COMBUSTION ENGINEERING

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LD-88-042

Docket No. STN-50-470F
(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

References: (A) Letter, LD-88-038, A. E. Scherer(C-E) to F. J. Miraglia (NRC), dated June 6, 1988

(B) Letter, LD-87-067, A. E. Scherer (C-E) to F. M. Miraglia (NRC), dated Noemver 24, 1987.

Dear Mr. Miraglia:

In Reference (A), Combustion Engineering submitted proposed resolutions for four (4) of the six (6) Topic Set 2 issues. At that time, we stated that submittal of the remaining two Topic Set issues would follow within 30 days. The purpose of this transmittal is to provide the proposed resolutions to the remaining two issues:

- o Direct Containment Heating (IDCOR Issue 8)
- o Debris Coolability (IDCOR Issue 10)

Combustion Engineering plans to adopt, in the development of the System 80+™ design, the fourteen (14) NRC/IDCOR resolutions which have been submitted previously [References (A) and (B)], as well as the resolutions to the above two (2) issues. We request your early concurrence on their acceptability.

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If you have any questions or comments 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
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AES:ss

Attachment: As Stated

DOE ADVANCED REACTOR SEVERE ACCIDENT PROGRAM

ARSAP Proposed Resolutions for Severe Accident Issues - TOPIC SET 2

- o Direct Containment Heating (DCOR Issue 8)
- o Debris Coolability(IDCOR Issue 10)

* The material in this attachment was developed by the DOE ARSAP in support of C-E's Design Certification Program. The page numbering for these issues is such that they integrate with the previously submitted Topic Set 2 issues.

SEVERE ACCIDENT ISSUE TOPIC PAPER
2.3 DIRECT CONTAINMENT HEATING (IDCOR ISSUE 8)

Issue Definition

For some reactor accidents, such as station blackout or certain small-break loss of coolant accidents (LOCAs), the reactor vessel (RV) may fail while the reactor coolant system is at high pressure. The failure of the RV at high pressure can potentially lead to direct containment heating. Direct containment heating (DCH) is a phenomenon wherein core debris is released under high pressure following the failure of the reactor vessel. The debris is then dispersed out of the cavity and into containment as finely fragmented material. This finely fragmented material transfers its thermal energy to the containment atmosphere, resulting in rapid containment pressurization which may in turn lead to early containment failure. In addition to the heat transfer from the debris to the containment atmosphere, the potential exists for additional oxidation of unreacted metallic core materials (principally zirconium). The oxidation of these unreacted metallic core materials would result in an increase in both hydrogen concentration and containment temperature. This combination in turn gives rise to the potential for hydrogen combustion and has been included in recent PRAs as a potential early containment failure mode.[1]

If, due to DCH, containment failure occurs early in the accident sequence, i.e., near the time of RV failure, there is essentially no containment residence time for fission product deposition within the containment, and the radionuclides (principally the volatile species of cesium iodide, cesium hydroxide, and tellurium) are assumed to be rapidly released to the environment as a high energy release. This would have a substantial impact on off-site consequences and public health risk. Hence, the question of whether DCH does or does not occur has a strong influence on the ultimate risk assessed for current and future plants, as well as on the potential for recovering from a severe accident state.

Bounding calculations for PWRs with large, dry containments show that the pressures associated with the ejection to the containment atmosphere of a large fraction (tens of percent) of the molten core materials from the reactor cavity[2] could exceed the containment ultimate pressure. These effects have been assumed by the NRC in NUREG-1150 (see Reference 1) to be sufficient to cause early containment failure.

Historical Perspective

Industry Actions to Address the Issue

The industry evaluations, beginning with the Zion Probabilistic Safety Study (ZPSS) and the Indian Point Probabilistic Safety Study (IPPSS), have concentrated on the influence of the reactor cavity geometry in reducing coherent debris dispersal directly into the containment atmosphere. From an industry perspective, the issue has not been whether or not debris could be dispersed from most reactor cavity configurations, but rather that amounts of high-temperature core debris sufficient to cause containment failure would not disperse directly into the containment atmosphere, would not be finely fragmented, and would not rapidly exchange its heat directly with the containment atmosphere.

IDCOR has evaluated[3] the spectrum of current PWR reactor cavity configurations which would influence the debris dispersal and concluded that for most designs, debris dispersal can be ruled out. For other current PWR plant designs, debris dispersal could take place; however, an excessive amount of fine aerosol is not likely to be generated. Thus, although some debris would enter the containment atmosphere, the resulting temperature rise would not pressurize the containment sufficiently to cause containment failure.

