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INTRODUCTION

Considerations of the behavior of LMFBR core debris begin just after the fuel has reached a subcritical state following a hypothetical core-disruptive accident (HCDA). In an early-sized LMFBR such as FFTF or CRBR, the loss-of-flow accident with failure to scram (LOF-HCDA) could bring the core to a largely molten state when either a mild nuclear transient or gradual overpressurization combined with a melting attack on the upper structures is able to overcome their structural resistance and cause molten core debris to be ejected from the core region. Core debris ejection is expected to be preferentially upward because of the tendency of molten stainless steel cladding to freeze in the lower portions of the core to form blockages early in the accident sequence. Eventually, some fraction of the core debris that remains in the original core region will melt through the lower blockages and downward removal of some of the core debris will occur.

Other hypothetical accidents such as the transient overpower accident with failure to scram and the single subassembly blockage accident could terminate with the ejection of only small quantities of core debris from the core region. Accordingly, analyses of core debris following the LOF-HCDA should encompass the behavior expected in other, less severe accident situations.

The initial post-accident consideration concerns the behavior of core debris within the reactor vessel. For small reactors, such as FFTF, it has been shown [1] to be very likely that the core debris would achieve a coolable configuration within the vessel and the core debris would remain stable as long as some heat removal from the reactor vessel continued. The FFTF reactor is shown in Fig. 1. For larger reactors or for the postulated failure of long term heat removal systems, reactor vessel failure could occur with consequent relocation of the core debris into the cavity below the reactor vessel.

The second important post-accident consideration is that of core debris

Fig. 1. FFTF Reactor

behavior in the reactor cavity. The design of the cavity may or may not include special features to cope with the core debris. The current design of the German SNR-300 reactor includes a cooled catcher-tank for the core [2], which is shown in Fig. 2. For the CRBR, a "parallel design" option included a cooled core-retention system in the reactor cavity [3]. A cross-section of the core retention bed is shown in Fig. 3. However, in the current design, reliance is placed on an inherent retention concept based on the large quantity of concrete in the cavity floor [4]. The third important consideration is the behavior of the core debris in the concrete below the reactor vessel.

In this paper, the available information on core debris behavior, within the reactor vessel, in the reactor cavity and in the concrete base mat are reviewed. A considerable quantity of

Fig. 2. Ex-vessel Core Catcher Concept for the SNR-300 Reactor

information has been developed over the past few years on which to base analyses of such behavior. Overall response of the reactor containment and the expected behavior of fuel, sodium, and fission products within the containment building are treated in the companion papers in this Seminar.

IN-VESSEL CONSIDERATIONS

Breakthrough of the structures just above the active core region following a LOF-HCDA will result in a vigorous ejection of two-phase fluid (fuel/steel liquid and vapor). The molten fuel and steel which is ejected is expected to be quenched in the large sodium plenum above the core.

Core Debris Fragmentation

An extensive series of experiments has demonstrated the strong tendency for molten UO_2 and molten steel to fragment on quenching in sodium [5]. Experiments have ranged from laboratory tests with single droplets to tests with 30 g batches and finally to tests with several kilograms of molten core debris. The

Fig. 3. Ex-vessel Core Catcher Sacrificial Bed Concept for the Parallel Design Option of the Clinch River Breeder Reactor Plant

tests have shown that when molten core debris is injected into an excess of sodium, the core debris forms particulate with a particle size distribution range from 0 to $\sim 5000 \mu\text{m}$ with mean diameters from 50 to $\sim 1000 \mu\text{m}$ (see Fig. 4). Particulate fuel and steel have also been generated in many in-pile tests performed as part of the LMFBF safety program and particulate having similar size distributions was obtained as indicated in Fig. 4.

Fig. 4. Particle Size Distributions of UO_2 from Out-of-pile Experiments M1, M2, M3, M5, M6, and EDT-1, and from In-pile Experiments S3 and E2.

Particulate Settling

The particulate generated by the ejection and quenching of molten core debris in the sodium plenum is expected to settle downward onto the available horizontal surfaces to form particulate beds. Coolant flow through the plenum would cause some of the particulate to be swept into the primary piping instead of settling downward. An analysis of settling and sweepout was made for the conditions expected in FFTF following a LOF-HCDA which showed that more than 85% of the core debris would be expected to remain within the reactor vessel [6]. Material escaping the vessel would be composed of fine particles which would tend to remain suspended in slowly flowing sodium. The rates of settling for typical particle size distributions are such that bed buildup is substantially complete in about 100 s.

