COMBUSTION ENGINEERING

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(Project 675)

Mr. Frank J. Miraglia
Associate Director of Projects
Office of Nuclear Reactor Regulation
Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Advanced Reactor Severe Accident Program - Topic Paper Set 2

Reference: Letter, LD-87-067, A. E. Scherer (C-E) to F. J. Miraglia (NRC), dated November 24, 1987

Dear Mr. Miraglia:

In the reference letter, Combustion Engineering submitted modified resolutions to four (4) NRC/IDCOR issues from Topic Set 1. Those four resolutions, combined with six (6) NRC/IDCOR issue resolutions which were already agreed upon, form the PWR-applicable portion of ARSAP Topic Set 1. The purpose of this transmittal is to provide the proposed resolutions for four of the six issues which make up Topic Paper Set 2.

The four issues which are being submitted are:

- o In-Vessel Hydrogen Generation (IDCOR Issue 5)
- o Core Melt Progression and Vessel Failure (IDCOR Issue 6)
- o Containment Performance (IDCOR Issue 15)
- o Hydrogen Ignition and Burning (IDCOR Issue 17)

The remaining two issues of Topic Paper Set 2:

- o Direct Containment Heating by Ejected Core Materials (IDCOR Issue 8)
- o Debris Coolability (IDCOR Issue 10)

will be transmitted as a separate sub-set in approximately 30 days.

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Combustion Engineering plans to adopt, in development of the System 80+[TM] design, the ten (10) NRC/IDCOR resolutions identified in the Reference, and the four (4) resolutions proposed by ARSAP which are attached to this letter. We request your early concurrence on their acceptability.

If you have any questions or comments on the attached material, please feel free to call me or Dr. M. D. Green of my staff at (203) 285-5204.

Very truly yours,

COMBUSTION ENGINEERING, INC.

A. E. Scherer
Director
Nuclear Licensing

AES:ss
Attachment: As Noted

DOE Advanced Reactor Severe Accident Program*

ARSAP Proposed Resolutions for Severe Accident Issues - TOPIC SET 2

*The material in this attachment was developed by the DOE ARSAP in support of C-E's Design Certification Program

SEVERE ACCIDENT ISSUE TOPIC PAPER

2.1 IN-VESSEL HYDROGEN GENERATION (IDCOR ISSUE 5)

Issue Definition

Reaction of the zirconium cladding (30,000 kg in an advanced PWR core) with steam in the reactor vessel during a severe accident is a major source of hydrogen that could subsequently undergo combustion in the containment. Containment pressurization due to hydrogen combustion is a major concern in severe accident risk assessment because of its potential for causing early containment failure. The IDCOR/NRC issue resolution process identified in-vessel hydrogen production as an issue on which significant differences remained between the IDCOR and NRC approach.

The issue of hydrogen generation is intimately connected with that of hydrogen combustion in containment. Although the containment will be designed according to the design basis, severe accident analyses will be performed to establish margin to failure. In this context the amount of hydrogen generated and the acceptable concentration of hydrogen in containment, that is, a concentration low enough to preclude detonation or deflagration, sets a lower limit on containment free volume. If sufficient containment free volume cannot be achieved without difficulty, hydrogen control measures, such as igniters, could be necessary. (Igniters may also be needed to prevent local detonations.) Thus, it is of importance to plant economics, as well as safety, that the amount of hydrogen generated in the event of a severe accident be less than the limit imposed by the containment design.

The issue of hydrogen generation is also directly affected by that of in-vessel core melt progression. While the kinetics of oxidation of zirconium by water vapor are fairly well known for intact core geometries, the reaction rates for a severely degraded core are uncertain. This is because of several competing effects; some can enhance the reaction rates and others can retard and even effectively stop oxidation in severely degraded core locations. The uncertainties associated with these effects are clearly recognized by the NRC:

> "The uncertainties in hydrogen generation become large with the onset of Zircaloy melting, relocation, and fuel dissolution, and the loss of the initial intact and well-characterized core geometry. There are significant uncertainties involving (1) the effects of the relocation of the molten unoxidized metallic Zircaloy accompanied by the dissolution (liquefaction) of some of the UO_2 fuel; (2) the surface area available for further oxidation of the relocated Zircaloy and questions about the presence of oxidation-limiting ZrO_2 films on the surface of the relocating molten cladding; (3) questions of steamflow blockage ... or diversion ... by the relocated material; and (4) hydrogen generation following slumping of the melt into the water in the lower plenum."[1]

Significant differences exist between the NRC staff and IDCOR on the details of the analytical models used to represent the phenomena affecting hydrogen generation, particularly with respect to the effect of flow blockages on inhibiting hydrogen production.[2] However, the central issue for advanced PWRs is the appropriateness of setting design criteria for a PWR containment, as embodied in the EPRI ALWR Requirements Document,[3] such that hydrogen concentration remains below 13% for an amount of hydrogen equivalent to oxidation of 75% of the Zirconium cladding in the active core. The 75% value encompasses hydrogen generated from other sources during in-vessel core melt progression. Because of debris cooling in the reactor cavity, hydrogen generation from core debris after reactor vessel meltthrough is not judged to be significant. The preponderance of analyses and experimental evidence indicates that 75% oxidation of the active core Zirconium cladding is a conservative assumption for hydrogen generation in severe accident analysis.

Historical Perspective

Industry Actions to Address the Issue

The industry, through IDCOR, has examined available experimental results (SFD, LOFT LP-FP-2, and TMI-2), including the hydrogen production values. In order to obtain agreement with observed data, IDCOR has incorporated first-order flow blockage models into MAAP.[4] Benchmarking of the PBF-SFD tests was performed as part of the IDCOR/NRC issue resolution effort[5] and was presented in the open literature.[6] Similarly, the accident in TMI-2 was simulated with MAAP 2.0;[7] this simulation is currently being updated with MAAP 3B as part of the TMI-2 standard problem exercise.[8] The latter activity is intended to carry out the simulation to 300 minutes after the onset of the accident, beyond the time after the core was, for the most part, cooled. Thus, the effects on hydrogen generation of coolant injection on hot debris will be modeled. This activity is expected to be concluded in early 1988.

Another important benchmarking activity was carried out on the LOFT LP-FP-2 experiment.[9] This activity covered the transient phase of the accident prior to core reflood. Efforts are currently underway to extend the calculation until after the core reflood. The PBF-SFD experiment at Idaho National Engineering Laboratory (INEL) (see Reference 6) did not allow sufficient bypass flow and steam diversion to be prototypic of reactor configurations.[10] Therefore, its results are considered only as a qualitative indication of the occurrence of the then blockage phenomena. Although significant hydrogen was produced (approximately 75%), retardation of hydrogen generation did occur in the degraded fuel. Analyses of LOFT LP-FP-2 and TMI-2 support the industry conclusion regarding limitation of hydrogen production to that equivalent to 75% of the zirconium cladding in the active core region.