IDCOR concluded that the amount of core material which could possibly contribute to DCH is 1.3 to 3.6 metric tons (see Reference 3), depending on the cavity configuration, which is far less than the 30 to 50 metric tons (see Reference 1) required to make DCH a threat to containment integrity. As a bounding example IDCOR considered the case of seven metric tons entering

the containment and transferring its energy by sensible heat transfer, zirconium oxidation, and hydrogen combustion. This resulted in a 0.08 MPa (12 psi) pressure increase above the ambient 0.3 MPa (44 psi) for a large, dry PWR containment with a free volume of 2.5×10^6 ft^3. IDCOR concluded that such an increase in pressure is well within the containment capability.

The IDCOR arguments were first demonstrated in EPRI-sponsored experiments at Argonne National Laboratory[4] using both real and simulant materials. Three-dimensional representations of the cavity, lower and upper compartments, and structure were included. In addition, the effect of water both in the cavity and lower compartment was tested. The result of these experiments showed essentially no pressurization from DCH.

The conclusion that the majority of the debris would be retained in the cavity was not supported by the Sandia National Laboratory experiments.[5] However, it was IDCOR's position that the Sandia National Laboratory experiments did not account for the influence of resistances in the true containment geometry. To demonstrate the importance of reactor cavity configurations, IDCOR sponsored simulant-material studies using an incremental approach to demonstrate that the influence of true containment geometry is the cause of the difference between the Sandia National Laboratory experiments (discussed in the next section) and the IDCOR-sponsored studies. The IDCOR experiments focused progressively on the debris movement for: (a) the reactor cavity only, (b) the reactor cavity and the instrument tunnel, (c) the reactor cavity, the instrument tunnel, and a two-dimensional representation of the lower compartment inside the missile shield, and (d) the reactor cavity, the instrument tunnel, and a three-dimensional representation of the lower compartment. These demonstrations clearly illustrated the influence of the reactor cavity, instrument tunnel, and lower compartment configurations on the amount of debris dispersal. The results indicated progressively high material retention as the experiment approached a realistic three-dimensional representation.

NRC Actions to Address the Issue

NRC recognizes direct containment heating as a major issue and has a major experimental program underway. Experiments performed at the SURTSEY facility at Sandia (see Reference 5) are currently modeling the movement of debris out of a 1/10-scale model of the Zion reactor cavity, as well as the Zion cavity plus the instrument tunnel. Debris is discharged from this portion of the simulated containment into a large (100 m^3) expansion volume intended to simulate the balance of containment. An iron-alumina thermite debris simulant is used, and the SURTSEY chamber is instrumented for pressure, temperature, and aerosol measurements.

Previous experiments also conducted at Sandia, the SPIT and HIPS experiments conducted in the HIPS series, were designed primarily for qualitative observations of debris behavior.[6] In these tests debris was discharged directly to the environment.

These Sandia experiments have not yet addressed structures associated with the containment's lower and upper compartments and the coupling between them, or the presence of water for mitigation.

The NRC Position

The following are representative quotes from the NRC position paper on the issue of DCH[7]

> "Although the staff finds the IDCOR approach to cavity configurations to have merit, it needs to be confirmed by experimental data ...
>
> "Our approach in resolving the DCH issue is to look at this issue within the total context of a core melt event, that is in a realistic best estimate frame work, and to make use of all existing information which is relevant to this issue.

"For PWR, the staff recognizes, subject to experimental confirmation, that some cavity configurations and miscellaneous structures in the lower containment might prohibit the core material ejected from the RPV from being dispersed to the upper containment atmosphere (depending on the relative velocity between the gas and the debris) and participating in DCH that pressurizes the containment. However, we will require IDCOR to provide a screening criteria, ... to determine the potential for entrainment and de-entrainment of the melted core material during the dispersal. For all plants, the licensee will be required to demonstrate that the simplified cavity configuration depicted by IDCOR is an accurate representation of their cavity configuration and that the screening criteria for debris dispersal confirm the conclusions regarding the potential for debris dispersal from that cavity."

"For those plants that have been identified, based on the screening criteria, to allow some fractions of the core material to be ejected from the RPV and dispersed to the upper containment to heat and pressurize the containment atmosphere, the staff will require those licensees to assess the plant's capability to withstand the effect of DCH."