Heat Generation in Particulate Beds

The magnitude of the afterheat generated by the radioactive decay of fission products and activation products depends on the operating history of the reactor. Short-lived activity builds up rapidly so that the heat generation rate relative to operating rate just after nuclear shutdown is almost independent of operating time after only a few days of operation. The decay power level at shutdown is $\sim 7\%$ of the operating power level and decreases to $\sim 3\%$ at 100 s and to $\sim 1\%$ at ~ 5000 s. An analysis of the decay heat source indicated that it was convenient to consider four groups of elements [7]. About 50% of the total heat generation is associated with non-volatile oxide-forming elements which should remain dissolved in the fuel. About 20% of the heat originates in non-volatile metallic elements. About 25% originates in volatile elements which are soluble in sodium and 5% is associated with the noble gas fission products. The analysis indicates that the non-volatile elements are likely to remain distributed within the particulate bed so that about 70% of the total decay heat source should remain dissolved in the particulate.

Particulate Bed Coolability

Long term stability of a particulate bed on a horizontal surface depends primarily on the ability to transfer the internally generated heat upward to overlying sodium. The limiting heat transfer capability corresponds to the initiation of a dryout within the bed. Bed dryout is expected to lead to overheating and a melting attack on the substrate. The important upward heat transfer processes are conduction and boiling. Conduction heat transfer depends upon having an overlying pool which is maintained in a subcooled condition. Conduction is capable of cooling beds up to a loading of about 130 kg-fuel/m^2 at an internal heat generation rate of 40 MW/m^3 of fuel ($\sim 3\%$ average FFTF power) [8]. Actual bed depth for such a case is about 25 mm. Coolable bed loadings are increased by a factor of between 2 and 3 by boiling before incipient dryout occurs. Moreover, heat transfer by boiling is effective even if the overlying sodium pool is at saturation temperature. Also, boiling has been shown to effect a significant degree of self-leveling of the particulate. Maximum heat fluxes under boiling conditions were determined by a series of experiments using electrical heating of the sodium phase of particulate UO_2 beds in sodium [8]. Conservative dryout limits are shown in Fig. 5. An understanding of bed

Fig. 5. Conservative Dryout Limits for UO₂ Beds in Boiling Sodium (Experimental Data Points for 100-1000 μ m Size Distribution)

behavior has been augmented by experiments with UO₂ and metallic particulates in water [8-10].

In FFTF, there is a baffle surface which is horizontal and covers most of the vessel cross-section above the level of the core [11] (see Fig. 1). The surface area is large enough so that if the entire core-debris inventory, as particulate, were spread uniformly on the baffle, the bed loading would be about 100 kg-fuel/m², which would be coolable by conduction alone [7]. The added margin to dryout under boiling conditions is a good allowance for the uncertainty regarding the initial uniformity of bed depth. The availability of sufficient bed area was probably the most important factor in the assessment of in-vessel coolability for FFTF. In CRBR, approximately the same cross-sectional area is available; however, about 2-1/2 times the quantity of core debris and decay heat are potentially available. For CRBR, in-vessel core-debris retention is possible but would be more difficult to establish with the available data base. For larger reactors, the likelihood of permanent in-vessel core debris retention might be expected to decrease with further increases in the ratio of fuel mass to cross-sectional area. On this basis, the potential for in-vessel coolability is greatly improved for pool-type reactor systems.

Behavior of Molten Core Debris

The quantity of core debris that is not ejected upward from the core region would be expected to move downward either flowing through or melting through the shielding structures below the core. In FFTF, there is a spherical surface at the bottom of the core-support structure which would have a significant capability for containment of molten core debris (see Fig. 1). Heat removal would occur by downfacing heat transfer to the sodium in the lower inlet plenum of the reactor. Assessment of capability depends on evaluation of the magnitude of local heat fluxes on the fuel-side of the steel structure and on the ability of natural convection to remove the heat on the sodium side [7].

Conditions in the Lower Plenum of the Vessel

Because of the large cooling capability at the lower surface of the core-support structure in FFTF, it was not expected that any significant quantity of core debris would accumulate in the lower plenum. However, there is additional capability for the safe storage of important quantities of core debris as particulate in the lower curved section of the vessel. It is less than at the upper baffle because of the curvature of lower section. The maximum coolable quantity would correspond to a limiting bed loading at the axial centerline of the vessel bottom. At 1% decay heat, this location could accommodate ~35% of the core debris inventory [7]. For CRBR, with 2-1/2 times the fuel quantity, about 15% of the core debris could be coolable on the same basis.