NRC Action to Address the Issue

The NRC has expended significant efforts on the hydrogen production issue; these efforts encompassed major experimental and analytical programs and have been supplemented by work in Canada and Germany. Experimental programs include the PBF-SFD tests at the INEL (see Reference 6), the ACRR tests at Sandia,[11] the NRU tests in Canada,[12] and LOFT LP-FP-2 (see Reference 9). Analyses have been performed with the MARCH 2.0 code[13] for BMI 2104[14], the updated MARCH 3.0 in the Source Term Code Package,[15] SCDAP,[16] and MELPROG.[17] These activities are summarized in Appendix J2 of NUREG 1150 (see Reference 1). Subsequently, a summary of internationally sponsored research programs in the United States on core melt progression, hydrogen production, and fission product release was presented in Reference 18.

The NRC Position

The NRC position is summarized in references 1 and 2. Relevant excerpts are given below. The staff summary paper on the NRC/IDCOR Issue 5 (see Reference 2) indicates the following:

> "On the basis of the MAAP and SCDAP code comparisons with PBF tests, the PBF post-test examinations, and the limited code comparisons with TMI-2, the staff concludes that the IDCOR assumption that complete channel blockage occurs following cladding/fuel relocation has not yet been adequately substantiated. The staff recognizes that the formation of significant blockages during relocation is very likely; this is evidenced by the results of several of the PBF tests and also appears to be supported by examinations of the central region of the TMI-2 core. Considerable uncertainty remains, however, concerning the degree of blockage, flow patterns around and above the blockage, and the extent of blockage effects on hydrogen generation. In this regard, the staff continues to believe that models which allow oxidation to continue in degraded fuel channels following cladding/fuel relocation (such as the models used in MARCH and MELPROG) would provide more realistic estimates of in-vessel hydrogen production.

"Additional comparisons between, MARCH, MELPROG, and MAAP code results provide some insight into the large uncertainties inherent in the modeling of in-vessel phenomena. In-vessel hydrogen production estimates predicted by MARCH 2 and an earlier version of MAAP were compared by [Battelle Columbus Laboratories] BCL[a] and were found to differ by a factor of 2 or more, with MARCH predicting the larger values. More recent MELPROG calculations for a station blackout sequence at Surry[a] resulted in total in-vessel hydrogen production comparable to that predicted by MAAP when a relocation temperature of 2200^{o}K was assumed in MELPROG. However, later MELPROG calculations in which cladding/fuel relocation was assumed to occur at 2500^{o}K instead of 2200^{o}K, resulted in total hydrogen production approximately twice that in the original calculation, i.e., comparable to that predicted by MARCH.[a] The relocation temperature referred to in the cited MELPROG sensitivity analysis calculations, it is very important to note, is a molten Zircaloy (and dissolved UO_2) relocation temperature and not a molten fuel and corium slumping temperature. Relocation temperature is just one of several parameters that are considered to contain large uncertainties.

"BCL[a] concluded that while some of the predicted differences between MARCH and MAAP results are due to modeling differences between the two codes, many are due to user selected input or model parameters. In BCL's view, given the present state of knowledge, both the IDCOR and NRC modeling approaches must be considered as plausible.

"The staff believes that actually, neither the MARCH nor the MAAP treatment is completely in accord with our current knowledge of the governing physical processes. In the early rod-geometry phase, MARCH is extremely simplistic in its treatment of Zircaloy relocation, whereas MAAP is inconsistent with existing PBF data in assuming that hydrogen generation is cut off in the initial

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[a] See Reference 2 for original references

stages of this molten Zircaloy relocation by blockage formation. In the later stage of molten corium slumping into the lower plenum water, MARCH, by a parametric treatment (particle size and fraction of unavailable Zircaloy) allows for steam generation/debris cooling and oxidation with hydrogen generation that ranges from essentially zero to 100 percent of the unoxidized Zircaloy in the slumped corium. MAAP, on the other hand, assumes essentially no interaction between the molten debris and water in the lower plenum and consequently no hydrogen generation from molten corium slumping. While the MELPROG code represents a marked advance in modeling capabilities, it also is subject to large uncertainties inherent in the modeling of in-vessel phenomena. It is likely that such uncertainties will always exist to the extent that the calculation of in-vessel phenomena could not be considered precise."

"Accordingly, it is the staff's position that a range of in-vessel hydrogen production estimates, encompassing the results of MARCH and MAAP calculations, should be considered by IDCOR in establishing uncertainty bounds on risk. Such estimates should be developed through parametric variation of key input and modeling assumptions governing hydrogen production, as well as through sequence variations including recovery actions. The effect of in-vessel hydrogen production significantly greater than predicted by MAAP will also be considered as part of the uncertainty analysis performed for NUREG-1150...."

The NRC staff position is continued in NUREG 1150, Appendix J (see Reference 1):

"...The uncertainties in hydrogen generation become large with the onset of Zircaloy melting, relocation and fuel dissolution, and the loss of the initial intact and well-characterized core geometry. There are significant uncertainties involving (1) the effects of the relocation of the molten unoxidized metallic Zircaloy accompanied by the dissolution (liquefaction) of some of the UO_2 fuel; (2) the surface area available for further oxidation of the relocated Zircaloy and questions about the presence of oxidation-limiting ZrO_2 films on the surface of the relocating molten cladding; (3) questions of steamflow blockages (BWR) or diversion (PWR) by the relocated material; and (4) hydrogen generation following slumping of the melt into the water in the lower plenum. As indicated previously, these effects are treated in MARCH and in the original version of MAAP as input parameters, with the relocation of all the material, including the Zircaloy, occurring at a single, assumed core-slump. (A later version of MAAP has a separate Zircaloy relocation model.) MAAP puts strong emphasis on steamflow blockage (BWR) and flow diversion in the open-lattice PWR core to significantly reduce hydrogen generation. MARCH has a user option for high-surface area oxidation of the unoxidized molten Zircaloy following core slump into the lower-plenum water that, based on the QUEST uncertainty study, could increase the total hydrogen generation by about 40 percent.[b] The Zircaloy relocation and continued oxidation are mechanistically modeled in SCDAP and MELPROG, and hydrogen and steam generation following core slump are being modeled mechanistically in MELPROG. There are few data currently available to support this mechanistic modeling, however.

