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Department of Energy
Washington, DC 20545

'86 JAN 23 A11:44 January 8, 1986

Mr. Robert B. Minogue, Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Minogue:

The U.S. Department of Energy's letter from James W. Vaughan, Jr., to Mr. Samuel J. Chilk, dated January 7, 1986, regarding the Nuclear Regulatory Commission's (NRC) draft report NUREG-0956 entitled "Reassessment of the Technical Bases for Estimating Source Terms" expressed our view that the the NRC should more accurately portray the significant progress made by others in the source term area. Furthermore, in order to issue NUREG-0956 in a timely manner, we suggested including qualifying statements in the text of the report that explicitly point out where the NRC methods or data may not represent the latest technical information or do not reflect a consensus of the best scientific views. The enclosed comments are intended to help the NRC improve the report in this regard.

If you should have any questions, do not hesitate to contact my office on 353-3456.

Sincerely,

David J. McGoff

David J. McGoff, Director
Office of LWR Safety and Technology
Office of Reactor Deployment
Office of Nuclear Energy

Enclosure

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ENCLOSURE

U.S. Department of Energy Comments Pertaining
to NUREG-0956, "Reassessment of the Technical Bases
for Estimating Source Terms"

Introduction

The draft report NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms," was submitted to a number of knowledgeable organizations/individuals for review in the Department of Energy (DOE) or who are under contract to the DOE. The reviewers were people with expertise in the source term area but not currently involved in the Nuclear Regulatory Commission (NRC) research program.

This enclosure represents a synthesis of the reviewers' comments and are organized in the following manner:

I. General Comments
II. Specific Technical Comments

I. General Comments

1. NUREG-0956 represents a major improvement in documenting the NRC's technical basis for estimating source terms. It is well organized and well written. However, the content of the report suffers from several limitations that would have made the report more pertinent. These limitations are listed below.

   a. In light of the DOE's viewpoint that the principal use of NUREG-0956 should be to establish the basis for enabling the NRC to approve the industry-generated codes, the report should address what procedures the NRC will follow to require industry to perform code and model comparisons. In turn, the NRC can confirm or require modifications to the analytical methods proposed for use by the industry in response to the August 8, 1985, severe accident policy statement.

   b. The Battelle Memorial Institute (BMI)-2104 calculations, on which the report is based, contain many nonmechanistic assumptions and omissions of known phenomena. Taken together, this results in calculated source terms higher than would be expected on the basis of best available scientific evidence and theory. The report should stress this fact and prepare the way for eliminating unrealistic regulatory assumptions through the NRC/Industry Degraded Core Rulemaking (IDCOR) interaction process. Specific nonmechanistic assumptions or omissions which should be referred to are as follows:

      o The use of an arbitrary containment failure time. The improved containment failure mode and ultimate pressure capability assumptions developed by the NRC's containment working groups should be incorporated.

- o The use of outdated models by the NRC for core slump, vessel failure, core-water interactions, hydrogen generation, and hydrogen burns. The analyses for ice condenser and Boiling Water Reactor (BWR) Mark I plants are unrealistic and are significantly affected by the use of less than best available models.

- o Lack of a sufficient auxiliary building performance model results in unrealistically high source terms in some sequences, particularly the "V" sequence in Pressurized Water Reactors and sequences in BWR Mark I's that involve suppression pool bypass.

- o The omission of operator action in sequences, particularly for suppression pool venting in BWR Mark I's. Operator action is very important and must be incorporated into any assessment of source terms and plant risk.

2. While there is some recognition in the report that the analysis of the BWR Mark I had some limitations, these "limitations" have resulted in an unrealistic assessment of this BWR product line. Specific deficiencies pertain to the modeling of accident sequences and the incorporation of auxiliary buildings within the models. The credibility of the BWR Mark I analysis must be questioned until, at least, better analyses which do not have these deficiencies are performed. The NRC should qualify the BWR Mark I analyses more explicitly.

3. The report gives the impression that source terms for many sequences are not lower than WASH-1400. The report should emphasize: 1) that the majority of the sequences do have lower source terms; 2) that some sequences with high source terms have an extremely low probability of occurrence; and 3) that even for these low probability sequences the high source terms are due to ultra-conservatism in both modeling and sequence definitions. A few sequences with high source terms in NUREG-0956 result from early containment failures, which, we believe, result from ultra-conservative modeling in the BMI-2104 analyses.