The most likely reason for failure of the reactor vessel in an LMFBR of any size would be because of the gradual buildup of a particulate bed to depths exceeding dryout limits. Details of the dryout process are not completely understood. It is likely that some quantity of steel particulate would be admixed with the fuel particulate. Overheating of the bed would result in steel melting which would then transfer heat very efficiently to the bottom of steel vessel. Recent analyses have indicated that vessel failure would occur as a result of the loss of strength of the steel as it approaches melting and no actual melting of the vessel wall would occur before failure [12]. The nature of the melt-through process indicates that a large area of the vessel would be expected to fail at one time so that the core debris, i.e., particulate fuel, molten steel, and overlying sodium, would flow out of vessel quite suddenly.

The total time elapsed from core disruption to vessel failure could range

from ~ 1000 to much longer times [4]. The longer delays could result from repeated processes of fragmentation, bed formation, bed dryout, debris overheating and failure of successive barriers within the internal structures and finally at the vessel bottom.

CORE DEBRIS BEHAVIOR IN THE REACTOR CAVITY

Core Debris Fragmentation under Ex-vessel Conditions

The rapid flow of core debris into the reactor cavity would occur almost simultaneously with the flow of sodium from the vessel. Two experiments, designated M-5 and M-6, were performed to simulate the conditions of entry of molten fuel and steel into a dry cavity, followed within a few tenths of a second by a large sodium flow [13]. In both experiments, the molten UO_2 was generated by a thermite-type chemical reaction and was found to have been completely fragmented because of interaction with the sodium. It was noted in the previous section that the entering UO_2 might be initially in a fragmented state submerged in a pool of molten steel. If this can be established, then there could be little question about the initial extent of fragmentation of fuel within the reactor cavity.

Cavity Liner Response

Current designs of reactor cavities usually include a steel liner which is designed to prevent contact between the underlying concrete or brick with sodium, especially in the event of a small sodium leak. The current HCDA evaluation for CRBR does not claim any credit for the potential ability of the liner to delay massive contact between the core debris-sodium mixture and the underlying material [4]. The liner would be subjected to severe thermal stresses over the entire base area because of contact with sodium. In addition, there could be a severe local attack caused by a concerted jet of molten core debris during entry. Recent analysis of the process of jet erosion of molten steel on solid steel indicates that jet flow sustained for only ~ 1 s could be sufficient to penetrate a 10-mm steel liner [14]. However, liner penetration could probably be prevented by a ceramic inner liner which would decrease the thermal stresses and prevent jet penetration.

Cooled Crucible Concept

One approach to assuring postaccident fuel containment and heat removal is to convert the reactor cavity into a cooled crucible. This was recommended as the best approach in a recent study [15]

as shown in Fig. 6. The concept is simple in principle; however, it suffers from the disadvantage that active cooling must begin almost immediately or the cooling surface could be destroyed. Design of the cooling system requires a knowledge of the magnitude of local heat fluxes from the core debris in the downward and sideward directions. Heat must be removed from the sodium pool above the core debris as well as from the bottom and sides.

Fig. 6. Cross-section of the Cooled Crucible Core-retention Concept of Ref. 15.

Heat transfer from fluids with uniform internal heat sources has been studied only recently. Kulacki and Goldstein [16] were the first to establish correlations for the upward and downward heat transfer from horizontal layers having equal boundary temperatures. The results were duplicated and extended to greater Rayleigh numbers by Jahn and Reineke [17]. The behavior of idealized horizontal layers has been confirmed by several independent analytical and computer simulation techniques [17-19]. An approximate model for the horizontal layer with unequal boundary temperatures has been reported [20]. Experimental and analytical data for sideward heat transfer from experiments with water have been reported [21,22]. These results were included into an approximate model for heat transfer in pools having vertical sidewalls [21]. Some data are also available on the local heat fluxes along curved pool boundaries [17].

The available results on heat transfer from pools with uniform internal heat sources, noted above, do provide a reasonable basis for conservative design of cooling systems. Preliminary results are available from experiments on heat transfer from internally heated pools of molten UO_2 [21]. These results have indicated that internal radiation in molten UO_2 is a significant mechanism of

heat transport which increased downward heat transfer. However, these effects can be taken into account in design.

One potential uncertainty is the effect of boiling or gas evolution on the heat transfer. The threshold for boiling under a particular set of conditions can be predicted using the nonboiling correlations. Heat transfer correlations for boiling conditions have been reported [23]. Boiling agitates the stagnant lower layer and increases downward heat transfer relative to sideward. This is shown by the plot of the ratio of sideward to downward heat flux vs boiling rate in Fig. 7.

Fig. 7. Experimental Values of the Ratio of Sideward Heat Flux, Q_s , to Downward Heat Flux, Q_d , vs the Upward or Boiling Heat Flux, Q_b , for Internally Heated Boiling Water Pools [23].

Upward heat transfer occurs by vapor transport in this case. Effects of gas release distributed throughout the volume of the pool have been shown to have effects very similar to those of boiling [24]. There is very little data on the effects of gas release from the boundaries; however, this would not occur on a continuous basis in a cooled crucible system.