"...It has been observed experimentally (PBF, ACRR, KfK) that downward relocation of molten unoxidized Zircaloy limits the autocatalytic oxidation temperature rise and the hydrogen generation by removing unoxidized Zircaloy from the hotter regions of the core. (This unoxidized Zircaloy, however, may become available for oxidation and hydrogen generation later in the accident, either in-vessel or ex-vessel.) Calculations (MELPROG, for example) have shown that increasing an assumed

[b] See Reference 1 for original references.

temperature threshold for relocation of molten Zircaloy from 2,200^oK to 2,500^oK can increase the hydrogen generation by as much as a factor of two. Thus, the MARCH modeling that allows no relocation of unoxidized Zircaloy before an assumed slump of a core region at 2,550^oK can give substantially greater hydrogen release in the autocatalytic oxidation transient than actually occurs.

"Three MELPROG calculations were made in the analysis of the Surry TMLB sequence. Two were with the one-dimensionsal version of MELPROG and one with the new two-dimensional version. The calculated conditions at vessel failure are shown in Table 1[b] along with the results of Surry TMLB calculations with MARCH 2.0 from BMI-2104.[b] The important Zircaloy relocation temperature (assumed input) was varied in the two MELPROG one-dimensional calculations, and modeling of the downcomer water was included in the second calculation at the higher relocation temperature.

"As seen from the results in Table 1, the assumed Zircaloy relocation temperature has a major effect upon the fraction of Zircaloy oxidized (and the hydrogen generation) as well as upon the core debris average temperature and melt fraction (the fraction of the debris molten) at vessel failure. At similar (high) assumed Zircaloy relocation temperature thresholds, MARCH 2.0 and MELPROG 1-D gave similar results on the fraction of the Zircaloy oxidized (hydrogen generation), but MELPROG gave a somewhat higher average debris temperature. Increasing the Zircaloy relocation temperature threshold from 2,200^oK to 2,500^oK increased the oxidized Zircaloy (hydrogen generation) by a factor of two.

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b See Reference 1 for original references.

TABLE 1. CONDITIONS AT VESSEL FAILURE (From Reference 1)

	MARCH (1-D)	MELPROG1-D Case 1	MELPROG1-D Case 2	MELPROG 2-D
$T_{relocate}$ (^oK)	2,550**	2,200**	2,500**	2,200**
Time from Saturation to Vessel Failure (min)	90	115	105	157
Zr Oxidized (%)	59%	31%	60%	40%
Hydrogen Mass (kg)	430	230	440	300
T_{mean} (^oK)	2,380	2,600	2,120	2,400
Debris Melt Fraction (%)	***	34%	16%	30%
Debris Zr Mass (kg)	6,770	11,400	6,500	9,000
Debris Steel Mass (kg)	35,000	900	10,300	19,000

**Parameter Assumed.

***Not modeled in MARCH.

Technical Approach to Resolve the Issue for ALWRs

As can be seen from the NRC discussion given above, significant differences exist between the NRC and IDCOR on the modeling of in-vessel hydrogen production. Much of this disagreement revolves around the blockage model that was invoked in MAAP to reconcile the analytical results with experimental data; generally, without this adjustment, predicted hydrogen production values are higher than observed values. However, for the purpose of setting criteria for the ALWR, it is not necessary to resolve the blockage issue. The predominance of experimental data has indicated significantly less hydrogen production than that associated with oxidation of 75% of the zirconium cladding in the active core. Also, both IDCOR and NRC analyses predict less than this value even without invoking blockage; for example, see Table 1, given above, for NRC-calculated values using MARCH and MELPROG.

The technical approach for resolution of the issue of hydrogen generation for advanced PWRs is (1) to comply with containment design requirements for control of hydrogen ignition and burning as specified in the EPRI ALWR Requirements Document, (2) to provide technical justification for the assumption limiting total hydrogen generation to the equivalent produced by 75% oxidation of the Zirconium cladding in the active core, and (3) to incorporate modeling improvements in MAAP for phenomena associated with in-vessel hydrogen production. The specific items in the technical approach are described below.

1. Electric Power Research Institute (EPRI) ALWR Requirements Document specifies that the containment be designed to withstand a burn of hydrogen equivalent to oxidation of 75% of the zirconium cladding in the active core. Best-estimate analyses of the core oxidation and of hydrogen combustion in containment will be performed to ensure that the design requirements will be met.

2. Technical justification for the assumption the the maximum amount of hydrogen generated during a severe accident is equivalent to that

generated by the complete oxidation of 75% of the zirconium cladding in the active core will based on the following:

- o Data from the NRC experiments which have produced hydrogen amounts consistently less than the 75% value. These results were achieved without accounting for possible flow blockage during a core relocation. These experiments, therefore, produced more hydrogen than would have been generated in more prototypic configurations.

- o Results of analyses with NRC codes consistently produce less than the 75% value without invoking blockage models.

- o Analyses with the MAAP code will be performed both with and without blockage to provide a range of results which envelopes the effect of flow blockage on hydrogen generation.

3. The MAAP improvements being developed by the Advanced Reactor Severe Accident Program (ARSAP) include models which describe the formation of molten Zircaloy-fuel eutectic, the relocation and refreezing of molten material, and separate treatment of control materials. The relocated material composition is determined by UO_2-ZrO_2-Zr equilibrium phase diagram. An assumed breakout temperature is used to determine mechanical breakthrough of eutectic from the oxidized clad surface. The behavior of relocated material flowing down the fuel pin and freezing is based upon mechanistic momentum and energy equations. The nodal geometry is determined by the relocation model in order to relate the effects of mass accumulation on coolant channel geometry. The model follows the location and amounts of UO_2-ZrO_2-Zr metallic eutectic from the start of material movement until the material enters the lower plenum.

References

1. U.S. Nuclear Regulatory Commission (USNRC), Reactor Risk Reference Document, NUREG-1150, App. J-O, Vol. 3, February 1987, pp. J-21 to J-2.28.

2. T. Speis, USNRC, "Summary Paper for the Resolution of NRC/IDCOR Issue 5," attachment to letter to A. Buhl, IT Corporation, March 11, 1987.

3. Electric Power Research Institute, Advanced Light Water Reactor Requirements Document, Chapter 5: Engineered Safeguards Systems, Palo Alto, California, December 1987.

4. Fauske & Associates, Inc., MAAP (3.0)- Modular Accident Analysis Program User's Manual, IDCOR Technical Report 16.2-3, Atomic Industrial Forum, February 1987.

5. Fauske & Associates, Inc., IDCOR Technical Support for Issue Resolution, IDCOR Technical Report 85.2, Atomic Industrial Forum, July 1985.

6. A. Sharon et al., "Analysis of the Steam Generation Rates in SFD Tests 1-1 and 1-3 and Their Application to Hydrogen Generation," Proc. ANS Winter Annual Meeting, San Francisco, California, November 1985.

