek, has been reviewed numerous times by both utility and NRC personnel during initial licensing and/or during the subsequent years. During all of these examinations, no design discrepancies in violation of either NUREG-0737 or Regulatory Guide 1.97, Rev. 2 have been identified.<br><br>The information provided in the topical report refers to current knowledge of fission product behavior that impacts the interpretation of the results of radionuclide samples. Based on the identified uncertainties in fission product behavior, the results of sample analysis were concluded to be less reliable (in terms of timeliness, accuracy and availability) than fixed in-plant instrumentation which facilitated the change in core damage assessment methodology. |
| P37, para3 | Describe what evidence exists to demonstrate that the RCS pressure/temperature relationship provides a reliable basis for judging whether clad rupture will occur. Justify that the codes do not contain conservatisms that may tend to over-predict clad failure. Should address this through a validation activity. | Refer to note 1. |
| P45, para2 | The WCAP states that for containment radiation levels corresponding to rupture of 60% of rods, little or no FP release from fuel pellet matrix should be occurring. The basis for the 60% value is not clear. Previous discussion and figures center on a 0.5% pellet release and show that radiation levels for a 0.5% pellet release would correspond to about a 20% gap release. However, the guideline and the value for CRM2 seems to be based on a 1% pellet over-temperature release. Clarify whether this is the cause of the disparity and make the text and guideline consistent. | A detailed study of the progression of a core damage accident shows that between two thirds and 90% of the cladding will experience rupture before any of the fuel pellets experience a bulk temperature significantly in excess of 2400°F, the point where fuel overtemperature releases begin to occur (see page 43 of the topical report). The two-thirds value was rounded off to 60% in the discussion on page 45. The upper end of the range is represented by large LOCA events and the lower end of the range is represented by non-LOCA events. Therefore, choosing a fission product release value corresponding to 1% fuel overtemperature as the onset of diagnosis of fuel overtemperature, which corresponds to about 35% clad damage (middle of page 43 of the topical report) assures that the onset of fuel overtemperature will be conservatively diagnosed (i.e., fuel |

| Ref | NRC Comment/Question | WOG Response |
|---|---|---|
| | | overtemperature may be diagnosed when none is occurring, but is highly likely to be diagnosed when overtemperature begins) |
| P50, item2 | We disagree with the statements that "the results of analyses of samples of plant fluids would not provide any clarification of the type and degree of core damage compared to that obtained using fixed in-plant instrumentation. Thus, the need for post accident sampling of plant fluids for the determination of the type and degree of core damage is not required by the new WOG Core Damage Assessment Guideline." Although the results from sampling may not be available in a short time for any decisions, they would be very helpful for confirmation of the readings given by the in-plant instrumentation. The results of the grab samples can be used for the determination of the type and degree of core damage. | The core damage assessment methodology presented in the topical report was developed for the purpose of providing input to the Emergency Planning decision-making process, even though it is not required to make EAL declarations or PARs. In this context, sampling would not provide any clarification due to timeliness, accuracy and availability issues described in the topical report.<br><br>In the longer term, core damage can be assessed by a diverse set of plant indicators. Samples are but one type of information that may be used in assessing the plant status. See also the response to comment P2 of 10 (directly below). |
| P2of10 | It seems that the revised guideline is relevant only while the event is continuing to deteriorate (since CETs and RCS pressures are different then at the time of core damage, and no longer indicative of what had occurred earlier), and that following recovery, some other type of assessment (i.e., PASS or grab samples) would be needed. It is strongly recommended that the scope of the guideline be expanded to: (1) include core damage assessment during both the core degradation and accident recovery phases, and (2) address the respective roles of CDAM and PASS/grab samples during each phase. As such, the guideline would represent an integral approach for assessing the state of the core during and following core degradation. | Evaluation of an accident may be described in simple terms of 1) diagnosis, 2) protection, and 3) explanation / recovery. Diagnosis and protection are addressed via the EALs and PARs. The EALs and PARs do not require information from a core damage assessment. However, information from a core damage assessment may be used to validate the EALs and PARs during an event. Additional evaluations of core damage using sample data would more appropriately be in the longer term accident explanation time frame.<br><br>The three phases of a core damage accident can be characterized by the criteria for entering and exiting the WOG based Severe Accident Management Guidelines (WOG SAMG). The SAMG are being used during the diagnosis and protection phases. During this time, the EAL should be at a General Emergency level and PARs are issued, based on the best available information. Fixed in-plant instrumentation is a more appropriate method for estimating core damage based on reliability (timeliness, accuracy and availability) and on minimizing radiation exposures to plant workers. The criteria for exiting the SAMG are based on the establishment of a long term stable core and containment state which is diagnosed from fixed in-plant instrumentation. After the SAMG are no longer in use, the EALs are likely to be de-escalated from a General Emergency condition and PARs are not being continually assessed on a high priority basis. This is the beginning of the explanation and recovery phase of the event. During the explanation / recovery phase, confirmation and/or refinement of the core damage estimates would be made by the utility to assist in planning for recovery and cleanup. |