One other potential uncertainty is the effect on heat transfer of disturbances caused by the continuous addition of core debris to the pool. Initially, this could be severe, especially for the case where the core debris entry is by way of a sustained jet concentrating a thermal attack over one small area. The concern over jet penetration of the crucible could be alleviated by providing a refractory lining on the cooling surface. The lining would be sacrificial in that it would eventually be melted.

Sacrificial Barrier Concept

The sacrificial barrier concept is similar to the cooled crucible with a sacrificial liner except that the liner is quite massive. The melted liner material must be mutually soluble with the fuel so that the heat source is diluted and considerable decay time occurs before the pool approaches the cooled wall. This assures that the cooling system will have to handle only a uniformly low heat flux. Moreover, the cooling system might not be needed for a considerable period of time after the accident.

Most oxide refractory materials are expected to be mutually soluble with mixed oxide fuel in the molten state. However, some refractory materials might be expected to react chemically with sodium. Silica-containing materials, in particular, are expected to react exothermically with sodium. Also, materials containing water, such as concrete or hydrated minerals, would interact strongly with sodium. A series of commercially available firebricks were exposed to sodium and it was shown that a number of them have adequate compatibility with sodium, including boiling sodium [25,26]. These included bricks which were nearly pure magnesia (MgO) or alumina (Al_2O_3). Silica-containing materials were not satisfactory.

Magnesia, in addition to being resistant to attack by sodium, has a high melting point, a high heat capacity, and a high heat of fusion and was considered to be especially promising as a sacrificial barrier material. Moreover, magnesia and UO_2 have been shown to be soluble in the molten state [27]. Accordingly, magnesia was specified as the material for the proposed CRBR parallel design concept [3] (see Fig. 3).

Analyses of the rates of penetration of core debris into beds of magnesia brick are complicated by the fact that the magnesia melting at the boundaries is less dense than the fuel pool. Consequently, additional convective effects are introduced at the boundaries because of density gradients. A study of these effects using simulant materials indicated that the downward penetration was greatly enhanced when the melting material is less dense than the pool [28]. This is shown by the differences in the transient pool shapes and penetration distances apparent from the experimental results in Fig. 8. Although there have been several approximate treatments, there is no accepted correlative or analytical procedure to take account of the perturbing effects of the mass transfer on the overall penetration rate. The magnitude of the effect is apparent in Fig. 9 where calculations of the growth of a molten fuel pool into a bed of

Fig. 8. Cross Sections of Aqueous Salt-Solution Pools after 50 min of Internal Heating on a Bed of Water Soluble Wax (Density 1.15 g/cc) for Various Values of Initial Pool Density [28].

Fig. 9. Pool Diameter and Penetration vs Time for a Sacrificial Bed of Magnesia by Two Versions of GROWS (dashed lines refer to case where effects of mass transfer were ignored; solid lines refer to case where mass transfer effect was included).

magnesia are plotted [29]. The dashed curves correspond to the case where the effects of mass transfer were ignored and the solid curves include an approximate treatment of the effect based on a heat transfer-mass transfer analogy. The results show that the extent of downward penetration depends strongly on the model used to take account of the mass transfer effect. However, the results, based on the total core inventory for FFTF, also show that the pool diameter increases from an initial value of 2 m to 3 m in about 16 hr while the downward penetration is less than 400 mm. These penetration rates are typical of the kind of

performance which might be expected from the sacrificial barrier concept.

One way to avoid uncertainties about the mass transfer effect is to employ a sacrificial material of density greater than that of the fuel. Such materials include urania and thoria. The current design of the core retention system in the SNR-300 does call for UO_2 bricks as a liner material. However, considerable cost is involved in using large quantities of these materials. Also, there is no available capacity for fabricating large quantities of UO_2 or ThO_2 into bricks. It would appear that available materials such as magnesia might be equally suitable and far more economical.

Stable Barrier Concept

In this concept, a refractory material is used which is capable of containing the core debris even without cooling. To do this, the refractory material must not melt or be attacked by core debris up to temperatures equal to the boiling point of fuel. Both graphite (and several carbides) and tungsten have sufficiently high melting temperatures. Tungsten or other refractory metals are not promising materials because they would be attacked rapidly by the molten steel content of the core debris. Graphite is a promising material for this application even though there are chemical reactions between graphite and core debris. Refractory carbides such as tantalum carbide might have better chemical resistance than graphite; however, the cost of such materials would be excessive.

Graphite can form chemical compounds with sodium. However, tests have indicated that high density graphite is not attacked by liquid sodium, including boiling sodium, for significant periods of time [25].