7. M. A. Kenton, R. E. Henry, G. R. Thomas, "Simulation of the TMI-2 Accident Using MAAP," Proc. of the International ANS-ENS Meeting on Thermal Reactor Safety, San Diego, California, February 1986.

8. A. Sharon et al., Simulation of the TMI-2 Accident, Phases 1 and 2 Using MAAP 3B, Draft, FAI/87-88, Fauske and Associates, Inc., November 1987.

9. Fauske & Associates, Inc., Simulation of LOFT Experiment LP-FP-2 Using MAAP 3.0, Report 86-27, Rev. 1, Burr Ridge, Illinois, June 1987.

10. A. Sharon, "Analysis of the PBF-SFD Fuel Bundle and LWR Channel Behavior in Degraded Conditions," AIChE Symposium Series, 83, R. W. Lyczkowski, editor, 1987.

11. Reactor Safety Research Semiannual Report, January to June 1986, Volume 35, NUREG/CR-4805, Sandia National Laboratory, May 1987.

12. "FLHT -2 & -4 Test Design and Operation, NRU Full Length High Temperature Tests, Coolant Boiling and Damage Progression Program," Presentation Prepared for the USNRC SFD/ST Semi-Annual Partners Meeting, Pacific Northwest Laboratory (Battelle Memorial Institute), October 1986.

13. R. O. Wooton, P. Cybulskis, S. F. Quayle, MARCH 2 (Meltdown Accident Response Characteristics) Code Description and User's Manual, Battelle Columbus Laboratories, NUREG/CR-3988, BMI-2115, September 1984.

14. J. A. Gieseke et al., Radionuclide Release Under Specific LWR Accident Conditions, Battelle Columbus Laboratories, BMI-2104, July 1983-July 1986.

15. M. Silberberg et al., Reassessment of the Technical Bases for Estimating Source Terms, NUREG 0956, July 1986.

16. C. M. Allison, F. R. Carlson, R. H. Smith, "SCDAP: A Computer Code for Analyzing Light Water Reactor Severe Core Damage," Proceedings of the International Meeting of Light Water Reactor Severe Accident Evaluation, Cambridge, Massachusetts, August 28-September 1, 1983.

17. W. J. Camp et al., A Mechanistic Code for Analysis of Reactor Core Melt Progression and Vessel Attack Under Severe Accident Conditions, Draft, Sandia National Laboratories, SAND85-0237, (Available in the NRC Public Document Room, 1717 H Street NW., Washington, DC).

18. C. Allison et al., "Severe Core Damage and Associated In-Vessel Fission Product Release," Progress in Nuclear Energy, 20, 2, 1987, pp. 89-132.

SEVERE ACCIDENT ISSUE TOPIC PAPER

2.2 CORE MELT PROGRESSION AND VESSEL FAILURE (IDCOR ISSUE 6)

Issue Definition

In-vessel core melt progression includes phenomena which determine the state of the reactor from the time the water level falls below the top of the fuel to the time of failure of the reactor vessel. This includes the relocation of molten material and refreezing to form a crust in the lower core region, formation of a rubble bed from remaining fuel pellets and oxidized clad, thermal attack of molten material or reactor core structures and lower metallic crust, core debris-coolant interaction in the lower plenum, and reactor vessel failure.

The nature of core melt progression affects the state of the core debris at vessel failure and the timing and failure modes of the reactor vessel lower head. The composition, amount, and temperature of the core debris can affect early challenges to containment integrity by direct heating of the containment atmosphere, the oxidation of zircaloy and steel, and the release of refractory fission products due to core-concrete interactions.

It is difficult to model in detail the complex nature of the melt progression phenomena. Simplified models used in integral codes tend to concentrate on lumped energy or momentum balances using various assumptions and adjustable parameters to reflect the uncertainty in the models. Differences in these parameters and differences in treating the applicable phenomena in MAAP[1] and SCDAP[2] lead to different results with respect to initiation of core relocation, blockage formation, and vessel meltthrough.

In-vessel hydrogen production during core melt progression is discussed in ARSAP Severe Accident Topic Paper 2.1. Other related phenomena, including the effect of natural circulation, fission product release, and energetic core debris coolant interaction, are discussed in other topic papers.

Eventually, due to continued generation of decay heat, the molten material slumps to the bottom of the reactor vessel. This, according to IDCOR analyses,[3] leads to localized meltthrough at locations where instrument tubes penetrate the lower vessel head. However, the NRC takes a position that, as the MARCH code[4] assumes, the entire lower head heats up and that, due to excessive stress, it will fail coherently rather than at localized points. The MARCH code further assumes that the entire core inventory is present in the lower head when the vessel failure occurs. IDCOR analyses estimated that only a fraction of the core inventory would exit the vessel at failure.

The mode of vessel failure is of secondary importance in sequences for which the system is at low pressure. In ALWRs equipped with a dedicated safety depressurization system, this low-pressure condition is expected in the most probable of severe accident sequences.

The outstanding uncertainties related to core melt progression phenomena include:

- Physical properties and important physico-chemical interactions
- The threshold and mechanisms of molten zircaloy relocation and extent of blockages
- The formation and characteristics of a metallic crust in the lower core region
- Collapse of ceramic fuel and oxidized zircaloy to form a rubble bed

- o Thermal attack and failure of the metallic crust by the molten corium pool

- o Core debris-coolant interaction in the lower plenum

- o Thermal and mechanical loads on the reactor vessel lower head and resultant failure modes.

Historical Perspective

Industry Action to Address the Issue

A simple model to track the candle-like relocation of clad material and fuel was developed for MAAP Version 3.0. The molten material is assumed to accumulate in the lowermost node until it becomes completely molten; at that time the molten material in that node and adjacent nodes enters the lower plenum without fragmentation. Limited core debris quenching is expected and the heatup of the vessel head and failure of local penetration welds occurs within tens of seconds to a few minutes. [illegible]ation of the surrounding steel is modeled in MAAP and results in blowdown within 4 to 80 seconds, depending on the sequence. In sequences for which the primary system is at elevated pressure, failure of the welds would cause the instrument tubes to be rapidly pushed out of the vessel. After core debris begins flowing through one or more of the instrument tube penetrations, the penetrations would be rapidly ablated. Past benchmarking activity with MAAP was used to demonstrate consistency with the available data from experimental tests and with TMI.[5,6,7]

NRC Action to Address the Issue

The mechanistic MELPROG and SCDAP codes treat the major phenomena during core melt progression. A semi-mechanistic melt progression model that considers the momentum and energy of molten film is part of the SCDAP code (see Reference 2). SCDAP and MELPROG analyses of hydrogen production have shown significant sensitivity to the temperature at which relocation begins.[8] MELPROG treats the thermal attack of the lower metallic crust and the interaction of core debris upon the reactor structure, the vessel head, and vessel penetrations. Recently, Sandia National Laboratory (SNL) has questioned these assumptions made by IDCOR concerning vessel failure.[9] First, SNL concludes that significant fragmentation and quenching would occur in the lower plenum. Next, even if a jet penetrated the water pool, the welds may be protected by a layer of low melting point material, e.g., a layer of control rod silver such is believed to exist at TMI. Finally, binding of the instrument tubes in the penetrations due to differential thermal expansion may prevent ejection of the tubes.