| Ref | NRC Comment/Question | WOG Response |
|---|---|---|
| | | Samples may be one of the tools that could be used at that time. However, it is not credible to develop a detailed core damage assessment method for this explanation / recovery phase due to the wide range of possible plant conditions that could exist at that time. Even if such a method were to be developed, the assembled experts would define the most appropriate method, which may include (but is not limited to) fixed in-plant instrumentation, portable instrumentation, etc. |
| P3of10 | It is not clear that evaluation of clad rupture based on pressure/temperature criteria is valid in practice. Some type of validation of this concept, as well as the recommended values for RCP2, CET3, and CET4, appears necessary. | See response to P37, Para3 (Above) |
| P18of27 | Recommend retaining the CH1 setpoint for ice condensers since this hydrogen concentration is well below lower flammability limit and would not be impacted by igniter operation. Also, rather than deleting the CH2 setpoint for ice condensers, the TSC should consider whether igniters are actuated, and whether there is evidence of a burn. | The setpoint portion of the Core Damage Assessment Guidance will be revised to retain all of the hydrogen setpoints (CH1 through CH5) for ice condenser plants, with the note that containment hydrogen would only be a reliable measure of fuel overtemperature for accident sequences in which the hydrogen igniters were not in operation. |
| P19of27 | In establishing CH2, CH3, CH4, and CH5, the WCAP recommends certain assumptions regarding the amount of metal-water reaction and fraction of hydrogen released to containment. The validity of the recommended values should be illustrated by comparing these values with the results from best estimate code calculations (MAAP, MELCOR, SCDAP) for representative severe accident sequences. | Refer to Note 2. |
| P2, para4 | Although there is no specific regulation for core damage assessment, 10 CFR 50.47(b)(9) requires licensees to have "adequate methods, systems..." Part of this assessment process may include core damage assessment if it is used as part of the licensees emergency plan to make protective action recommendations or to classify events requires emergency action levels (EALs). | Based on generic information from WOG utilities, the diagnosis of core damage (and the amount of core damage) is not used in the prescriptive formulae for the selection / declaration of Emergency Actions Levels (per NESP-007/ NEI-97-03 or NUREG 0654 methodologies) or the issuance of Protective Action Recommendations. The appropriate EALs and PARs are decided based primarily on plant parameters that can be determined from fixed in-plant instrumentation, the emergency operating procedures in-effect, and/or the status of plant systems.<br><br>The only possible exception to this is the EAL criteria based on a reactor coolant system radioactivity level of 300 microcuries per gram DEI. The 300 uCi / gm DEI is defined in the EAL bases as an indicator of potential fuel damage of ~2 - 5%. Although reactor coolant activity levels are not used in the revised core damage assessment presented in this topical report, using the new CDAM would provide a different method of coming to |

| Ref | NRC Comment/Question | WOG Response |
|---|---|---|
| | | similar conclusion (e.g. Two CETs exceeded 2000°F).<br><br>The exclusion of reactor coolant activity as a direct indicator of core damage was based on a review of core damage sequences (documented in WCAP-14986) that leads to the conclusion that other EAL criteria would direct the appropriate EAL declaration in a shorter time frame than the coolant activity level and therefore the 300 microcurie per gram DEI specification is redundant. |
| P3, para6 | Incorrect statement regarding activity levels used as basis for EAL for RCS clad barrier (it is based on normal coolant activity) | See response to comment P4, Para 1 (directly below). |
| P4, para1 | Incorrect statement regarding the amount of clad damage (5- 10% should be 2 - 5%) | The topical report will be revised to read "2 to 5%". |
| P5, para4 | Should be updated to reflect latest RTM (96) information | The topical report will be revised to reference RTM-96. No other change to the topical report is required as a result of referencing this later document. |
| P7 | Table 1 should have another column with "Indicated Core Exit Temperature". | Since most of the temperatures in the table are beyond the core exit thermocouple range, it would only be confusing to add this information. Adequate information is provided in the text to make the transition from the core temperature to a core exit thermocouple indication where appropriate. |
| P9, para1 | Should define which FP species are considered "non-volatile". The last sentence is not true if non-coolable geometry forms or molten pool is retained due to external reactor vessel cooling. | This is discussed in the topical report in Section 2.2. |
| P9, para2 | Next to last sentence seems to say that any tellurium (Te) or small portions of non-volatiles would indicate that the core is ex-vessel. This is not true, as evidenced by the early in-vessel source term in NUREG-1465, and should be clarified. | While the core is in-vessel, it is expected (based on MAAP analyses) that the release of Te and non-volatiles to the containment would be negligible for all core damage sequences (except possibly a hot leg break near the reactor vessel nozzle) due to retention in the reactor coolant system. Therefore, it is believed that the presence of Te and non-volatiles in the containment is not a reliable indicator of fuel overtemperature, even in the case where the molten pool is retained in-vessel due to external reactor vessel cooling. However, it is recommended in the topical report that the NUREG-1465 source term be used to predict the containment radiation monitor response. Sensitivity studies performed during the WOG CDA methodology development showed that the impact of Te and non-volatiles on the containment radiation monitor response for core damage indication are very small and well within the variations in containment radiation monitor indications for differing assumptions on Cs and I retention in the reactor coolant system. |
| P11, para4 | Provide basis for following statement: From the perspective of potential offsite releases of fission products and the need to recommend offsite protective actions, there are only three | The dominant radionuclide species for offsite radiological protection recommendations are noble gases, iodines and cesiums. Since nearly 100% of the noble gases, iodine and cesiums are released from the fuel pellet at pellet temperatures below the melting point, |