Molten steel dissolves considerable quantities of carbon to form a mixture with a eutectic melting temperature about 1150°C . The $(\text{U},\text{Pu})\text{O}_2$ of the fuel will react to form mixtures of $(\text{U},\text{Pu})\text{C}$ and $(\text{U},\text{Pu})\text{C}_2$. These carbides also tend to form double carbides with the steel compounds such as $(\text{FeU})\text{C}_2$. The principal problem with these interactions is the conversion of the oxide content of the fuel into carbon monoxide gas. A recent study of the interactions of graphite with UO_2 and steel indicated that rapid reaction began for the graphite- UO_2 system at about 2400°C where a molten carbide is generated [30]. In the presence of steel, the temperature for the onset of rapid reaction was reduced to about 2000°C . At these temperatures, substantial conversion of the oxide fuel to carbides and the release of CO occurred in less than 15 min. The magnitude of the gas release was a principal

reason for not using graphite as the liner material in the SNR-300 reactor (see Fig. 2). The current design employs UO_2 bricks as the liner material [2].

Although CO generation is a disadvantage of the graphite barrier concept, there are significant advantages. Once the reactions are completed, the heavy metal carbides reach a chemical equilibrium with the underlying graphite and a stable containment situation should be achieved. Also, graphite is relatively inexpensive. The high thermal conductivity of graphite would tend to prevent localized regions of high heat flux and facilitate design of heat removal systems.

Catch Tray Concept

In this concept, a system of trays is arranged so as to intercept the downward motion of the core debris. The trays may be immersed in sodium initially or reliance must be placed on the expectation that adequate sodium will be available from the primary system at the same time as the core debris begins entering the cavity. The catch tray concept is especially promising for pool-type reactors where space is more available for a tray system that is submerged in primary system sodium within the reactor vessel. A tray system, designed for collection of a fraction of the total fuel inventory, for the Super-Phénix reactor is indicated in Fig. 10.

Fig. 10. Cross-section of Super-Phénix Reactor.

The downward flow of molten core debris through sodium is expected to result in quenching and fragmentation into particulate form, as noted previously.

Core debris entering trays would be in particulate form. However, the performance of a tray system does not necessarily depend on the completeness of particulate formation. Core debris in the form of molten material or large solid pieces would also be intercepted by a tray system. Since the trays are cooled by sodium from the underside as well as from above, there is no need to depend upon infiltration of the debris bed by sodium.

Heat removal from the tray system is facilitated by the arrangement of the trays to achieve vigorous natural convection heat transfer to the sodium. Heat removal from the sodium pool at higher elevations is needed to maintain the convection process and afford an ultimate heat sink.

INHERENT RETENTION CAPABILITY

The C&R Project has contended that a cooled core-retention system in the reactor cavity is not necessary because of the inherent retention capability of the concrete base mat below the reactor cavity. This capability is one of several features referred to as "third level thermal margins". The third level refers to the third level of reactor safety [31] and the thermal margins refer to the fact that the processes which would function to mitigate the radiological consequences of HCDAs occur spontaneously. One of the expected processes is the entombment of the non-volatile core debris within the concrete of the base mat. Because of the CRBR approach, the interactions of core debris with concrete have come under intensive study since about 1975. Core debris interactions with concrete are reviewed in the following paragraphs.

The general features of the behavior of concrete at temperatures up to the order of 1000°C have been established by experiments concerned with the fire resistance of structures. Detailed specification of the behavior of concrete under transient thermal conditions are complicated by the large number of types and compositions of concrete. While the general features of dehydration and decarboxylation of concretes are known [32], means to predict detailed behavior for specific compositions were not available until very recently. Also, information on the behavior of concretes in contact with core debris is in a preliminary stage of development.

Transient Heating and Dehydration of Concrete

Transient heat transfer in concrete is a key aspect of core debris interaction processes. This is especially true because of concrete decomposition which leads to dehydration and gas re-

lease. The release of water vapor and other gases, which depends on the penetration of heat into the concrete, may greatly affect the composition of the core debris as well as the heat transfer processes within the core debris.

The water content of concrete can be considered in two parts. One part, called free water or evaporable water, is contained within capillaries and is largely released by heating to temperatures slightly above 100°C. The other part of the water content is more strongly bound and much of it is chemically bound within the cement phase. Bound water is released over a rather wide range from just above 100°C to about 900°C. The peak release rate generally occurs around 500°C, so that to a first approximation, bound water can be considered to be released at ~500°C.