The NRC Position

The NRC does not believe that the available data is sufficient to resolve the issue of blockage formation, core melt progression, and the mass and composition of core debris that is expelled at vessel failure.[10] Although the NRC states that the industry assumptions, as incorporated in MAAP, are plausible and consistent with some data,[11] the uncertainty in these processes is sufficiently large to preclude endorsement or agreement at this time. Therefore, the NRC staff recommends that modeling of melt progression, with and without blockages and with a large range of material quantities that can be expelled from the reactor vessel, should be used in accident analysis to cover the range of uncertainties.

> "It is also our judgement that core melt progression phenomena, including multi-dimensional natural circulation effects and failure of steel structures, are sufficiently uncertain that the mass and composition of core debris released at vessel failure should be treated parametrically in plant analyses. Accordingly, it is the staff's position that the release of a larger mass of core debris (than presently assumed by IDCOR) containing carious amounts of steel should be considered by IDCOR in establishing uncertainty bounds on risk, particularly with regard to the issues of hydrogen combustion and direct containment heating. The range

of debris mass and composition considered should encompass the results of MARCH calculations, or alternatively could be based on the results of calculations using more mechanistic codes such as MELPROG with conservatism applied to account for uncertainties.

Technical Approach to Resolve the Issue for ALWRs

The approach to resolution of the issue of core melt progression and vessel failure for ALWR is (1) to improve the core relocation model and to provide the technical basis for those improvements, (2) to provide additional technical justification for the present model of vessel failure, (3) to provide sensitivity studies of debris and vessel failure during depressurization and (4) to provide sensitivity studies to estimate the potential effect of other important phenomena on plant response and source terms.

The purpose of this work which is described below, is to provide the NRC with the technical basis to verify the acceptability of these models for ALWR severe accident analyses.

1. Changes in MAAP to describe more mechanistically the melt progression will be introduced as part of the ARSAP program. These changes will include the prediction of eutectic compositions using the UO_2-ZrO_2-Zr phase diagram, the use of a threshold breakout temperature to describe release from the oxide layer, a melt flow and heat transfer model for the eutectic, a model for the molten core debris behavior, and consideration of the attack of the metallic lower crust. The new models will be compared with existing detailed analyses and experiments of core melt progression to demonstrate their validity for ALWR accident analysis.

2. Studies will be performed by ARSAP to provide additional technical basis for the present models of vessel failure:

- A review of the relevant literature will be conducted to assess and supplement the basis for fragmentation assumptions currently used.

- Results from the detailed core melt progression model being developed will be used to assess whether sufficient low melting point materials will exist to protect the welds.

- Calculations performed by Sandia National Laboratories to assess the binding potential of the instrument tubes will be reviewed and compared with the current model assumptions.

3. Sensitivity studies of debris behavior in the lower plenum, in which the safety depressurization system is operational, will be conducted to establish the time required for debris quenching, debris bed dryout, and repressurization to occur. The results will be used to support the conclusion that the system will be depressurized at vessel failure and that vessel failure occurs at a penetration rather than around the circumference of the vessel.

4. Sensitivity studies of important parameters in the improved models will be performed to determine if these are any identify the significant remaining uncertainties which could cause early challenges to containment and could affect fission products released during core debris/concrete interaction. This analysis will be performed to show that a containment designed to meet the severe accident mitigation requirements of the EPRI ALWR Requirements Document is not significantly affected by uncertainties in core melt progression.

References

1. Fauske & Associates Inc., MAAP(3.0)- Modular Accident Analysis Program, User's Manual, IDCOR Technical Report 16.2-3, Atomic Industrial Forum, February 1987.

2. G. A. Berna et al., SCDAP/Mod 1/VO: A Computer Code for the Analysis of LWR Vessel Behavior During Severe Accident Transients, EG&G Idaho, IS-SAAM-84-002, Rev. 1, Idaho Falls, Idaho, July 1984.

3. Fauske & Associates, Inc., Debris Coolability, Vessel Penetration, and Debris Dispersal, IDCOR Technical Report 15.2B, Atomic Industrial Forum, August 1983.

4. R. O. Wooton, P. Cybulskis, S. F. Quayle, MARCH 2 (Meltdown Accident Response Characteristics) Code Description and User's Manual, Battelle Columbus Laboratories, NUREG/CR-3988, BMI-2115, September 1984.

5. A. Sharon, J. R. Gabor, R. E. Henry, "Simulation of the Severe Fuel Damage Tests Using MAAP," Proceedings of the International ANS/ENS Meeting on Thermal Reactor Safety, San Diego, California, February 2-6, 1986, Vol 4, p. XXIII 2-1.

6. Fauske & Associates Inc., Simulation of LOFT Experiment LP-FP-2 Using MAAP Version 3.0, Report 86-27, Rev. 1, Burr Ridge, Illinois, June 1987.

7. M. A. Kenton et al., Simulation of TMI-2 Accident Using MAAP 2.0, Electric Power Research Institute, EPRI NP-4292, Palo Alto, California, 1984.

8. C. M. Allison, EG&G, private communication, ARSAP Technical Workshop at EG&G, Idaho Falls, June 1987.

9. M. Pilch, Presentation to NUREG 1150 Review Committee on In-Vessel Issues, Sandia National Laboratory, November 13, 1987.

10. T. Speis, USNRC, "Summary Paper for the Resolution of NRC/IDCOR Issue 5," attachment to letter to A. Buhl, IT Corporation, March 11, 1987.

11. U.S. Nuclear Regulatory Commission (USNRC), Reactor Risk Reference Document, Draft, NUREG 1150, App. J-O, Vol. 3, February 1987, pp. J.2-1 to J.2-23.

SEVERE ACCIDENT ISSUE TOPIC PAPER
2.4 CONTAINMENT PERFORMANCE (IDCOR ISSUE 15)

Issue Definition

This issue is concerned with the way in which PWR containment performance is addressed in severe accident assessments. Containment structural failure can occur by overtemperature, overpressurization, or a combination of the two. A potential cause of overtemperature failure is the contact of hot core debris with the containment structure. This failure mode is not a major contributor to PWR containment failure and can be easily precluded by design. Failure by overpressurization can occur by two modes, gross rupture (a large failure allowing rapid depressurization) or "leak-before-break" failure (containment leakage increases until the pressurization is terminated).