| Ref | NRC Comment/Question | WOG Response |
|---|---|---|
| | levels of core damage that are important: no damage, fuel rod cladding damage and fuel over-temperature damage..... Thus , the diagnosis of core melting as specified in reference 4 will not be developed as part of this core damage assessment | the onset of fuel melting is not a significant technical issue for core damage assessment with respect to its use as input to decisions for EALs and PARs. Also, it should be noted that code predictions show that the time required for a portion of the core to go from 2400°F (the onset of overtemperature releases) to 4000°F (the onset of core melting) is on the order of 5 minutes; the heat-up rate is enhanced by the exothermic zirc water reaction heat. From a non-technical perspective, one could equate core overtemperature with core melting without loss of concept. |
| P12, para1 | Gravitational settling is said to be primary removal mechanism. Is this based on IDCOR or most recent work? If the former, confirm that this statement is supported by results of the more recent work. | The fission product behavior models in the MAAP code, which is the primary analytical tool used in the development of the core damage assessment methodology, are continually validated against recent experimental and analytical data. These benchmarks continue to show that gravitation settling is the primary means of aerosol deposition in the containment |
| P13, para4 | The document states that analysis of samples of reactor coolant, containment sump water and containment atmosphere for specific radionuclides could be useful in estimating the core damage. This statement is in direct conflict with the final conclusion given on page 50. | The statement on page 13 is made in the context of possible alternatives to assessing core damage. The statement on page 50 is made in the context of a conclusion as a result of the investigations that were made to develop the WOG core damage assessment methodology. |
| P18, para3 | The statement that there is no information related to the delay time for the hydrogen reading to stabilize appears incorrect. Explain why this type of information isn't already available from hydrogen monitor calibration testing. | Typically, calibration testing of the containment hydrogen monitor is performed using several gas mixtures of known, but different, hydrogen concentration. Typically, one calibration gas mixture is in the range of 0.5% to 1% by volume, at room temperature and another near 4% by volume. The operating manual from the manufacturer of one type of hydrogen monitor used by some plants suggests that an initial waiting period should be used to permit the reading to stabilize and an additional much smaller waiting period should be used when calibration settings (which are step changes) are changed. These waiting periods are used during calibration to accurately align the monitoring channel. In a core damage accident such instantaneous step changes are not likely and the hydrogen monitor should be able to adequately trend the hydrogen concentration with no delay for readings to stabilize. Even in the event of a step change in containment hydrogen concentration during a core damage accident, the monitor would be expected to trend the step change with only a slight, unspecified delay for the monitor response. This delay cannot be derived from calibration testing due to potential differences in the timing and/or magnitude of any step change.<br><br>The topical report will be revised to replace the last two sentences of the third paragraph on page 18 with the above explanation. |
| P18, para4 | Should include some discussion of how auto-ignition, random ignition, or diffusion flames in large dry | This is included in the Background Document for the CDA Guidance step. See page 19 of 24 of the Background Document. |

| Ref. | NRC Comment/Question | WOG Response |
|---|---|---|
| | containments may also result in low hydrogen concentrations not indicative of the degree of core damage | |
| P19, para3 | The requirement is for sampling and analyzing within 3 hours of the decision to do so (not accident initiation). We are not aware of any plant who has a 24 hour sample requirement | This statement will be corrected in the topical report to refer only to the time period of 3 hours of the decision to sample. |
| P22, para2 | The document indicates that a 0.1% clad damage is used for EAL classification. Please describe how this level of clad damage would generally be detected | To be consistent with the response to Comment P4, para1, the topical report will be revised to change the reference for the EAL basis from 0.1%, 1% and 5% clad damage to 2 and 5% clad damage. |
| P22 para3 | Should note that dose projection should not just take into account CDA but also PASS results | Generically (and specifically at Wolf Creek which is the lead plant for the NRC review of this topical report) dose assessment chooses the appropriate data for the application, regardless of where the data is obtained. The offsite dose projection techniques generally do not use core damage assessment quantitative data or PASS results as the primary method of defining the source term for the offsite dose projection. PASS does not provide timely, useful information; it is point in time data and is, by its nature, always historical. See also response to comment P4 above. |
| P24, para2,3 | The discussion seems to over-emphasize the amount of information that can be inferred from core exit thermocouples (CETs). This level of discussion seems unnecessary since the operators will not be able to discern the differences between sequences based on the CET data (given that it is only available up to about 2100F), and since the guidelines do not include such guidance/instructions. | The information in this discussion is meant to provide background on the core heat-up signature and is used to support the assumption that core melting generally does not occur until 60 to 90 % of the fuel rod cladding is failed as discussed in the response to Comment P45 Para2. |
| P25-28 | Figures 2 and 3 present "core exit thermocouple indications" values which would not be possible for the existing instruments. Should indicate that these are "predicted peak fuel temperatures" rather than "core exit thermocouple indications." | The title of these figures will be changed to indicated that they are predicted core temperatures, instead of core exit thermocouple indications. |
| P29, para1 | The discussion seems to imply that operators would assign a different level of reliability to CET readings based on RCS pressure. This level of discussion seems unnecessary since the guidelines do not include such guidance/instructions | The last sentence will be deleted from the paragraph since it provides information that may be misleading and does not directly support the case for the revised core damage assessment. |
| P29 | Should include some discussion of hydrogen production from radiolysis and corrosion, and how it compares to: (1) production from clad over-temperature, and (2) the 1% | As stated in the response to Comment P2 of 10, this core damage assessment methodology is meant for use during the transient portions of a core damage accident until the plant is brought to a controlled stable condition. For purposes of developing a |