   There is a general lesson to be learned from these analyses. Specifically, when a calculation starts with overconservatism when there is evidence to include better treatments of various phenomena, the calculation misinforms. Regulatory decisions should not be based on the lowest probability events which have overly conservative inputs. The NRC should, as a matter of practice, use best estimate data and models within the likely range of uncertainty in the future.

4. The general discussions in the report regarding uncertainties leave much to be desired. It is not clear whether the uncertainties relate solely to models used in the BMI-2104 analyses or relate to the fact that numerous processes have been identified (e.g., direct heating, mechanistic determination of containment leak area, etc.) that are not modeled in the present analyses. The report should clarify the uncertainty discussion in this regard.

We presume the NRC staff wants both uncertainty analyses and sensitivity analyses. Uncertainty analyses (the determination of the effects of plausible variations in physical processes and parameters) are not well described in NUREG-0956. Whereas, sensitivity analyses (a determination of the effects of variations in models or conditions beyond physically realistic ranges for the purpose of identifying the relative importance of certain assumptions), as presented in earlier NRC sponsored studies, have been well described.

5. The Source Term Code Package is said to be designed to provide "best-estimate" results. While there may be no intentional bias, NRC contractors have tended to select "conservative" input parameters that give answers on the high side of the likely range of variation. For example, BMI's choice of containment failure pressures tend to be conservative. It is not likely that BMI-2104 underestimated releases as opposed to overestimated releases for the sequences covered. We do not agree that the source terms calculated from the BMI-2104 code suite represent the "best-estimate source terms." BMI-2104 represents extremely conservative source terms. This is a major shortcoming of the NRC analysis to date and should be acknowledged in NUREG-0956.

6. The statement that "the BMI-2104 analysis procedure is in place, and the results from the accident scenarios analyzed show that the procedure is applicable to other plant types" implies that the BMI-2104 procedure is correct and is all that will and should be used. This diminishes the importance of significant advances made in other programs and does not recognize that there are major deficiencies in the BMI-2104 suite of codes that are not found in industry's codes. The NRC should correct this misleading statement. Further, the NRC should clarify its intent, or lack thereof, to adopt technology developed by others to correct these deficiencies for future use. The DOE believes the best approach is for the NRC to focus on industry's analysis procedure rather than fixing the BMI-2104 code suite.

7. The treatment of the hydrogen-burn issue is weak. Hydrogen burn control is important, not only in its effect on the release rate of fission products, but also due to the fact that uncontrolled hydrogen combustion can lead to a breach of the containment. Since the 1979 Three Mile Island Unit-2 (TMI-2) accident, the NRC mounted a large-scale experimental program to resolve hydrogen combustion problems and their effects on the integrity of Light Water Reactor containments. Little mention is made of this fact and how any data from that program is being factored into the source term program. Instead, there is a reference to future experiments to determine the effect of a hydrogen burn on the release of fission products, which from reading the report appears not to be a significant issue. This treatment should be restated to acknowledge work already performed.

8. <u>NRC Conclusion 3</u>: Remaining areas of uncertainty have been identified in the new source term analytical procedures and indicate areas of research that should be pursued. Uncertainties persist in some of the areas where major advances have already been made.

Comment: The need for additional research can most effectively be demonstrated by an explicit identification and discussion of the regulations that are expected to be impacted by the reassessment and a determination made of the level of accuracy necessary for regulatory change.

9. NRC Conclusion 10: A comparative risk appraisal for the Surry plant using the Reactor Safety Study accident frequencies, source terms based on BMI-2104 results, and a preliminary reevaluation of the behavior of the containment shows a reduction in the estimated risk compared with the Reactor Safety Study. The reduction results about equally from new source terms and new evaluations of containment behavior.

   Comment: While NUREG-0956 does indicate a substantial reduction of risk for Surry in relation to the Reactor Safety Study, even greater reductions would be anticipated if containment failure phenomena were properly calculated and the V sequence analyzed properly. While there is some discussion in NUREG-0956 of risk in the Peach Bottom plant, we do not believe the NRC BWR computations are sufficiently advanced for a meaningful prediction of risk. We recommend that if risk assessments are to be published in NUREG-0956, then the BWR assessments need to be improved.