"Obviously, the resulting containment loads due to DCH effects depend principally on the corium masses and the amount of unoxidized metal contents that enter the containment atmosphere. Because of the wide spread in this parameter between IDCOR and Containment Loading Working Group (CLWG) and due to lack of prototypical experimental data, it is the staff opinion that sensitivity analyses to cover the uncertainty range between CLWG estimates and IDCOR estimates be undertaken."

Technical Approach to Resolve the Issue for ALWRs

The proposed resolution to this issue for advanced PWRs with large, dry containment is (1) adherence to a two-fold design requirement, which is being incorporated into the EPRI Requirements Document, Chapter 5,[8] and (2) performance of sensitivity studies of the PRA required to support the design. The specific points in this proposed technical approach to resolution are described below.

1. The advanced PWR with large, dry containment will use a cavity configuration that prevents extensive debris dispersal from the cavity, while allowing the cavity to vent pressures generated by the meltthrough. This will be accomplished by designing the cavity so that it can de-entrain and capture debris which may be entrained during core ejection from a failed vessel and subsequent vessel blowdown. A configuration similar to the Millstone 3 design, shown in Figure 1, has the appropriate design characteristics. Two of the most important design characteristics are cavity geometry and the area of exits to containment volumes surrounding the reactor cavity. Advanced PWR reactor cavities will be required to demonstrate that they have the following design characteristics:

   - The design basis of the reactor cavity will remain the same as for current LWR designs with bottom entry core instrumentation. The cavity shall provide for proper installation and operation of the core instrumentation and provide for inspection and personnel access requirements. Based on current LWR experience, it is concluded that such a design basis will result in a reactor cavity that is a minimum of 3 meters high (see Figure 1), and whose cavity exit area is sufficiently displaced from the reactor pressure vessel to prevent direct transport of debris of

material from the cavity to the lower compartment during interaction of ejecting debris with the cavity structure. Furthermore, the cavity's floor area will be controlled by the debris coolability requirements of 0.02 m^2 per megawatt of thermal power (EPRI ALWR Requirements Document, Chapter 5, Requirement 6.6.3.2). This requirement provides the basis for sizing the cavity floor.

- The reactor cavity must have a collection volume that shall be a minimum of twice the core volume and be located downstream from the path which is the interface between the reactor cavity and the containment. The cavity exit should be located in the ceiling of the horizontal reactor cavity such that the exiting debris stream must experience a 90^0 turn to pass through it. Also, the cavity exit area should be larger than the exhaust area that surrounds the RPV.

- The requirement for a manually operated reactor coolant depressurization system has been included in the EPRI ALWR Requirements Document, Chapter 5, Section 6.6.4.3. The ability to depressurize the reactor coolant system reduces the probability of vessel failure at high pressures, thereby further minimizing the potential for DCH. The system is required to be operable, independent of the status of both off-site and on-site power supplies.

2. The PRA performed in support of the design will include a sensitivity analysis to assess the impact of a representative range (given the ALWR design) of corium masses and amounts of unoxidized metals released to containment.

References

1. U.S. Nuclear Regulatory Commission (USNRC), Reactor Risk Reference Document, Draft, NUREG-1150, Vol. 3, App. J-0, February 1987, pp. J-21 to J-2.28.

2. USNRC, Estimate of Early Containment Loads from Core Melt Accidents, Draft, NUREG-1079, December 1985.

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

4. B. W. Spencer et al., "Corium/Water Dispersal Phenomena in Ex-Vessel Cavity Interaction," International Meeting on LWR Severe Accident Evaluation, Cambridge, MA, August 28 - September 1, 1983, Vol. 2, pp. 15.5-1 to 15.5-7.

5. Reactor Safety Research Department, Summary of FY 86 Activities and Results, Sandia National Laboratory, March 1987.

6. W. W. Tarbell et al., High Pressure Melt Streaming (HIPS) Program Plan, Sandia National Laboratories, SAND82-2477, NUREG/CR-3025, August 1984.

7. T. Speis, USNRC, "Summary Paper for the Resolution of NRC/IDCOR Issue 8: Direct Containment Heating (DCH) by Ejected Core Material," attachment to letter to A. Buhl, IT Corporation, March 11, 1987.