Limestone concrete contains quantities of calcite, CaCO_3 , and dolomite, $\text{MgCa}(\text{CO}_3)_2$, which decompose to calcia, CaO , and magnesia, MgO , and release CO_2 gas. The decomposition of the carbonates occurs in the range 600 to 1000°C. Dolomite decomposition to MgO and CaCO_3 occurs in a temperature range near 700°C while the calcite decomposes around 900°C [33].

Transient heating experiments with concrete have been performed in several ways: (1) Constant rate heating at ~3°C/min [33]. (2) Exposure to a constant temperature of ~900°C at one plane boundary [34] and (3) Exposure to a constant heat flux ranging from 300 to 3000 kW/m^2 at one plane boundary [35]. There was no spalling or fragmentation observed in any of these experiments. However, sample dimensions were less than about 200 mm and none were strongly constrained nor did they have steel reinforcing. Surface spalling has been observed in fire tests with large sections [36]. In general, the spalling is confined to the material within a few centimeters of the heated surface.

The thermal conductivity of many gravel concretes, by a standard (steady-state) method [37], can be represented approximately [32,33]:

$$k = 1.95(1 - 0.000563 T, ^\circ\text{C}), \text{ W/mK}$$

which is the dashed line in Fig. 11. Concretes prepared from low conductivity aggregates can have conductivity values of the order of 0.5 J/mK.

Transient heating experiments with concretes have shown that for one-dimensional experiments, the results can be represented by a similarity variable which combines the effects of distance and heating time [34,38]. A similarity plot for a limestone concrete is shown in Fig. 12. It is apparent from Fig. 12 that the transient heating results could not be accounted for on the basis of a

Fig. 11. Thermal Conductivity of Gravel Concretes from Several Sources, (Points are from [32]; Solid Lines are from [34] and [42]; Dashed Line is from [33].

Fig. 12. Diffusion of Heat in Concrete at Elevated Temperature from [34].

single value of thermal diffusivity, independent of temperature. Instead, an empirical equation for an effective thermal diffusivity, α_{eff} , which varied from a constant high value at 25°C, α_0 , to a constant low value at temperatures above 800°C, α_s , was used:

$$\frac{\alpha_{\text{eff}}}{\alpha_0} = 1 - \frac{1 - \alpha_s/\alpha_0}{2} \left\{ 1 + \text{erf} \left[\frac{4 \left(\frac{T - T_0}{T_s - T_0} \right)^n - 2}{1} \right] \right\} \quad (1)$$

The results for two limestone concretes are shown in Fig. 11. Curve A corresponds to the silica-containing limestone concrete of Fig. 12 while Curve B refers to a limestone concrete containing flyash which corresponds closely to

the mixture specified for CRBR.

The high values of conductivity at low temperature (Fig. 11) obtained by the transient method may have resulted from the heat transport effect of water or water vapor moving away from the heated surfaces, while the low values at high temperatures could have been caused by the motion of steam toward the heated surface. In the experiments represented by Curve A, liquid water was rejected at the cold end of the sample. This did not occur for the concrete represented by Curve B. The low conductivity at high temperature could have been caused partly by the porosity created by the loss of CO₂ gas.

Equation (1) has been used in conjunction with an approximate analytical treatment of transient conduction which takes account of the heat absorption by evaporable water release at 100°C, bound water release at 500°C and CO₂ release at 700°C [34]. The model is capable of predicting the extent of gas release on the assumption that the three types of gases are released, each at a single value of temperature. The temperature values and the magnitude of the heat effects can be varied to obtain parametric results. A model of gas release has been proposed that is based on first order kinetic release of the three species based on an interpretation of thermo-gravimetric analyses of small concrete specimens [39]. Concrete heat transfer and gas release models in the form of finite element codes are in various stages of development.

Other concrete properties, including specific heat, density, and heats of decomposition can be evaluated from an analysis of the composition using the methods described by Harnathy [32].

Based on the methods currently in hand or under development, it would appear that the transient heat transfer and gas release for concretes can be calculated within a reasonable uncertainty provided small-scale experimental results are available for the specific compositions of interest. Changes in behavior which might be caused by cracking, spalling, or by the presence of reinforcing bars in large sections cannot be predicted adequately by available methods at this time.

Concrete Interactions with Sodium

The principal initial interaction of concrete with core debris is expected to be the reaction with sodium. Preliminary results are available from experiments in which hot sodium was poured on concrete samples [40,41]. Results have been obtained for interactions at horizontal surfaces, and at vertical surfaces. Penetration rates were about 12 mm/hr for basaltic concrete and about 25 mm/hr for high-density magnetite concrete. A rate

of about 12 mm/hr was found for a Tennessee limestone concrete for five hours of exposure to sodium very close to the boiling temperature [33].