10. Page 8-7 NRC Recommendation 2: This recommendation discusses the anticipated use of the Source Term Code Package. One of the uses stated is to "...audit licensees submittals that may be based on other analytical procedures." The DOE believes this application of the code is inappropriate. The NRC should only use the code package to complete the risk rebaselining effort and serve as a basis for reaching resolution on the approval of the industry code package. Subsequent NRC review of licensee submittals should then be based on the use of NRC approved IDCOR methodology.

II. Specific Comments

1. Executive Summary

   a. Page xvii, paragraph 3. The absence of a BWR Mark-II analysis in the NUREG-0956 effort is notable. The report should indicate how Mark-II's will be addressed in the future.

   b. Page xx, Table ES., #6. Although these areas have been improved relative to WASH-1400, the NRC methodology given in NUREG-0956 does not represent the present state of the art.

2. Chapter 1 - Status and Applications

   a. Page 1-3, paragraph 6. Groups other than BMI and IDCOR have been involved. The Stone and Webster Engineering Corporation, the Electric Power Research Institute, the New York Power Authority, the American Nuclear Society (ANS), and foreign groups should be acknowledged. The effort is international.

b. Page 1-4, paragraph 2. BMI-2104 is not the most recent quantification. The process is worldwide and continuous.

c. Work reflected in the Report of the Kemeny Commission, the Rogovin Report, the Report of the ANS Special Committee on Source Terms should be referenced. The impetus for the reconsideration of source terms came first from industry and should be acknowledged.

3. Chapter 2 - Historical Perspective

a. Page 2-7, paragraph 2. An important weakness in WASH-1400 that should be cited is the prominence given in-vessel steam explosions leading to early containment failure.

b. Page 2-10, paragraph 6. The phrase, "cover other initiating events as an envelope," is a very useful generalization. The same philosophy was used in the ANS Source Term Study. The NUREG-0956 should not ignore this possibility for generalization in the conclusions.

c. Page 2-13, paragraph 6. The NRC-Sandia "uncertainty" study is, in fact, a limited sensitivity study.

4. Chapter 3 - Computer Models

a. Core Melt and Aerosol Generation

1) Page 3-9, paragraph 4. A major deficiency in MARCH is the meltdown model. The model is inadequate by limiting core temperatures to 2227 C. Experimental evidence obtained from the Loss-of-Fluid Test Facility and TMI-2 demonstrate this fact. This undermines the credibility of any conclusions reached through such analyses except to say the meltdown stage is conservatively represented. This deficiency should be noted.

2) Page 3-9 (middle). There should be a discussion regarding the apparent inconsistency between the NRC's stated approach of calculating "best estimates" and the assumption in the BMI-2104 MARCH calculations that the entire core and lower internals "fall coherently from the vessel upon failure." Furthermore, there should be a description and rationalization for the assumed vessel failure mode.

3) Page 3-14. In the first paragraph under "Status of the CORCON code" it is indicated that the viscosity model "results in unreasonably high temperatures under certain conditions." However, this problem is dismissed in the next sentence, "...a check of the particular calculations...revealed that overall results were fortunately not affected in a major way by the use of CORCON-Mod 1". This conclusion seems somewhat inconsistent or perhaps incomplete when compared with the statement on page 3-24; "...the uncertainty in melt temperature, which is derived from CORCON...is believed to be the most important related

uncertainty because vaporization and chemical reactions depend exponentially on temperature." We concur that the melt temperature derived in CORCON is important to the uncertainty determinations; thus, the report should reconcile these inconsistencies.

4) Page 3-15, paragraph 5. The uncertainty mentioned in the ORIGEN calculations is unduly pessimistic.

5) Page 3-24. The relative humidity in the containment atmosphere affects the water uptake by aerosol, but is not known accurately. There is disagreement between code analysts, primarily due to different assumptions of decay heat sources. This should be reflected in NUREG-0956.

6) Page 3-32. Conclusion 7 mentions, "For [containment] failure at late time, the amount [of radionuclides] leaked can be much larger than the amount suspended if the hole is large enough to cause resuspension." Resuspension phenomena are not modeled in NAUA. The NRC should describe how they arrived at this conclusion. The IDCOR evaluations indicate that resuspension will not occur.

7) Page 3-39, paragraph 3. It is stated that zirconium oxidation is important for at least three reasons. A fourth reason is tellurium holdup.