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

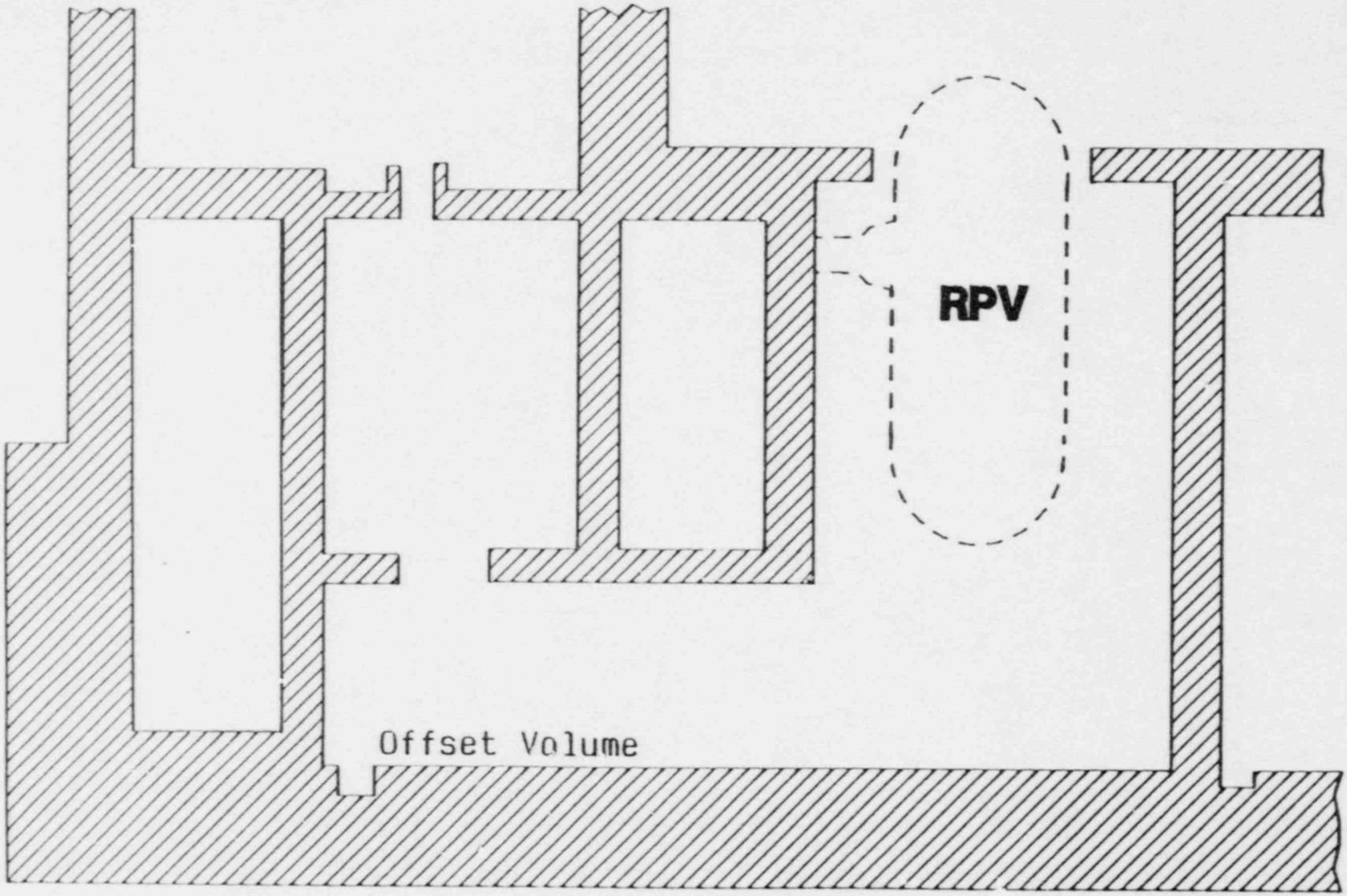

Figure 1. Millstone 3 type lower reactor cavity configuration.

SEVERE ACCIDENT ISSUE TOPIC PAPER
2.6 DEBRIS COOLABILITY (IDCOR ISSUE 10)

Issue Definition

In those accident scenarios in which the reactor vessel fails, high temperature core debris falls into the reactor cavity where it may interact thermally and chemically with the structural concrete. IDCOR Issue 10 was associated with the magnitudes and mechanisms of energy transfer from the molten core debris to the concrete and to the containment atmosphere and surrounding structures, or to an overlying pool of water if one is present. The interactions between NRC and IDCOR evolved to the point of focusing on differences in the modeling approaches used in their respective analyses, and on the impact of those differences on predicted thermal and pressure loadings on the containment.