The sequence of chemical reactions involved in the attack of sodium on concrete is complex and poorly understood. Experimental results combined with thermodynamic analysis have indicated which chemical reactions are likely to be most important. The water evolving from the concrete reacts with sodium in two ways:

Reaction (2) is expected to predominate in the presence of excess sodium. These reactions are major sources of heat during the interaction. The CO₂ evolving from the concrete can react with sodium oxide or with sodium as follows:

Reaction (4) is favored when sodium oxide is available in the melt. Reactions (5) and (6) are thermodynamically favored over similar reactions which would evolve CO. Elemental carbon has been found in the reactant mixtures in experiments [34]. These results indicate that little or no CO₂ or CO may be evolved so long as an excess of sodium is present above the concrete surface.

The most reactive of the concrete constituents is expected to be silica, SiO₂, or silica containing minerals such as basalt, which react to form a very stable series of silicates, including sodium silicate:

Concrete containing magnetite, Fe₃O₄, or ferrite Fe₂O₃, would be expected to react with sodium to generate FeO and possibly Fe. The principal reaction is:

The other concrete constituents, CaO, MgO, Al₂O₃, etc., would be expected to be soluble to some degree in the molten mixture of sodium oxides, carbonates, and silicates, along with ferrous oxide.

As long as sodium is present, the temperature is limited to about the boiling point of sodium (883°C). At this temperature, the calcia and magnesia content of the dried concrete surface does not melt and should not react with sodium. Because of this, it is expected that sodium attack on concrete would stop at some depth depending upon the quantity of

the sodium available to form the sodium oxide/carbonate mixture. Dissolution of the concrete constituents would stop once the mixture was saturated in calcium and magnesia. This solubility limit is consistent with available observations of a limited penetration; however, more experimental work is required before a satisfactory understanding of the process is available.

Concrete Interactions with Molten Steel

The steel content of the core debris should be in the form of metallic particulate which would remain intact as long as an excess of sodium is present. However, once the metallic sodium is largely boiled off, the temperature in the oxide layer would be free to increase and steel melting would occur at $\sim 1400^\circ\text{C}$. The molten steel is more dense than the molten oxide-carbonate mixture and would tend to form a layer separating the concrete from the molten oxide/carbonate layer.

The general nature of the interaction of molten steel with concrete has been established by induction heating of the steel and by pours of thermite-generated steel into concrete crucibles [34,35]. The continuous release of gases from the concrete causes a vigorous initial bubbling of the melt layer that tapers off with time as the gas release rate decreases. Gas analysis has shown that the evolving water vapor and CO_2 are efficiently reduced to H_2 and CO , respectively. When the gases are not shielded from air, a steady flame is produced. The oxygen lost from the gases oxidizes the steel to an Fe_3O_4 type of oxide mixture in which the nickel and the chromium contents of the stainless steel take the place of some of the iron. The oxide mixture in the experiments formed a growing overlying layer which appeared to be a single phase. This is shown by the experimental cross-section in Fig. 13. Under the reactor accident conditions, the upper layer would include the residual oxide mixture composed of sodium oxide, carbonate, and silicates. Whether this layer would be a single phase or several phases is not known.

The continuous generation of molten oxides of steel causes a continuous dissolution of the dried concrete constituents at a steady penetration rate from 70 mm/hr (Ref. 34) to as high as 1 m/hr. With basaltic concretes, an incipient melting region was observed between the concrete and the steel layer [35]. With a limestone concrete having very low contents of silica and magnetite (CRBR-type concrete), the region between the intact concrete and the melt appeared refractory and somewhat powdery [42]. In the latter case, concrete penetration was

Fig. 13. Cross-section from Molten Steel-Concrete Interaction Experiment [34].

probably by dissolution and not by melting. Recent tests have shown that the melting point of the dried concrete is much greater than the melt temperature in the experiments with CRBR-type concrete [42].

Concrete Interactions with Molten Fuel

As noted previously, fuel content of the core debris which would interact with the concrete of the cavity would be expected to be in particulate form. The dense particulate would tend to settle into the growing layer of molten oxides and carbonates. The $(\text{U,Pu})\text{O}_2$ fuel would be likely to dissolve in the layer, although no actual values of the solubilities are available so that it is uncertain just how long the dissolution process would take. In this situation, there would be no direct attack of pure molten fuel on concrete.

Some small-scale tests of the attack of molten UO_2 on concrete were performed to establish the general nature of the interaction process [34]. Electrically heated molten UO_2 was found to penetrate concrete at about the same rates as molten steel. The evolution of gas resulted in ejection of freezing droplets of a glassy material composed of solutions of UO_2 and the oxides of concrete. The solidified pools appeared to be an opaque glass having a density of the order of 3000 to 4000 kg/m^3 .