8) Page 3-40. The modeling of low-volatile fission product release during core-concrete interaction is recognized as a unassessed model, but the "5-to-10 times-higher" (than WASH-1400) release estimates in NUREG-0956 are used with apparently the same confidence as the high-volatile releases. This is inappropriate and should be qualified.

b. Transport in Reactor Cooling System

1) Page 3-10, paragraph 1. Properties of the gas mixture (H , steam) should be used in the future, rather than those of pure steam, as was done, at least in early versions of MARCH-2 and MERGE.

2) Page 3-10, paragraph 3. MARCH-2 and MERGE do not calculate the pressure drop through the emergency core cooling line properly. This significantly affects the velocity and aerosol behavior. This item should be added to the two listed phenomena.

3) Page 3-21, Section 3.2.3. Interaction between the containment thermal hydraulics and primary system fission products does not appear to be modeled dynamically in the suite of codes. We are not aware that this coupling is being addressed in the current improved versions. The IDCOR has demonstrated that for certain reactor types and accident sequences, the heat loss from the vessel to the containment can directly control fission product revaporization within the primary system. Containment

temperature histories feed back on this phenomena and cannot be ignored. This should be reflected in NUREG-0956.

4) Page 3-22, paragraph 2. There is a discrepancy between deposition velocities measured in the Severe Fuel Damage (SFD) tests and those noted in the second paragraph. In TRAP-MELT the irreversible sorbtion deposition velocities were based on quick-look experiments performed at Sandia. The deposition velocities in the SFD do not agree with these values. We believe the deposition velocities measured in the SFD tests are more prototypical.

c. Transport in Containment

1) Page 3-24, Section 3.2.5. Deposition in containment leakage paths has been ignored. In some containment failure modes, such deposition can reduce the release fraction significantly. Even if deposition is minimal, the particle size may be increased by agglomeration or resuspension, leading to more rapid fallout downstream. Although difficult to quantify, some reference to this potentially important phenomenon should be made.

2) Page 3-24, fourth full paragraph. Steam condensation and/or evaporation on aerosol particles has not been well characterized, especially as affected by the presence of hydroscopic constituents in agglomerated particles. Liquid water associated with aerosol particles affects aerosol removal significantly. This can be of particular importance for BWR Mark I reactor building attenuation and should be noted in NUREG-0956. Further, specific plant assessments should not ignore this phenomena.

3) Calculations performed in the BMI-2104 analyses assumed 100 percent of the core exiting the vessel at the time of vessel breach. Less molten material present at the time of core-concrete interactions would tend to lower the thermal energy present in the melt. The NRC should state this is an overconservative assumption that should not be used in future analyses.

d. Containment Failure

1) It is well known that the timing and mode of containment failure (if any) is extremely important to source terms. However, none of the BMI-2104 codes predict containment failure; it is an input to the codes. This should be stated in the report.

e. Uncertainties

1) Page 3-29, Item 2. The importance of an uncertainty (a factor of 100 is mentioned) depends on the base case estimate. This should be stated.

2) Page 3-38, Table 3.5. Another major uncertainty that should be mentioned is the set of input assumptions made by the user of the code.

5. Chapter 4 - Accident Sequences

a. Page 4-1, paragraph 3. It is alleged that a Mark II containment system would be comparable in its chemical and physical environment during an accident sequence to a Mark I containment system plant. This is not the case based on available design information, IDCOR-NRC interactions, Stone & Webster work on Shoreham, and IDCOR work on plant assessments other than those addressed in NUREG-0956.

Core debris distribution after vessel failure in a Mark I containment is predominantly limited to the drywell and pedestal floor regions. Mark II containments vary widely in design, yet have the potential for core debris draining into the suppression pool. The accident progression in a containment with core debris deposited in the suppression pool will be much different than an accident in which all of the debris remains on the drywell floor. NUREG-0956 should reflect this difference.

b. Major differences exist between the IDCOR and NRC in-vessel hydrogen generation models when cladding material begins to melt. After the core begins to degrade, IDCOR tracks the accumulation of molten material and calculates whether or not sufficient steam and hydrogen flow exist to levitate the molten mass. Once the molten material can no longer be supported, it is allowed to move into lower portions of the core where it has the potential to freeze and create a local blockage. This blockage will then serve to divert steam flow to other regions of the core. The BMI-2104 analyses do not model this core blockage and the result is an overestimate of hydrogen production. We believe the IDCOR treatment is more realistic based on SFD experiments and TMI-2 evaluations made to date. Further, this significantly affects the magnitude of the source term in Table 4.13, page 4-41, for Peach Bottom. This other view should be reflected in NUREG-0956.