The NRC staff concurred that the "overall heat transfer predictions [of the IDCOR model] appear to be in reasonable agreement with available data," and that Issue 10 is resolved, "contingent on comparisons being made with new experimental data when such data becomes available." ARSAP ALWR Severe Accident Issue Set 1 included Issue 10 as a "resolved NRC/IDCOR issue" and recommended that the proposed resolution of the issue for ALWRs be obtained in precisely the manner agreed upon and documented in the NRC/IDCOR interaction process. Part of the ARSAP program planning, in fact, includes continued comparison of models to experimental data as those data become available.

However, an important part of the issue that was not considered by the NRC to be resolved is the coolability of the debris. The NRC staff did not believe that IDCOR had provided adequate justification for its criteria for determining coolability of the core debris. This paper is written to address for ALWRs this unresolved issue of the coolability of core debris in scenarios which lead to core debris falling into the reactor cavity.

Coolability of the debris requires two conditions: (1) quenching of the debris and (2) establishment of a heat transport path (to an ultimate heat sink) sufficient to remove decay heat produced in the bed. It is the ARSAP position that these conditions will be met in ALWRs, that the debris will be coolable, and that these accident sequences therefore would be terminated with the plant in a safe stable state.

Historical Perspective

Industry Actions to Address the Issue

The industry-sponsored IDCOR program has based the assessment of debris coolability on experiments carried out by SNL and on the assessment of debris-water interactions during the TMI-2 accident progression/termination.[1] Evaluations of the TMI-2 accident for those conditions where water was introduced into an overheated core, as well as those times in which the core material migrated to the water-filled lower plenum, demonstrate that a critical heat flux representation for the quenching rate provides adequate description for comprehending the debris behavior. In addition, EPRI-related efforts with respect to quenching of lava beds[2] have demonstrated that water ingression is sufficient to sustain a quenching rate long after conduction limitations would be achieved by crust formation at the interface. This indicates that water ingression into the porous quenched region is sufficient to allow sustained quenching of the molten core material. Hence, crust formation would not provide a substantial limitation to sustained quenching.

IDCOR has developed models based on these concepts and these have been incorporated into the MAAP code.

NRC Actions to Address the Issue

Research at Sandia National Laboratories (SNL) has been the principal focus of NRC-sponsored research and has concentrated on core-concrete attack experiments with water added after concrete ablation has been initiated. In the initial experiment, a 50-kg solid ingot of stainless steel was heated inductively in a concrete crucible.[3] In the SWISS-1 and SWISS-2 tests,[4] molten stainless steel was added to a concrete crucible and inductively heated from the outside with water added both at the beginning (SWISS-2) and end (SWISS-1) of the test. The first experiment was limited in its instrumentation, and therefore provided limited information of debris quenching rates, but the available experimental evidence is consistent with the IDCOR approach. The follow-on SWISS-1 and SWISS-2 tests had much more extensive instrumentation and also measured the heat flux to the overlying water pool. This measured heat flux was within 25% of the nominal value recommended by the IDCOR program and essentially equal to the lower uncertainty bound assigned to the debris quenching rate.

Consequently, the SNL experiments carried out under NRC sponsorship are consistent with the IDCOR approach used in the reference plant containment analyses.[5]

The NRC Position

The NRC stated the following in their response to IDCOR on the issues of core-concrete interactions and debris coolability:[6]

> "This issue is associated with the magnitudes and mechanisms of energy transfer from the molten core debris to the concrete, and to the containment atmosphere and surrounding structures or to an overlying pool of water if one is present. This issue can potentially impact the mode and timing of containment failure and the chemical forms of the released fission products. This is an area in which major differences exist in the modeling approaches

used in the NRC and IDCOR analysis. In general, the NRC analyses involve more heat going into concrete attack with an associated increased production of non-condensible gases and more rapid pressurization of the containment but lower containment atmosphere temperatures, particularly in BWR analyses. These differences are partly the result of assumptions made regarding debris dispersal and partly differences in the core-concrete attack models. When water is present in the cavity, the IDCOR approach is to use a critical heat flux correlation accounting for some uncertainty to determine if the debris bed is in a coolable state. In the NRC treatment, heat transfer from the debris bed to the water pool is calculated explicitly and the resulting heat balance determines whether the debris cools or heats up further.