| Ref. | NRC Comment/Question | WOG Response |
|---|---|---|
| | hydrogen concentration value discussed in Section 6.1 | numerical methodology, this time period is assumed to be less than the first 24 hours of the accident. Hydrogen production from radiolysis and corrosion during the first 24 hours of an accident, using conservative design basis assumptions, is generally a factor of 5 to 10 less than the hydrogen production from zirc-water reactions during core uncovery. Additionally, within the WOG CDA methodology, containment hydrogen is a secondary indication of the amount of core overheating due to large uncertainties in the amount of hydrogen produced by zirc water reactions for any individual core damage accident sequence. The hydrogen production from radiolysis is within the uncertainty of the amount of hydrogen produced from zirc water reaction. Therefore, hydrogen production from radiolysis can be neglected in this methodology. |
| P32, para1 | The amount of cesium hydroxide in the containment atmosphere is said to be much higher in sequences with sprays or fans. Confirm that this is correct. | Cesium hydroxide is a hygroscopic species. The amount of steam in the containment has a large impact on the depletion rate of cesium hydroxide from the containment atmosphere. The gravitational settling of cesium hydroxide is enhanced when water (steam) is absorbed. The depletion rate is higher for accident sequences without containment spray or fans due to the higher steam content in the containment atmosphere. |
| P39, para3 | The discussion seems to overstate the impact of instrument accuracy on low end readings. Isn't accuracy given as a function of measured value? The message could be incorrectly interpreted to mean that hydrogen concentration values less than 1% are not high enough to be taken seriously and used in the assessment. | Instrument errors are typically provided as a percentage of the full scale indication. Thus, in the case of the hydrogen monitor (full-scale is 10 percent hydrogen, maximum error is 10% of full-scale), an indication of 1% means that the containment hydrogen is between 0% and 2%. |
| P2of10 | In the event of an SGTR or ISLOCA, containment radiation monitors will not provide useful information. The assessment of plant status would then rely primarily on CETs. Explain why use of other radiation monitors (e.g., in the steam line or stack) is not suggested to provide confirmation of core damage. | For SGTR and ISLOCA, the core exit thermocouples provide the only reliable method for estimating the amount of core damage. These two accidents result in the direct release of radioactive material to the environment and the appropriate EALs and PARs would be made based on plant parameters, emergency procedures in effect, and systems availability indicating loss of (or potential loss of) barriers. Release point monitors are also directly used in EAL and PAR assessments. In general, the appropriate EALs and PARs would be recommended prior to core damage; the use of the release point rad monitors and onsite / offsite radiation surveys serves directly as a backup in the EAL and PAR assessments. CDA is not needed for the early response to these events. Release point monitors could be useful in validating the relative amount of core damage (in broad categories) estimated from the core exit thermocouples, but are not included in the revised CDA methodology described in the topical report (nor in the existing CDA methodology approved by the NRC in 1987). |
| P4of10 | It is not clear whether agreement within 50% can realistically be achieved. This should be evaluated through some type of validation. | See response to Comment P19 of 27 |

| Ref | NRC Comment/Question | WOG Response |
|---|---|---|
| P6of1?, B2 | The equation in Section B2.b estimates the extent of over-temperature based on the number of CETs exceeding CET3. However, CET3 is used to denote clad rupture from pressure effects. Explain why a temperature value associated with over-temperature damage (e.g., CET2) is not recommended instead. | The topical report will be revised to change CET3 to CET2. |
| P4of27 | The recommended value for CET2 (2000F) seems too high to represent a stable condition. If you reach this value, PCT and pellet temp would be several hundred degrees higher and CETs would rapidly increase beyond 2000F. (Discussions in Section 2.5 of WCAP-14986 suggest that a PCT above 1800F would result in significant hydrogen generation and core melting.) Explain why the value selected for CET2 is not one that can be sustained without proceeding to core melt. Should address this through a validation activity. | The value of 2000°F for core exit thermocouples was chosen to represent a best estimate of core temperatures approaching the fuel temperature for significant fission product release from the fuel pellet. This acts as a screening threshold between clad damage and overtemperature. A stable core exit thermocouple indication of 1800°F as suggested in the comment is a hypothetical condition not representative of typical core damage scenarios. If a number of core exit thermocouples became stable at 1800°F or 2000°F, this would be indicative of a case where little or no zirc water reactions are occurring and there is little or no likelihood of fuel overtemperature releases (fuel pellets greater than 2400°F). Thus, clad damage is the appropriate indication and either value for CET2 would result in the same diagnosis. The setpoint value of 2000°F was chosen to be a realistic indication of core overtemperature and the use of a value of 1800°F introduces some degree of unwanted conservatism. |
| P5of27 | The recommended value for CET3 appears inconsistent with the values discussed on P37. Should address this through a validation activity. | The topical report will be revised to correct this inconsistency. A value of 1600 psig will be shown on page 37 to represent boundary between high and low reactor coolant system pressures for estimating fuel rod clad rupture. |
| P6of27 | Use of a value for CET4 based on conservative analyses, as recommended, will tend to over-predict the extent of clad damage. This may or may not be important depending on the particular sequence. Explain why this value is not based on best estimate analyses, similar to approach for determining CET3. | It is stated that the core exit thermocouple indication of 1200°F is based on conservative analyses. These conservative analyses provide support for the use of 1200°F in the EOPs for actions to recover from inadequate core cooling before peak clad temperatures exceed about 2200°F. The analyses are conservative in that context. More realistic analyses show that the clad temperature can be in the range of 1400°F to 1600°F when the core exit thermocouples indicate 1200°F. With a completely depressurized reactor coolant system, this can result in clad failure as discussed in the response to Comment P37, Para3. Thus, the value of 1200°F is appropriate here. |
| P8of27 | The recommended value for RCP2 appears inconsistent with the values discussed on P37. Should address this through a validation activity. | The topical report will be revised to correct this inconsistency. A value of 1600 psig will be shown on page 37 to represent boundary between high and low reactor coolant system pressures for estimating fuel rod clad rupture. (same as P5 of 27) |
| P10of27 | Explain how sensitive the recommended RTD temperature is to the specific scenario. Provide an estimate of RTD temperature that would be seen at the time the CETs reach 1200F in several representative sequences, e.g., (1) LOCA | The RTD indications are discussed in detail in the WOG SAMG Executive Volume, Section 9.1, "Instrumentation". Since the range of the RTDs is limited to 700°F, the indication should be off-scale high for most core damage sequences. |