Specifically, the results of the SFD experiments at Idaho indicate the formation of a eutectic at temperatures in the range of 2500 K. These experiments also show movement of molten clad and the formation of very substantial blockages. Where bypasses were available in these experiments, the steam generated in the core had the potential to be diverted and therefore did not participate in any further clad oxidation. The effect of this steam diversion can be seen in the total hydrogen generated for these experiments. Simple energy balances would indicate that the core water level in the Peach Bottom AE sequence should drop below the bottom of the core as a result of the rapid depressurization. When molten core material begins to flow into the lower plenum, the remaining core would be expected to be partially blocked with the concurrent formation of eutectic material. Subsequent steam generation should not lead to further significant clad oxidation.

c. Pages 4-12 and 4-13. The upper limit of peak temperature is not realistic and has a significant effect on predictions. NUREG-0956 should address why the NRC limits peak temperature.

d. Page 4-24, Figure 4.6. It is not clear why Peach Bottom would have a larger source rate of aerosols from concrete attack than Grand Gulf. Core debris would tend to spread out over a larger area in Peach Bottom and result in less concrete attack than Grand Gulf. A better explanation of this difference between plants is required to substantiate Figure 4.6

e. Page 4-33, Figure 4-12. Figure 4-12 shows sharp pressure spikes due to global hydrogen combustion for a Grand Gulf sequence for which the igniters are available. A global hydrogen burn for Grand Gulf is not likely. Igniters are placed throughout Mark III containment buildings to allow for the controlled burning of hydrogen produced by Zircaloy oxidation. The igniters are located below the hydraulic control unit floor above the suppression pool. Hydrogen combustion at low concentrations from these igniter locations will generate partial burns as opposed to a global burn. Local burning does not threaten containment integrity and thus will not generate sharp pressure spikes.

Another important consideration is the generation of high compartment temperatures due to local hydrogen combustion. Natural circulation between the lower compartment and cavity in an ice condenser brings oxygen into a hot compartment which then reacts gradually with the hydrogen. Therefore, one would expect the concentration of hydrogen to be less than that required for a global burn. The source term implications should be clarified for NUREG-0956.

f. Page 4-36, Figure 4.14. The description of this figure should explain what is happening during the 173 minute delay from vessel head failure (corium ejection) to the start of concrete attack.

g. Page 4-37, first paragraph under 4.10. It is stated that the "amount and timing of releases of radioactive material to the environment" were determined. The timing of release in the BMI-2104 methodology was more assumed than "determined." Furthermore, the analyses of the containment working groups should have been factored into the BMI sequences. The writeup should more accurately describe what was actually done in calculating the amount and timing of releases. Use of the word "determined" should be qualified or changed to "assumed."

6. Chapter 5 - Review of Source Term Work

   a. Page 5-3. Major factors identified by the peer review group, consisting of 14 scientists, included better accounting of water and the effectiveness of auxiliary buildings. It is not clear how or whether these recommendations were followed. The NRC should clarify.

   b. Page 5-6, first full paragraph. The NRC analyzed its reference plants after IDCOR started on the reference plant analyses. Although the "answer" given by the IDCOR and NRC computer codes cannot be compared directly, the differences in models used in the two analyses are not "obscured" as stated. The key modeling differences can be and should be identified and their influence on the results determined.

7. Future Research

   a. Pages 7-2 & 3, Item 1. Natural circulation is being addressed by calculation and needs to be experimentally validated. This can be attempted through benchmark calculations based on the TMI-2 data base. The NRC should state its intent, or lack thereof, to do so.

   b. Page 7-9, sixth paragraph. It is stated that concern still exists with respect to steam explosions even though NRC's own panel of experts (Steam Explosions Review Group - November 1984) concluded that it is not a matter of concern. Steam explosions should be dropped from consideration permanently. It is a non-issue and should so be stated in NUREG-0956.

   c. Page 7-12, Section 7.2.3. The first paragraph in this section claims that the BMI-2104 computer code suite and the closely Related Source Term Code Package are examples of integrated codes. The codes are not integrated, nor do they properly address feedback between parts of the codes that reflect different, although interrelated, phenomena. This should be restated.

Idaho
National
Engineering
Laboratory

Severe Accident Research Review

Presented by:

R.R. Hobbins

January 9, 1986

EG&G Idaho, Inc.