"While considerable effort has been expended by IDCOR and the NRC and its contractors in developing these detailed, mechanistic codes used for analyzing core/concrete interactions, there are several problems that affect our confidence in using these codes to predict severe accident behavior in real plants.

"First, there is a sparsity of data that could accurately duplicate the large spectrum of conditions that could be expected during severe accidents. Such experiments are costly and technically difficult to perform, however, there is promise that improved data is forthcoming. The SURC (sustained urania/concrete) experiments that are being performed at SNL during 1986 and 1987 should provide the most prototypical information to date. The SURC tests cover a broad range of conditions and are expected to result in data that will be useful for both verification of certain aspects of the present models and for identifying further model improvements.

"We do not believe that adequate justification has been provided by IDCOR regarding its criteria for determining coolability of the core debris when there is water present. Future code comparisons with the SURC data will be used to help resolve this issue.

"The overall heat transfer predictions appear to be in reasonable agreement with the available data. If the SURC data and the comparisons of the IDCOR codes with that data indicate that the IDCOR treatment of core debris coolability is not supported, we will require modifications to the codes.

"This resolution [of Issue 10] must be viewed as conditional and is based on our current understanding of core-concrete phenomenology. This approval is contingent on comparisons being made with new experimental data when such data becomes available. It is our understanding that IDCOR has agreed to this approach. If significant discrepancies are identified through these comparisons, we will require that further model revisions be made to address these discrepancies. A significant step forward in acquiring this data is expected to be made with the completion of the SURC tests later this year and in 1987. In addition, there is a strong international effort involved in research and code development in the core-concrete interaction area attesting to the recognized importance of this subject.[7] It is expected that this international interest will ensure that the ongoing improvements in our capability to predict the risk contribution from core-concrete interactions will continue."

Technical Discussion of the Issue

The ARSAP position is that the design of ALWRs, in accordance with criteria specified in the EPRI ALWR Utility Requirements Document, will assure that water will be available in such scenarios to cool the

debris in the cavity, that heat transport paths will be available to conduct heat from the debris to an ultimate heat sink, and that the heat removal from the debris will exceed the heat generated due to decay heat. A critical requirement in the EPRI document directly related to this issue is the requirement (Chapter 5, Requirement 6.6.3.2) that the ALWR cavity design allow a surface area of water overlying the debris of 0.02 m^2 per rated thermal megawatt of reactor power. For accident scenarios in which debris would reach the cavity, the decay heat would be equivalent to about one percent or less of full power by the time that debris could be present in the cavity. The cavity design requirement therefore implies that the heat flux off of the debris surface can be greater than 0.5 MW/m^2. And, much of the technical basis for the ARSAP position therefore lies in establishing that the heat fluxes from debris to coolant in the reactor cavity can be greater than or equal to this value of 0.5 MW/m^2.

This value of heat flux is put into perspective by consideration of the maximum heat flux which can be carried away from an upward facing flat surface by water without significant surface temperature escalation, i.e., the critical heat flux. At this point, the physical mechanism of heat transfer changes from nucleate boiling to film boiling. The heat transfer limitation then would be in the coolant. The critical heat flux for saturated water for this configuration at atmospheric pressure is about 1.3 MW/m^2. As the water is subcooled, or as pressure is increased to higher values attainable in a containment, the critical heat flux increases. Thus the requirement of 0.5 MW/m^2 is equivalent to stating that the heat transfer limitation on debris coolability will always be in the debris. Further, demonstration that the limiting heat flux in the debris is not below this value is tantamount to demonstration of debris coolability (given the availability of water and of heat transport paths as noted above).

Two cases can be distinguished for assessment, based on the nature of the debris configuration: (1) the case of a discontinuous

debris bed composed of discrete particles and (2) the case of a continuous debris slab or partially molten pool. The debris bed configuration can arise when sufficient water is initially present in the reactor cavity to quench as it leaves the reactor pressure vessel. This case also includes the possibility of such a bed drying out but not extensively remelting. The continuous slab or pool configuration arises when there is insufficient water, so that initially molten debris can channel through the water (if any) and spread out. It also can result from remelting of initially coolable debris after dryout. Concrete attack can be expected in this configuration, so that concrete slags are added to the debris and concrete offgas stirs and reacts chemically with the debris. An initially solid slab which causes concrete attack will quickly develop molten sections because of the insulating effect of an eroded slag skin, and because of the solubility of oxidic debris.