IDCOR considered the dominant containment failure mode to be leak-before-break as a result of a large strain of the containment wall. This is generally due to the failure of the liner at penetrations at high containment strains, a conclusion supported by several studies carried out in individual probabilistic risk assessments.[1,2,3]

The NRC does not agree with IDCOR concerning the mode of overpressure failure, believing that a spectrum of containment failure modes and sizes should be examined.[4] NRC's position results from experimental evidence and prediction of gross failure from the Sandia test[5] performed with 1/8-scale pneumatically pressurized steel vessels. However, analysis performed on actual designs containing penetrations indicate that leaks would occur at the penetrations prior to reaching containment ultimate failure conditions (see Reference 3). For reinforced concrete containments, leak-before-break is not as controversial, as demonstrated by a large-scale test.[6] However, prediction of the location and area of failure remains an outstanding question. NRC is more confident in predictive techniques of ultimate failure pressure for steel shell containments than for reinforced concrete containments. Due to inconsistency and lack of confidence in prediction of leakage rates, NRC believes that a threshold model which allows no leakage before failure should be used when analyzing containment performance and that a spectrum of failure and sizes should be considered.

Historical Perspective

Industry Actions to Address the Issue

As part of the NRC/IDCOR issue resolution, a new model was developed for the MAAP code for a strain-induced failure due to pressurization.[7] This model allows failure of the containment by either reaching the ultimate stress or satisfying the criteria of the leak-before-break mode, depending upon the specific containment construction and criteria for these modes. Leak-before-break occurs when a maximum tolerable displacement at a penetration is exceeded. The model is applicable to large, dry PWR steel or concrete containments. IDCOR benchmarked the strain model against experiments for both steel shell and prestressed concrete containments (see Reference 7). The benchmarking exercise showed good agreement between the strain models and the experimental results.

NRC Actions to Address the Issue

At the time of the NRC/IDCOR issue resolution process, the NRC had established a containment integrity research program. This program includes large-scale tests at Sandia National Laboratories to investigate containment failure due to pressurization. Currently, a 1/8-scale steel shell containment experiment (without the surrounding concrete shield building) has been completed and results have been made available (see Reference 5). No significant leakage was detected prior to rupture, and the failure was a gross rupture after significant overpressurization. The gross failure occurred at about 190 psig as compared to a design pressure of 50 psig. Further, the experiment did not take into account the potential for tears due to interactions between prototypic equipment penetrations and the containment, nor did it include interactions between the containment wall and the shield building.

Another experiment utilizing a 1/6-scale reinforced concrete containment has recently been performed (see Reference 5). The containment model was pressurized to about a factor of three above its design pressure, and substantial leakage prevented further pressurization. The leak before break was caused by a liner tear adjacent to equipment penetrations.

The NRC Position

With regard to the IDCOR commitment for uncertainty analysis on containment failure size, the NRC has stated in Reference 4:

> "The staff finds the IDCOR commitment to consider the spectrum of containment failure modes for various containment designs and the influence on environmental releases to be acceptable."

In general, the NRC has stated:

> "Although we conclude that the SNL 1/8 model steel experiment demonstrated that analytical methods can predict with a high degree of confidence the onset of failure of a steel containment, we can not make a similar conclusion for the concrete containment. In addition, since leakage criteria for penetrations have not been developed and verified by either the staff or IDCOR, we can not concur on the adequacy of the IDCOR conclusion that the dominant containment failure mode is leak before break. It is, therefore, the staff position that until such time that the leakage criteria have been developed based on the results of separate effect experiments that have been conducted on electrical penetration assemblies, isolation valves and seal and gasket material, it should be assumed in severe accident analyses that the containment fails upon reaching the threshold pressure."

NRC has further stated:

"Finally, since rupture is often caused by highly localized phenomenona that may be difficult to anticipate, analyses with large containment failure sizes (e.g., values used in NRC risk studies) must be undertaken. For containments that are completely surrounded by an enclosure building where credit for deposition of fission product is assumed, several failure locations should be considered in the analyses to establish the most likely place for containment failure. The rupture criterion for steel containments should be based on the uniaxial tensile strains at maximum load. This will yield reasonable estimates of the bursting strength provided the maximum strain in the containment is accurately predicted. For concrete containments, if the following criteria are used:

- o yield of reinforcement for reinforced concrete containments,

- o one percent tendon strain in prestressed containments, deformations will be small enough that no significant leakage would develop and the containment retaining capability is assured."

Technical Approach to Resolve the Issue for ALWRs

The proposed technical approach to resolve the issue of how to address issue of containment performance in severe accident assessment is:

- o Detailed structural analysis using realistic configurations and severe accident conditions will be performed to assess the potential for leakage at penetrations prior to reaching ultimate containment failure conditions. This analysis will be used to support the position that leak-before-break will occur, and that realistic leakage criteria should be applied in containment performance analyses.

- Further, the PRA performed in support of the design will include a sensitivity analysis to assess the impact of various containment leak sizes and locations on the PRA results.

- Advanced PWR containment designs will be developed such that core debris will not come into contact with the containment boundary in a manner or amount which could lead to containment overtemperature failure. This approach is consistent with the EPRI ALWR Requirements Document, Chapter 5,[8] Requirement 6.6.3.1, and in Chapter 6 of the EPRI Requirements Document).[9]

References

1. Zion Probabilistic Safety Assessment, Commonwealth Edison Company, Chicago, Illinois, September 1981.

2. Pickard, Lowe and Garrick, Inc., Seabrook Station Risk Management and Emergency Planning Study, PLG-0432, Newport Beach, California, December 1985.

3. Wolfgang K. E. Braun et al., "The Reactor Containment of Standard Design German Pressurized Water Reactors," Nuclear Technology, 72, March 1986, pp. 268-280.

4. T. Speis, USNRC, "Summary Paper for the Resolution of NRC/IDCOR Issue 15: Containment Performance," attachment to letter to A. Buhl, IT Corporation, March 11, 1987.

5. D. B. Clauss, Comparison of Analytical Predictions and Experimental Results for a 1:8 Scale Steel Containment Model Pressurized to Failure, NUREG/CR-4209, SAND85-0679, July 1985.

6. Atomic Industrial Forum Infowire message 87-51, Atomic Industrial Forum, Bethesda, Maryland, 1987.

7. Fauske & Associates, Inc., Modeling Additions to the Modular Accident Analysis Program Version 3.0, FAI/86-38, July 1987.