| Ref. | NRC Comment/Question | WOG Response |
|---|---|---|
| | in hot leg and in cold leg, (2) transient, (3) SGTR | |
| P12of27 | Describe why RCS pressure isn't a consideration in using reactor vessel level instrumentation for confirmation, since it would affect swollen level. Confirm what pressure was assumed in determining the recommended value, and whether this value bounds the RCS pressure effect. | Since RVLIS is used as a confirmatory indication, the precision of the level with respect to core damage is not of key importance. The core mid-plane value were chosen to be consistent with the EPRI Severe Accident Management Technical Basis Document (TR-101869) description of the OX plant damage state, Table Z-1 of that report. The value of top of core was chosen as a positive indicator that the core is not uncovered. |
| P9of24 | Bullets 3 and 4 indicate that the core should be partially uncovered if clad damage has occurred. This is true if core damage is occurring at that moment, but the core may have uncovered previously, and then been recovered. The discussion should indicate that the reactor vessel level and SRM histories need to be considered rather than the instantaneous values. (Same comment on P18 of 24) | The topical report will be revised to state that the core exit thermocouple, loop RTD, reactor vessel level and SRM histories need to be considered in addition to current values. |
| | **Editorial Comments** | |
| P2, para2 | Change "Criteria II.B.3" to "Item II.B.3" | The topical report will be revised to read "Item II.B.3". |
| P37, para3 | Change "if the RCS temperature is less than 1050 psig" to "if the RCS pressure is less than 1050 psig." | The topical report will be revised to read "if the RCS pressure is less than 1600 psig", per this comment and our response to comment P5of 27. |
| P44 | Figures 5 and 6 should be replotted on a scale that would permit interpolation | Since plant specific versions of these figures are developed as part of developing a plant-specific CDA, Figures 5 and 6 only serve as examples. Therefore, there is no need to re-plot these figures to permit interpolation |
| P46, Table7 | The words "fuel rod over-temperature" and "fuel over-temperature" seem to be used interchangeably, as are the words "core over-temperature" and "fuel over-temperature". Please clarify in the document whether these words are intended to be synonymous | The topical report will be revised to state that the descriptive phrases are meant to be completely interchangeable |
| P7of24, para6 | The words CET4 and CET3 are switched in the second sentence | The topical report will be revised to correctly identify CET3 and CET4 on this page |

Note 1:

NRC Comment P37, paragraph 3:

Describe what evidence exists to demonstrate that the RCS pressure/temperature relationship provides a reliable basis for judging whether clad rupture will occur. Justify that the codes do not contain conservatisms that may tend to over-predict clad failure. Should address this through a validation activity.

WOG Response:

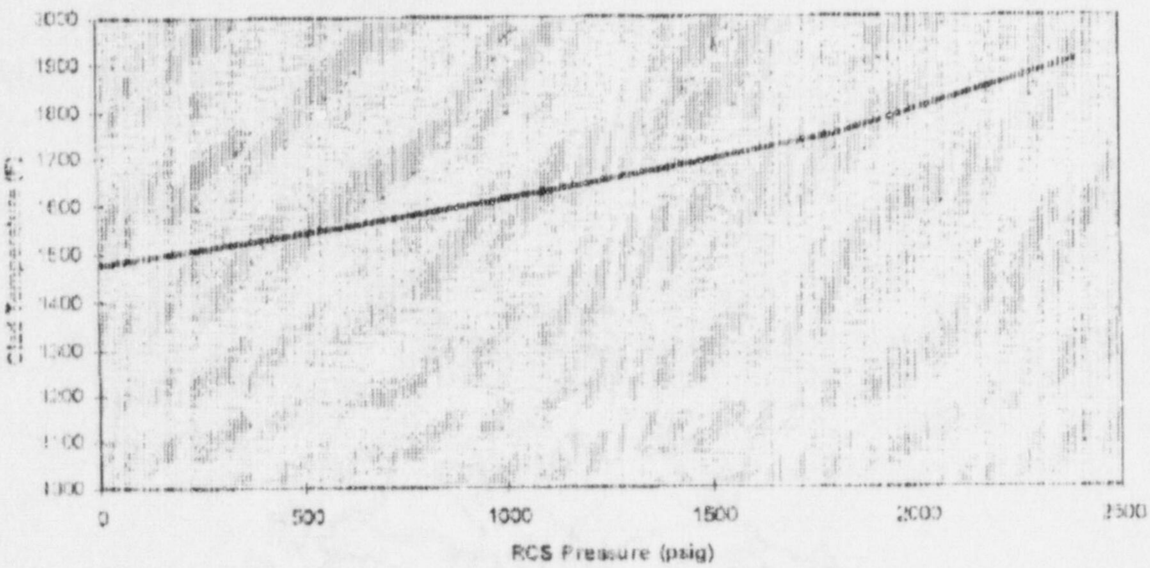
The modeling of fuel rod cladding burst using MAAP3B and MAAP4 computer code analyses, for a wide range of core damage accidents, was used to develop the correlation for clad damage in the WOG Core Damage Assessment Guidance (WDPA-14696)