For the case of a debris bed, coolability is primarily a function of particle size, and the physical mechanism of cooling is water ingression into the bed with outflow of steam from the bed. The coolability limit for debris beds is a hydrodynamic limitation within the bed itself, and beds which are quite deep relative to anticipated bed depths in ALWRs are coolable.

For a continuous debris slab or pool, the initial depth considering the full core fuel plus clad inventory is about 20 cm if the 0.02 m^2/MW cavity sizing requirement is met. The internal temperature of the debris bed containing (low conductivity) oxide fuel can be high enough to produce significant decomposition of concrete. Coolability of such a configuration requires cracking of the slab or overlying crust and ingression of water into the quenched portions of the debris. Since concrete attack produces offgas which can only be relieved by fissures in the crust, and since the debris phase change and cooling to the boiling point of water is accompanied by a volume reduction of about 10 percent, debris in the cavity can be expected to crack, allowing water ingression. As the process continues, the

initially continuous slab or pool configuration approaches the discontinuous debris bed configuration, although the structure of the debris differs (fissured versus particulate).

The discussions below summarize the experimental and theoretical bases for coolability of debris in each configuration.

Experimental Basis for Debris Coolability

It has been shown by several investigators that debris sizes characteristic of quenched light water reactor materials are large, on the order of centimeters. For example, Benz[8] reports results in which 60% of debris is of a size larger than 4 cm, and cites other references with consistent results. In his experiments, initially molten uranium dioxide was dropped into an interaction vessel containing an excess of water necessary to quench the debris. More recently, Spencer[9] has reported a series of experiments with prototypic core debris mixtures of uranium and zirconium oxides and steel. A driving pressure or simple gravity drop was used for injection of the molten material into a cavity (the interaction vessel) filled with varying amounts of water. This region was vented into a larger expansion vessel which could also contain water pools. Particle sizes were measured for debris swept out from the interaction vessel and into the expansion vessel, and typically 40% or more of the debris was of a diameter greater than 1 cm. Debris remaining in the expansion vessel was in the form of solid but porous slabs with varying amounts of coarse fragments.

Much data has been compiled on debris bed heat transfer with varying particle sizes, depths, and coolants.[10,11,12] Of interest is the dryout heat flux, the maximum heat flux which can be sustained without dryout and temperature escalation within the bed. Data taken with water as a coolant show little dependence on debris depth within the range of 3 to 300 cm, but a strong dependence on particle size.

The dryout heat flux for such beds increases with particle size, and is about 2 MW/m^2 for 1-cm diameter, particles. The flux is 1 MW/m^2 for particles of about 2-mm diameter, and falls off more quickly for smaller sizes. A heat flux of 0.5 MW/m^2 corresponds to particle sizes of about 1 mm. It is interesting to note that for particles slightly larger than 2 mm in diameter, the maximum heat flux from a debris bed is approximately equal to the saturated flat plate critical heat flux.

Taken together, these experimental data suggest that debris beds formed during severe accidents will be coolable because large particle sizes should be expected, and the dryout heat flux for such sizes exceeds that of the requirement.

The coolability of an initially continuous slab or pool debris by water ingression has not been conclusively demonstrated by experiment with oxidic urania, but some tests with steel have been made. One test performed at Sandia (see Reference 3) involving penetration of a heated stainless steel slug into concrete was terminated by cutting power and quenching with water. The modeling of this test is discussed further in the following section on analytical basis for debris coolability. Another test series performed at Sandia, the SWISS tests (see Reference 4), was intended to show debris coolability limits for this type of configuration. In the discussion below, modeling of this test will be used to show that the power level was too high for coolability.

The existence of an ingression mechanism has been reported in the cooling of magma, which is also oxidic. One experiment conducted with magma has been documented in which a 12-m thick layer of rock was solidified over an area of about 7000 m^2 (see Reference 2). For 14 days, water was poured on this area at a rate of 100 kg/s. Drillhole measurements showed that the water-cooled rock was at the saturation temperature of 100° C, and samples were observed to be intensely fractured. Below the fractured region lay a conductive transition

layer which separated the cooled rock from the molten magma. Thus, the solidification front was followed by a cracking front, and water ingression allowed deep cooling of the magma field.