Outline

- Current status
- Outstanding issues
- Issues being addressed by research at INEL
- Gaps in current research

NUREG-0956

- Summary of NRC severe accident methodology and current status
- Summary of outstanding issues identified by IDCOR/NRC interaction and by the APS report
- Advances over WASH-1400 methodology are clear

P208-KM168-03

NUREG-0956
(continued)

- Technical basis of current methodology not presented
- No assurance that current severe accident source term calculations are valid
- Largest outstanding problem is the validation of computer codes used in the NRC severe accident source term methodology

P208-KM168-04

Outstanding Issues

- * In-vessel fission product release and generation of aerosols
- Natural circulation in the reactor vessel
- * Core melt progression and hydrogen generation
- * Retention and revaporization of fission products in RCS
- Fission product release and aerosol generation from core-concrete interaction
- Scrubbing efficiency of suppression pools and ice compartments
- Aerosol physics in containment
- Containment pressure loads
- Containment failure modes
- Volatile iodine

P208-KM168-068

In-Vessel Fission Product Release and Generation of Aerosols

- Major needs
  - Chemical forms of fission products
  - Generation of aerosols
  - Less volatile fission products
    Ba, Sr, Ru, La
  - Chemical interactions of fission products with aerosols and structural surfaces
  - Release under high pressure conditions (15 MPa)

- Codes
  - SCDAP (FASTGRASS), MELPROG (VICTORIA)

- On-going experimental work
  - ANL (EPRI)
  - SANDIA (NRC)
  - ORNL (NRC)
  - BCL (NRC)
  - INEL (NRC)

P208-KM168-06

Core Melt Progression and Hydrogen Generation

- Major needs
  - Temperatures and mechanisms of core melt relocation
  - Melt-core structural material interactions
  - Hydrogen generation after loss of original core geometry
  - Penetration of core melt into lower plenum, vessel attack and melt ejection
  - Fission product release

- Codes
  - SCDAP, MELPROG

- On-going experimental work
  - ANL (EPRI)
  - SANDIA (NRC)
  - INEL (NRC)
  - NRU (NRC)
  - ORNL (NRC)
  - KfK

P208-KM158-07

Retention and Revaporization of Fission Products in RCS

- Major needs
  - Chemical forms for deposition
  - Chemical forms for revaporization
  - Interactions within deposit and with structural surface

- Codes
  - SCDAP/RELAP5 (TRAP-MELT)
    TRAC/MELPROG (VICTORIA)

- On-going experimental work
  - ANL (EPRI, DOE)
  - SANDIA (NRC)
  - INEL (NRC)

P209-KM168-08

CsI

- Preponderance of evidence points to CsI as the dominant chemical form of iodine under most conditions expected in severe accidents

- Iodine behaves like CsI in PBF tests and TMI-2
  - Iodine and cesium co-deposit in PBF tests
  - Iodine and cesium are distributed similarly in the TMI-2 reactor systems
  - CsI has been tentatively identified in PBF SFD 1-3 with optical fluorescence

- Only very small quantities of volatile iodine have been observed in the LOFT FP-2 test, the PBF tests, and TMI-2

P206-KM188-10

CsI
(continued)

- Recent tests at Sandia showing disproportionation of CsI under a radiation field need to be confirmed and additional tests need to be performed under conditions more typical of severe accidents

- Sandia experiments used CsI ratio of 1 and radiation field of 20 Rad/s at 1.2 MEV

- More representative conditions are CsI ratio of 10 and radiation field of 10^6 Rad/s at 0.7 MEV

P208-KM188-13

Gaps in Current Research

- Release of fission products during melt progression, vessel meltthrough, and core-concrete interaction
- Release of fission products from fuel at high system pressure
- Validation of aerosol generation and behavior models at high pressure
- Fission product-aerosol chemical interactions
- Revolatilization of deposited fission products
- Effect of radiation on fission product chemical forms under high fields and high pressures

P206-KM168-11A

Gaps in Code Development and Application

- Development of SCDAP/RELAP5
  - Complete assessment (Marviken, TMI, etc.)
  - Introduce fission product and aerosol chemistry
  - Include debris model
  - Automated links with MELPROG, CONTAIN & CRAC
  - Enhance usability

- Application of SCDAP/RELAP5
  - Perform test analyses (ACRR-ST, ORNL FPR, NRU)
  - Develop plant specific code decks
  - Benchmark risk codes
  - Use for Accident Management studies

P209-KM158-14