MAAP (both the 3b and 4 code versions) models fuel rod cladding rupture based on the stresses developed in a fuel rod and the burst strength of the fuel rod cladding. The model contains several details that are critical to the modeling of fuel rod cladding rupture:

- The burst strength of the fuel rod cladding is a function of cladding temperature, a realistic assessment of the strength of the fuel rod zircaloy, from NUREG/CR-0497, is used.
- The stress developed in the fuel rod is compared to the burst strength to predict fuel rod failure, the cladding stress is a function of the delta-pressure across the cladding, the cladding thickness and cladding diameter.
- The rod internal pressure during a core temperature transient is a function of the initial fill pressure, the xenon generation and the fuel rod temperature; the xenon generation is a function of the fuel rod burnup
- Elastic and plastic strain is evaluated for the fuel rod conditions. The fuel rod diameter and thickness and internal pressure are adjusted to account for elastic and plastic strains
- Since plastic strain is a rate phenomena, the fuel rod failure is a function of the fuel rod temperature and time at temperature.

The results of a number of MAAP analyses for a typical 4 loop Westinghouse PWR, at different reactor coolant system pressures, were used to develop a generic profile of clad burst vs. RCS pressure. This is reproduced in Figure 1.

Typical Fuel Rod Clad Failure Temperatures




This figure shows rod burst for a large LOCA at a clad temperature of about 1475°F. This corresponds to a core exit thermocouple indication of just over 1200°F (Page 37 of the WCAP-14696). At a 2200 psig reactor coolant system pressure, the corresponding clad temperature at which failure would be predicted to occur is about 1800°F this corresponds to a 1600°F core exit thermocouple indication.

Taking into account variations in fuel rod strength, clad temperature vs. core exit indication, burnup, time at temperature for plastic strain to progress, etc., the values of 1200°F and 1600°F were chosen to represent high and low RCS pressure conditions at the time of clad overheating to the damage point.

A review of core damage sequences from PRA studies shows that most sequences either remain above a value of about 1600 psig or fall to at least the steam generator secondary side pressure relief setpoint (e.g., 1050 psig) before fuel rod overheating occurs. For those events above this value (1600 psig), natural circulation in the reactor coolant system and reflux cooling from the steam generators is generally occurring which slows the core heatup rate and results in the a "typical heatup rate, regardless of RCS pressure. In addition, the core exit thermocouples would be expected to more closely track the cladding temperatures (i.e., less than 2000°F). Thus, a value of 1600 psig for the RCS pressure to distinguish between high and low pressure core damage events is appropriate.

Note 2

NRC Comment Page 19 of 27:

In establishing CH2, CH3, CH4, and CH5, the WCAP recommends certain assumptions regarding the amount of metal-water reaction and fraction of hydrogen released to containment. The validity of the recommended values should be illustrated by comparing these values with the results from best estimate code calculations (MAAP, MELCOR, SCDAP) for representative severe accident sequences.

WOG Response:

The estimates of fission product behavior are based on a large number of analyses, primarily with the MAAP3B and MAAP4 computer codes. The MAAP3B analyses were, in large part, used in the development of the WOG Severe Accident Management Guidance. The MAAP4 analyses were used to validate the MAAP3B analysis results. These analyses provide key evidence that the containment hydrogen and radiation levels can be impacted by the integrity of the reactor coolant system during core damage and can be correlated to the reactor coolant system pressure. Further, these analyses were used to develop the estimates provided in WCAP-14696 for the hydrogen, noble gas, cesium and iodine retention in the reactor coolant system for core damage accidents at high and low reactor coolant system pressures. In addition, the various in-vessel hydrogen generation estimates provided in WCAP-14696 were also derived from these MAAP analyses as discussed below.

A range of analyses are readily available and are summarized in Table 1 below. The WOG Core Damage Assessment (CDA) Guidance, WCAP-14696, is based on an assessment of these analyses. Fundamental to the WOG CDA is the philosophy that the core damage estimate should be as realistic as possible, but on the "conservative", or high, side. From the results below there is significant holdup of cesium and iodine in the reactor coolant system regardless of the reactor coolant system pressure. Additionally, there is significantly more retention in the reactor coolant system when the RCS is at high pressure, due to the increased residence time for these fission products in the reactor coolant system. It is also noted that depressurization of the reactor coolant system can but does not necessarily result in the transport of significant amounts of cesium and iodine aerosols to the containment. Within the context of developing a core damage assessment methodology that is easy to use (i.e., does not require expert knowledge of severe accidents), the amount of cesium and iodine fission products in the containment was directly related to the RCS pressure. Thus, if the RCS pressure was high, 98% of the cesium and iodine was assumed to be retained in the RCS; if the reactor coolant system pressure is low, 30% of the cesium and iodine is assumed to be retained in the RCS. These best estimate "bounds" are supported by the analyses.