8. Electric Power Research Institute, Advanced Light Water Reactor Requirements Document, Chapter 5: Engineered Safeguards Systems, Palo Alto, California, December 1987.

9. Electric Power Research Institute, Advanced Light Water Reactor Requirements Document, Chapter 6: Building Design and Arrangements, (to be published).

SEVERE ACCIDENT ISSUE TOPIC PAPER

2.5 HYDROGEN IGNITION AND BURNING (IDCOR ISSUE 17)

Issue Definition

The critical question in this issue is the temperature and pressure loads imposed upon the containment as a result of hydrogen combustion. The NRC and IDCOR interactions evolved to the point that clear differences in the modeling of the phenomena, both in the hydrogen burn models and the integrated accident analysis, were identified between NRC and IDCOR models. The advanced PWR designs currently being developed will have flexibility that was not available to current generation plants. Many of the modeling issues identified during the IDCOR/NRC interaction process can be resolved by accounting for the potential severe accident loads during the design of the ALWR. The proposed resolution of this issue deals with designing the containment such that hydrogen detonation is precluded and with providing adequate structural margin in the containment design such that it can accommodate the containment loading due to combustible gas burning.

The first aspect of this issue, the load produced by detonation, is addressed by two design features. First, although the containment will be designed based on the design basis, the containment free volume will be such that the maximum global concentration of hydrogen, based on oxidation of the equivalent of 75% of the zirconium cladding in the active core, will not exceed that necessary (13% by volume) to support detonation. In order to determine the limiting global containment concentration a review of experimental data was performed to identify the concentration necessary to support hydrogen detonation.[1] The result of this review indicates that global detonation can be precluded for concentrations below 13% by volume. This free volume requirement will eliminate containment loads associated with global detonation. To address localized detonation the second design feature will be, to the extent possible, to design the containment to promote mixing of the containment volume.

This mixing precludes any significant local high concentrations above the global concentration limit. Any containment location which has the potential for local detonation will be evaluated to assure that the containment can withstand the local detonation or will be provided an ignitor system to preclude hydrogen buildup to detonable levels.

The second issue related to hydrogen combustion loads is associated with the containment pressurization caused by global hydrogen burn. A global concentration of about 8% hydrogen in dry air will allow a global burn to occur. For humid environments the required hydrogen concentration is adjusted upward using a correlation that has a substantial basis of experimental data.[2] This criterion is invoked to account for the effects of steam on the hydrogen burn threshold concentration. In order to simplify this complex phenomena for purposes of design, the conservative assumption will be made that the burn takes place adiabatically in dry air, and therefore the threshold concentration is 8%. This is consistent with present regulation related to containment conditions for assessing hydrogen burns.[3] Thus, the potential exists, given permissible containment hydrogen concentrations, for a global burn to occur in the containment. The ability of the containment to withstand hydrogen burn loadings will need to be addressed. The containment's ability to withstand this loading will be evaluated by the performance of a best-estimate analysis to determine the ability of the containment, given the design margins present, to withstand the pressurization equivalent to a complete global hydrogen burn. This evaluation provides assurance that the containment will be robust and will provide adequate capability to mitigate the effects of hydrogen burns.

There remain several clear differences in the NRC and IDCOR hydrogen models. These differences are highlighted in the following discussion. In order to reduce the uncertainties associated with the modeling issues, modifications to the MAAP code will be performed to allow MAAP to predict experimental results more closely. These modifications reflect MAAP code modifications performed in response to the issues presented by the NRC during the IDCOR/NRC issue resolution process.

Elements of the IDCOR and NRC burn models for which differences exist are ignition criteria, evaluation of the rate and completeness of burns, intercompartmental flame propagation, allowance for phenomena of recombination at high temperatures and ignition of hot jets, and the treatment of localized burning at igniters. Aspects of integrated analysis which contribute to the issue are: containment nodalization, gas transport by natural convection, and coupling of core-concrete interactions with containment models.

IDCOR burn models (MAAP 3.0) use a flame temperature criterion for the ignition criteria and to determine if complete combustion occurs. The models then scale the combustion time using the laminar flame speed and characteristic compartment length.[4] The flame temperature criterion is a threshold adiabatic flame temperature (calculated a priori given the gas concentrations and temperature) above which burns are allowed to occur. This criterion is adjusted based on calculated conditions (for example, humidity) to provide a correlation between MAAP predictions and experimental data. The effects of hydrogen/oxygen recombination are accounted for in the models when the gas temperature exceeds the adiabatic flame temperature. Of the three important factors needed to pass the threshold for ignition, the oxygen concentration is limiting because adequate hydrogen is present and temperatures are extreme. When jets of combustible gas enter an oxygen bearing compartment, MAAP models the process where burning occurs as the jet entrains oxygen. Finally, MAAP contains an explicit model for local burning at igniters.

In contrast, the NRC code, HECTR, accepts user-input hydrogen concentration as a threshold for combustibility, and checks for inerting by excess steam concentration or insufficient oxygen concentration.[5,6] Burn time is treated in a manner similar to that used in MAAP 3.0, but burn completeness is determined by correlations to Sandia test data. There are no explicit NRC models for recombination, jet burning, or igniters.

Propagation of flames between compartments is allowed by MAAP 3.0 in two ways. First, a burn is allowed into an overlying compartment when there is a vertical line of sight between the igniter or other ignition source and the upper volume. Second, any burn in one compartment can induce a burn in another when the resulting increase in temperature and pressure, or transport of combustible gases, causes the flame temperature criterion to be met in the second compartment. NRC models allow flame propagation when user-input threshold concentrations are within bounds in a compartment adjacent to the burning compartment.

Containment nodalization is fixed by specific containment types in the MAAP 3.0 code. It is usually left fixed in the NRC CONTAIN model, which performs much of the containment thermal-hydraulics, but is varied in the HECTR. Nodalization can influence the analysis of combustion by artificially allowing or prohibiting mixing within the containment, by assigning heat sources or sinks differently, and by assigning injection or removal junctions differently. In NRC analyses, the reactor cavity volume is sometimes lumped in with other compartments, in which case effects of high temperatures in this region are lost. The capability to predict conditions allowing recombination or jet burning is also lost with such nodalization. Upward heat losses during core-concrete interactions are imposed upon the cavity in MAAP analyses, but are not present in some NRC analyses. There is a feedback of cavity temperature on the rate of concrete erosion and thus the production rate of combustible gases during these interactions.

In summary, the key issues are the ability of the containment to withstand temperature and pressure loads generated during hydrogen combustion and the ability of the containment design to preclude detonation. An important issue related to these key issues deals with the ability of the analytical models to model hydrogen ignition and burning adequately.