Analytical Basis for Debris Coolability

When experiments are modeled so that the heat balance can be examined, and the relative amount of heat loss to concrete, sidewalls, and coolant determined, the coolability of debris and potential for ingression can be demonstrated. One test conducted by Sandia (see Reference 3), initially intended to scope downward heat transfer, was terminated by quenching with water, thus providing a means of testing the ingression model. A 50-kg slug of stainless steel was induction heated and allowed to penetrate sideward and downward in a concrete cavity initially 20.3 cm in diameter. This test was modeled with the DECOMP code through both the heatup and quench, which occurred at about 213 minutes after the start. The debris temperature from this simulation is in excellent agreement with observation. Both the depth of the debris and temperature agreement indicate that water must have caused cracking and then ingressed.

The SWISS tests performed at Sandia (see Reference 3) have also been modeled with DECOMP.[14] In these tests, about 45 kg of stainless steel initially at 1925 K was poured into a crucible and inductively heated for a period of about 40 minutes. An annular sidewall fabricated of MgO prevented sideward penetration, but downward penetration occurred in to the limestone-common sand cylindrical plug. In SWISS-1, water was poured atop the debris about 32 minutes into the test, and in SWISS-2 water was poured on almost immediately. Concrete erosion was nearly equal and proceeded at a nearly constant rate in both tests, attaining a depth of 17 cm. Analysis of these experiments shows that the input power minus side losses to the MgO was high enough to preclude debris coolability. In SWISS-1, about 30 kW were available after side losses are considered, and more if heat due to chemical

reactions is added. This corresponds to an upward heat flux of about 0.9 MW/m^2. This net power corresponds to about 1.2 MW/m^2 in the case of SWISS-2. The measured upward heat flux was about 0.8 MW/m^2 in these tests. Agreement between DECOMP and the experiment was quite good, showing a nearly constant attack rate for both tests.

Availability of Water

The above discussions address the coolability of the debris given the presence of water in the cavity. In order to assure that water is available, the configuration of the reactor containment internals must not prohibit the delivery of water to the core debris. The principal water sources are the primary coolant system, including accumulators, and the in-containment refueling water storage tank (IRWST), which will be located inside containment in ALWR designs.

Technical Approach to Resolve the Issue for ALWRs

As noted previously, it is the ARSAP position that, given (a) an appropriately sized cavity and (b) the availability of water, debris in the reactor cavity, whether in the form of a discontinuous debris bed or of a continuous debris slab or partially molten pool, will be coolable. Further, the existence of heat transport paths to ultimate heat sinks can assure a safe stable state and long-term containment integrity.

The technical resolution, therefore, involves: (1) providing design requirements that assure that the above conditions are met and (2) providing the technical basis to demonstrate that these design requirements will provide the necessary conditions for debris coolability. The technical resolution is described below:

1. The following design requirements will be provided:

- The reactor cavity will be designed to assure that it has sufficient area to permit debris coolability. The current requirement in the EPRI ALWR Requirements Document (Chapter 5 Requirement 6.6.3.2) requires that the cavity design allow a surface area of water overlying the debris of 0.02 m^2 per rated thermal megawatt of reactor power.

- Ample flow paths and areas will be provided to assure that water released in the lower compartment will drain into the reactor cavity. In addition floor slopes and curbs will be designed to enhance drainage to lower elevations. EPRI Requirements Document Chapter 5, Requirement 6.6.3.3 addresses such requirements.

- The IRWST will be configured such that it overflows to the reactor cavity once its inventory significantly exceeds its normal operating volume by a volume (e.g., by a volume to one half the normal operating volume of the reactor coolant system).

- The containment will be arranged to enable heat transport paths (convection, refluxing, or radiation) from the debris to ultimate heat sinks to exist. Provision will be made to assure that additional water can be added to the containment at a rate sufficient to preclude significant boiling. This will be implemented in the event that the heat transport paths cannot be assured. Additional heat capacity in the form of water can extend the time to containment failure and thus permit time for alternative actions. This provision of heat transport paths and additional heat capacity is consistent with the EPRI Requirements Document Chapter 5, Requirement 6.6.4.1, which calls for decay heat removal capability.

2. The technical basis, as highlighted in the "Technical Discussion" section of this paper, will be provided to establish debris coolability given the above design requirements.

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