The noble gases behave in manner similar to hydrogen in that the only retention is due to the gases that are pressurizing the reactor coolant system. The hydrogen / noble gas retention in the RCS is illustrated by the two analyses presented on page 29 of WCAP-14696. These analyses are consistent with the results of numerous other analyses reported in individual plant severe accident analyses. These analyses show that for the "transient case" where the reactor coolant system pressure is high (e.g., 2250 psig), as much as 50% (e.g., at 25 minutes after onset of zirc water reactions in the table) of the hydrogen (and consequently noble gases) can be retained in the reactor coolant system. The analyses illustrated in the report also illustrate a significant difference in the in-vessel hydrogen production that has been observed in other analyses, which can be correlated to RCS pressure at the time of hydrogen production. There is also applicable information on in-vessel hydrogen production presented in Expert D's elicitation on PWR In-vessel hydrogen for the NUREG-1150 review panel (NUREG/CR-4551, Vol. 2, Rev. 1, Part 1).

The RCS pressure cut-off for high and low pressure is based on engineering judgment from a large number of severe accident analyses.

For fission product retention in the RCS, the specific value of 1600 psig was chosen to represent those core damage sequences in which there was residence time in the RCS for fission products. This pressure would only include events that can be classified as transients (non-LOCAs) and very small LOCAs which depressurize the RCS sufficiently to generate an SI signal.

For hydrogen, the specific value of 1050 psig was chosen to represent those accident sequences that remain at or above the steam generator secondary side relief setpoint. In these sequences, the hydrogen generation is increased (compared to sequences at lower pressures) due to refluxing from the steam generators or natural circulation processes, both of which result in continued steam supply to the overheated core. Core damage sequences that occur at RCS pressures below 1050 psig tend to have higher steam losses from the reactor coolant system and the hydrogen generation is limited by steam availability

Table 1
Comparison of RCS and Containment Volatile Fission Products
for Core Damage Accident Scenarios

| Time (hrs) | Event | RCS Pressure (psig) | Cesium Iodide RCS | Containment |
|---|---|---|---|---|
| Station Blackout; No EFW (Ref 1) | | | | |
| 1.2 | SG Dry-out | 2250 | 0 | 0 |
| 1.5 | Core Uncovery | 2250 | 0 | 0 |
| 2.5(-) | Just Before Vessel Failure | 2250 | 96 | 4 |
| 4 | After Vessel Failure | 15 | 92 | 8 |
| Station Blackout; TD EFW Available for 4 hours; RCS Cooldown/Depressurization w/ SGs (Ref 1) | | | | |
| 7.3 | SG Dry-out | 200 | 0 | 0 |
| 9.3 | Core Uncovery | 2250 | 0 | 0 |
| 11.0(-) | Just Before Vessel Failure | 68 | 98 | 2 |
| 18.0 | After Vessel Failure | 15 | 92 | 8 |
| LOFW; EFW Available; RCS Cooldown/Depressurization w/ SGs (Ref 1) | | | | |
| 7.2 | Core Uncovery | 175 | 0 | 0 |
| 9.3(-) | Just Before Vessel Failure | 200 | 90 | 10 |
| 12.0 | After Vessel Failure | 15 | 88 | 12 |
| RCP Seal LOCA, ECC Recirc Failure; EFW Lost @ 8 Hrs (Ref 1) | | | | |
| 9.6 | Core Uncovery | 175 | 0 | 0 |
| 1[illegible].7(-) | Just Before Vessel Failure | 175 | 90 | 10 |
| 15.0 | After Vessel Failure | 15 | 89 | 11 |
| Station Blackout; No EFW; PORVs Open @ 1200°F CET (Ref 1) | | | | |
| 1.2 | SG Dry-out | 2250 | 0 | 0 |
| 1.5 | Core Uncovery | 2250 | 0 | 0 |
| 1.9(-) | Just Before PORV Opening | 2250 | 0 | 0 |
| 2.8(-) | Just Before Vessel Failure | 175 | 59 | 41 |
| 4 | After Vessel Failure | 15 | 58 | 42 |
| PORV LOCA; No EFW (Ref 1) | | | | |
| 1.5 | Core Uncovery | 1070 | 0 | 0 |
| 2.3(-) | Just Before Vessel Failure | 900 | 89 | 11 |
| 4 | After Vessel Failure | 15 | 87 | 13 |
| SGTR; No ECC; PORVs Open @ 1200°F CET (Ref 2) | | | | |
| 3.3 | Core Uncovery | 1800 | 0 | 0 |
| 3.6(-) | Just Before PORV Opening | 1800 | 0 | 0 |
| 4.3(-) | Just Before Vessel Failure | 120 | 50 | 42 |