Historical Perspective

Industry Actions to Address the Issue

As part of the NRC/IDCOR issue resolution process, IDCOR performed benchmarking of its burn models against a variety of experimental configurations. The volumes for the benchmark experiments varied from 0.3 to 2408 m^3; hydrogen concentrations varied between 5 and 13% by volume and steam concentrations varied between 0 and 41%. Most of these experiments were chosen because the igniter burn model was tested during the experiment, but in some cases the global burn model was exercised as well.

The flame temperature ignition criterion was not re-evaluated by IDCOR as part of the resolution process, but the assumption of complete global burn was reviewed (see Reference 2). It was found that no kinetic or equilibrium barriers existed to prevent completeness. It was shown in original IDCOR work[7] that, for hydrogen concentrations above 8% in dry air, combustion is essentially complete, and this corresponds to invocation of the flame temperature criterion. For steam addition to dry air, the flame temperature is modified in accordance with an experimentally determined correlation (see Reference 3).

NRC Actions to Address the Issue

As part of its ongoing research, the NRC completed large-scale hydrogen burn studies at the Nevada Test Site.[8] Some data from these tests have been made available and used by IDCOR. Much of the current HECTR work is based upon the VGES series[9] completed at Sandia prior to the issue resolution process.

The NRC Position

The NRC staff has concluded in Reference 10:

"With regard to hydrogen ignition models, the staff believes that the effect of ignition delays beyond the time at which the IDCOR flame temperature criterion is first satisfied should be considered by IDCOR in all future MAAP analyses for plants and sequences in which igniters are not available, and in estimating uncertainties in risk for all plants. This should include consideration of delays in ignition until (1) the time of reactor vessel failure, and (2) various times following vessel failure, up to 25 hours. For plants and sequences in which igniters are available, the IDCOR ignition criterion (incomplete combustion) is unacceptable in its present form, as it does not adequately account for the effects of steam and suspended water droplets. In addition, the burn rate model results in burn times inconsistent with experimental data. For these plants and sequences, future analyses should incorporate revised models to rectify the noted deficiencies.

"With regard to recombination in the reactor cavity, the staff believes that the IDCOR models are appropriate for dry cavity sequences in which significant recirculation flows are predicted to occur, provided that IDCOR modifies their model to account for the reverse reaction of steam with steel in the cavity. This position is contingent upon satisfactory verification of the adequacy of the MAAP model in predicting natural circulation flow. For flooded cavity sequences no recombination should be assumed."

Technical Approach to Resolve the Issue for ALWRs

The proposed approach to resolution is (1) to implement EPRI ALWR design requirements which will ensure that advanced PWR containments are designed with sufficient margins to accommodate temperature and pressure loads imposed during severe accidents, and (2) to incorporate modifications in the MAAP code which address NRC concerns and improve agreement between the MAAP code and experimental data. The technical approach to resolution is described below:

1. The advanced PWR containment design will meet the following criteria:

   o The design of the advanced PWR containment will be such that the uniformly distributed concentration of hydrogen is assured to not exceed 13% under realistic severe accident containment conditions. Containment dilution (i.e., increased free volume) is the preferred approach to limit this concentration. If this is not achievable then hydrogen igniters will be used to assure that this limit is not exceeded. However, if the use of igniters is the means of control, critical plant equipment, including that equipment necessary for maintaining containment integrity, must be capable of performing their function during and after their exposure to hydrogen burning. This is consistent with EPRI ALWR Requirements Document, Chapter 5,[11] Requirement 6.5.2.1.

   o Containment design will ensure effective mixing of gases in the containment atmosphere so that local detonations of hydrogen are unlikely. This will be accomplished by meeting the containment mixing requirement specified by the EPRI Requirements Document, Chapter 5, Requirement 6.5.2.4 (see Reference 11). This requirement addresses mixing in areas in which hydrogen could be introduced to the containment from the primary system, such as relief valves, rupture disks, and breaks in the coolant loops. If there are locations for which it cannot be shown that the atmosphere is mixed, or that the hydrogen concentration cannot be limited to 13%, local detonations will be considered in the assessments of the functional capability of key equipment required for severe accident prevention and mitigation.

- A best-estimate assessment of containment performance will be performed based on realistic accident scenarios identified by PRA. This assessment will include a structural analysis verifying that the containment can withstand pressures resulting from global hydrogen burns of up to 13% hydrogen concentration in the absence of igniters.

- The technical basis supporting the 13% hydrogen concentration limit for detonation, will be provided.

2. Modifications to the MAAP code will be made to address NRC concerns presented by the NRC position letter. This includes modifications to the igniter model, the burn completeness model, and improvements to the model for the transition between incomplete and complete burns. The models dealing with ignition criteria and the interaction of steam and steel in the reactor cavity during recombination will be reviewed.

References

1. Fauske & Associates, Inc., Technical Support for the EPRI Hydrogen Control Requirements for Advanced Light Water Reactors, Advanced Reactor Severe Accident Program, [to be published].

2. Fauske & Associates, Inc., IDCOR Technical Support for Issue Resolution, IDCOR Technical Report 85.2, Atomic Industrial Forum, July 1985.

3. Code of Federal Regulations, 10CFR50, Part 44, United States Government, 1987.

4. Fauske & Associates, Inc., MAAP (3.0)- Modular Accident Analysis Program User's Manual, IDCOR Technical Report 16-2.3, Atomic Industrial Forum, February 1987.

5. U. S. Nuclear Regulatory Commission (USNRC), Reactor Risk Reference Document, Draft, NUREG-1150, Vol.3, App. J-O, February 1987, pp. J-21 to J-2.28.

6. Allen L. Camp et al., HECTR Version 1.0 User's Manual, Sandia National Laboratories, NUREG/CR-3913, SAND84-1522, February 1985.

7. Hydrogen Combustion in Reactor Containment Buildings, IDCOR Technical Report 12.3, Atomic Industrial Forum, 1983.

8. A. C. Ratzel, Data Analyses for Nevada Test Site (NTS) Premixed Combustion Studies, Sandia National Laboratories, SAND85-0135, May 1985.

9. W. B. Benedick, J. C. Cummings, P. G. Prassinos, Combustion of Hydrogen-Air Mixtures in the VGES Cylindrical Tank, Sandia National Laboratories, SAND83-1022, May, 1984.

10. T. Speis, USNRC, "Summary Paper for the Resolution of NRC/IDCOR Issue 17: Hydrogen Ignition and Burning," attachment to letter to A. Buhl, IT Corporation, September 22, 1986.

11. Electric Power Research Institute, Advanced Light Water Reactor Requirements Document, Chapter 5: Engineered Safeguards Systems, Palo Alto, California, December 1987.