Inherent Retention Scenario

The downward flow of molten core debris from the failed reactor vessel is expected to result in an interaction with sodium producing a particulate bed of fuel and steel at the bottom of the sodium pool in the reactor cavity. The bed should spread over most of the available cross-section of the cavity. If it is assumed that the steel liner has failed, then the reaction of sodium with concrete will begin immediately. The heating of the concrete causes the release of water vapor and CO_2 gas. The water vapor will generate sodium monoxide according to Eq. (2) and the CO_2 will convert a portion of the sodium monoxide to sodium carbonate according to Eq. (4). These reactions should lead to the buildup of a growing oxide/carbonate layer between the sodium and the concrete. Limited penetration of the concrete will occur as the oxides of concrete dissolve in the oxide layer up to the solubility limit.

During the time of buildup of the oxide layer, the dense UO_2 and stainless steel particulate would tend to settle into the layer where some or all of the UO_2 , including the fission product oxides, would dissolve in the oxide melt. Most of the decay heat would be transferred upward to the boiling sodium which would gradually be driven out of the cavity as vapor. The distillation of sodium from the reactor cavity to the containment building is estimated to require about 10 days under the conditions of the CRBR cavity [4].

Following completion of sodium removal from the cavity, the decay heat generated within the oxide layer could no longer be removed at or near the temperature of boiling sodium. Consequently, the temperature would rise until sufficient heat loss could occur. When the temperatures reached slightly above the melting temperature of steel, $\sim 1400^\circ\text{C}$, a large heat sink would become available, corresponding to the heat of fusion of the reactor and guard vessels and other steel components within the cavity. The small-scale molten steel-concrete experiments have shown that the continued release of water vapor and CO_2 results in steel oxidation to FeO and Fe_3O_4 . These oxides would mix with the existing oxide pool. The continued growth of the oxide pool and the increased pool temperature would lead to a continuing penetration of the concrete base mat.

Concrete penetration would stop permanently when the heat removal capability in the upward and downward directions becomes equal to the decay heating rate at a pool temperature less than the melting point of the dried concrete. Upward heat removal is expected to occur by thermal radiation from the upper surface of the pool while downward heat removal

is by conduction through the remaining concrete, into the earth below the concrete. Both processes of heat removal are improved by radial growth of the pool which increases the heat transfer area. Detailed evaluation of the ultimate penetration capability depends mainly on achieving an improved understanding of the relative processes of pool growth in the sideward and downward directions. The high ratios of sideward to downward heat flux observed in water pool experiments under nonboiling and under boiling conditions (see Fig. 7) offer considerable promise that large radial growth will occur so that downward penetration will be quite limited.

The potential for concrete spalling is a significant uncertainty in penetration analysis. Available information indicates that spalling is limited to thin layers close to the heated surface. Also, spalled material would be melted into the pool and thereby absorb heat of fusion and contribute to the dilution of the heat source to the same degree as concrete which had not spalled. Spontaneous breakup of thick sections, which would have major effects on penetration predictions, appears to be very unlikely.

CONCLUDING REMARKS

Post-HCDA, core-debris behavior is considered in three locations: (1) within the reactor vessel, (2) within the cavity below the reactor vessel, and (3) within the concrete below the reactor cavity. In-vessel behavior is dominated by considerations of the fragmentation of molten fuel and steel on quenching in an excess of sodium to form particulate debris beds. In-vessel coolability depends primarily on the availability of sufficient horizontal surface area so that the particulate bed depth is not excessive and on the requirement that removal of small quantities of heat from the reactor vessel continues.

Core debris behavior in the reactor cavity can be controlled in several different ways. An uncooled cavity liner could maintain its integrity for many days while the decay heat is absorbed by sodium boiloff from the cavity. An engineered cooling system in the cavity could result in the permanent coolability of core debris in the cavity. Such a system could be based on one of four general concepts, including (1) the cooled crucible concept, (2) the sacrificial barrier concept, (3) the stable barrier concept, and (4) the catch tray concept. Each concept has a different set of advantages and disadvantages. However, each would appear to be a viable approach to core retention.

Core debris behavior within the concrete below the reactor cavity is a key element in the "inherent retention

concept". Evaluation of the concept depends upon development of an improved understanding of several phenomena, especially pool radial growth and concrete spalling. Based on current understanding, there is an expectation that core debris penetration into a typical reactor base mat would stop short of total penetration [4].

Recent studies of core debris behavior relative to all three locations have led to a great improvement in the understanding of the appropriate phenomena. It now appears that a range of options are available to the reactor designers and the safety evaluators concerned with the retention and coolability of the debris that could be generated in the extremely unlikely event of a hypothetical core-disruptive accident in an LMFBR test reactor or power plant.

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