| | | | | |
|---|---|---|---|---|
| 6 | After Vessel Failure | 15 | 49 | 42 |
| Note: Releases do not add to 100% due to release to atmosphere via SGTR | | | | |
| Medium size ISLOCA; No ECC; PORVs Open @ 1200°F CET (Ref 2 | | | | |
| 0.9 | Core Uncovery | 1100 | 0 | 0 |
| 1.1(-) | Just Before PORV Opening | 1100 | 0 | 0 |
| 2.0(-) | Just Before Vessel Failure | 120 | 37 | 18 |
| 4 | After Vessel Failure | 15 | 34 | 18 |
| Note: Releases do not add to 100% due to release to atmosphere via ISLOCA | | | | |
| Large Hot Leg Break; No ECC (Ref 3) | | | | |
| 0.4 | Core Uncovery | 20 | 0 | 0 |
| 1.25(-) | Just Before Vessel Failure | 20 | 38 | 62 |
| 4 | After Vessel Failure | 20 | 36 | 64 |
| Large Cold Leg Break; No ECC (Ref 3 | | | | |
| 0.4 | Core Uncovery | 20 | 0 | 0 |
| 1.25(-) | Just Before Vessel Failure | 20 | 58 | 42 |
| 4 | After Vessel Failure | 20 | 56 | 44 |
| Station Blackout; No EFW (Ref 4) | | | | |
| N/A | Core Uncovery | 2250 | 0 | 0 |
| N/A | Just Before Vessel Failure | 2250 | 93 | 7 |
| N/A | After Vessel Failure | 15 | 93 | 7 |
| Note: Not used in original CDA development, but included here for completeness | | | | |
| Station Blackout; No EFW; Loop Seals Clear (Ref 4) | | | | |
| N/A | Core Uncovery | 2250 | 0 | 0 |
| N/A. | Just Before Vessel Failure | 2250 | 92 | 8 |
| N/A. | After Vessel Failure | 15 | 92 | 8 |
| Station Blackout; No EFW; PORV Open @ 1200°F CET(Ref 4) | | | | |
| N/A. | Core Uncovery | 2250 | 0 | 0 |
| N/A. | Just Before Vessel Failure | 400 | 77 | 23 |
| N/A. | After Vessel Failure | 15 | 77 | 23 |
| Station Blackout; No EFW; Surge Line Creep Failure (Ref 5) | | | | |
| 1.7 | Core Uncovery | 2250 | 0 | 0 |
| 3.1 | Just Before Surge Line Failure | 400 | 95 | 5 |
| 4.0 | After Vessel Failure | 15 | 82 | 18 |
| Station Blackout; No EFW (Ref 6) | | | | |
| N/A | Core Uncovery | 2250 | 0 | 0 |
| N/A | Just Before Vessel Failure | 2250 | 91 | 9 |
| N/A | After Vessel Failure | 15 | 89 | 11 |
| Small LOCA in Cold Leg; No EFW (Ref 6) | | | | |
| N/A | Core Uncovery | 2250 | 0 | 0 |
| N/A | Just Before Vessel Failure | 2250 | 91 | 9 |
| N/A | After Vessel Failure | 15 | 89 | 11 |
| Large LOCA in Cold Leg; No ECC (Ref 6) | | | | |
| N/A | Core Uncovery | 20 | 0 | 0 |
| N/A | Just Before Vessel Failure | 20 | 39 | 61 |
| N/A | After Vessel Failure | 20 | 38 | 62 |
| Large LOCA in Hot Leg; No ECC (Ref 6) | | | | |
| N/A | Core Uncovery | 20 | 0 | 0 |
| N/A | Just Before Vessel Failure | 20 | 29 | 71 |
| N/A | After Vessel Failure | 20 | 28 | 72 |

| | | | | |
|---|---|---|---|---|
| Large LOCA in Cold Leg; No ECC (Ref 7) | | | | |
| N/A | Core Uncovery | 20 | 0 | 0 |
| N/A | Just Before Vessel Failure | 20 | 25 | 75 |
| N/A | After Vessel Failure | 20 | 25 | 75 |
| Note: Analysis was for Zion-like plant using MAAP4 | | | | |
| Loss of All Feedwater; No ECC (Ref 7) | | | | |
| N/A | Core Uncovery | 2250 | 0 | 0 |
| N/A | Just Before Vessel Failure | 2250 | 93 | 7 |
| N/A | After Vessel Failure | 15 | 92 | 8 |
| Note: Analysis was for Zion-like plant using MAAP4 | | | | |

Ref 1. Ringhals Unit 3 Severe Accident Analyses to Support Development of Severe Accident Procedures; WCAP-11607; Westinghouse Electric Corp.; September 1987.
Ref 2. Ringhals Unit 3 Severe Accident Analyses to Support Development of Severe Accident Procedures; WCAP-11607, Addendum 1; Westinghouse Electric Corp.; February 1988.
Ref 3. Unpublished MAAP4 Analyses for D. C. Cook SAMG Drill; Westinghouse Electric Co.; January 1999.
Ref 4. Controlling Phenomena in PWR Blackout Sequences; M. G. Plys, et. al., International Symposium on Severe Accidents in Nuclear Power Plants, Sorrento, Italy; March 1988.
Ref 5. Creep Rupture Failure of Primary Coolant Piping Prior to Reactor Vessel Failure for Severe Accidents. WCAP-11910; Westinghouse Electric Co.; July 1988.
Ref 6. Vogtle IPE Source Term Notebook; Fauske and Assoc.; September, 1992.
Ref 7. Westinghouse Owners Group Core Damage Assessment Guidance, WCAP-14696, Westinghouse Electric Corp.; July 1996.

Westinghouse Owners Group

cc:

Mr. Nicholas Liparulo, Manager
Regulatory and Engineering Networks
Westinghouse Electric Corporation
Mail Stop ECE 4-15
P.O. Box 355
Pittsburgh, PA 15230-0355

Mr. Andrew Drake, Project Manager
Westinghouse Owners Group
Westinghouse Electric Corporation
Mail Stop ECE 5-16
P.O. Box 355
Pittsburgh, PA 15230-0355

Jack Bastin, Director
Westinghouse Electric Corporation
11921 Rockville Pike, Suite 107
Rockville, MD 20852

Mr. Hank Sepp, Manager
Regulatory and Licensing Engineering
Westinghouse Electric Corporation
PO Box 355
Pittsburgh, PA 15230-0355