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CORE MELT MATERIALS INTERACTIONS EVALUATIONS

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Prepared by

Dr. D.G. Swanson
Applied Science Associates, Inc.
P.O. Box 214
Hawthorne, CA 90250

Contract Supplement
Prepared by
Dr. I. Catton

Prepared for

Division of Systems Integration
U.S. Nuclear Regulatory Commission
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NOTICE

ABSTRACT

This annual progress report describes work performed as part of consultative support to the U.S. Nuclear Regulatory Commission on a variety of topics related to the materials interactions that would occur following a postulated core meltdown accident in a Light Water Reactor.

The main topics addressed include (1) an evaluation of candidate core retention system materials for the Zion and Indian Point (ZIP) nuclear reactors, (2) an examination of various core retention system concepts with emphasis on the restrictions imposed by the conditions present in the ZIP reactors, (3) development of a concept for retrofitting the ZIP reactors with a molten core retention device, and (4) an examination of the existance of the temperature threshold for the generation of hydrogen in light water reactor systems.

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I. SUMMARY

During the first year of this contract, an emphasis has been placed on establishing the feasibility and cost-effectiveness of concepts that could be used in retrofitting the Zion and Indian Point (ZIP) nuclear reactors with a core retention device. The purpose of this device is to prevent containment failure by basemat penetration and mitigate the potential for over-pressurization failure. Several problems greatly reduce the number of available options. First, the reactors have operated for some time and, consequently, the residual radioactivity in the region below the reactor vessel is quite high, even if the reactor fuel is removed. Second, instrument guide tubes have been installed which enter the reactor vessel from its bottom. These tubes, which are designed to operate at high pressure, come to within approximately 9 inches of the cavity floor and, as a practical matter, cannot be altered. Also, the overall cavity free volume is relatively limited. Although development of a core retention device is difficult in view of such limitations, it is concluded that effective core retention can be achieved.

As the initial step in this study, a number of core retention materials were examined for their potential for use in retrofitting the ZIP reactors with a core retention device. Over 4000 materials were examined initially. Two important considerations in material selection were the material density, to avoid floatation in molten core debris, and the prevention of gas evolution during contact between the material and the molten core debris. Melting point, boiling point and cost were also important. Other considerations included toxicity, availability, the existence of fabrication facilities, and compatibility with water, air, and core debris.

When the particular requirements imposed by the environment in the ZIP reactor cavities are considered and used to eliminate candidate materials, remarkably few choices remain. Even if the criteria are relaxed (for example, by considering low density substances), there are few suitable materials. None are really outstanding, in that all of the materials have drawbacks.

Based on the evaluations contained herein, it is concluded that the best of the available materials for use in retrofitting the ZIP plants is thoria. It will not float in molten core debris, it is compatible with water, and it has a high melting and boiling point. The material does not evolve gas and is the most chemically inert material of any studied in a molten core debris environment. Its disadvantages include a moderately expensive raw material ($16.40/kg) and substantial fabrication costs. It is very mildly radioactive but is in the hands of the public in the form of gasoline lantern mantles. There is a very large potential supply at the W.R. Grace Co. in Tennessee that may be available at lower cost.

At the next stage in this study, different reactor retrofitting concepts were examined. In selecting a concept for retrofitting a reactor, it is desirable to reduce the exposure of workers to radiation to a minimum. This means that the installation time must be as short as possible. Other problems, as noted above, include a very limited available space which contains instrument guide tubes that cannot be altered. It is also desirable that the concept selected offer some protection in the event of any core retention system or containment cooling system failure. It should be resistant to jet penetration by core debris and offer as little prospect as possible of a vapor explosion and the generation of gas, particularly hydrogen.

A number of core retention concepts are presented and examined from the point of view of their feasibility and cost effectiveness for use in retrofitting the ZIP reactors. The comparison of core retention concepts presented here is applicable only to the specific environment of the ZIP reactors. In other environments, the concepts would rank and compare differently.

The best core retention device concept examined for the ZIP application was a thoria rubble bed flooded with water. It can be rapidly installed by pouring pellets of the material in place with minimum radiation exposure to personnel. Water can be forced through the bed with minimal difficulty. There are adequate supplies of thoria and facilities are available for its fabrication. The expected cost of implementation is not unreasonable.

It is important to note that the system will provide protection against concrete attack and penetration for a time even if water is not initially available in the cavity, or if the arrival of water is temporarily delayed. Most importantly, the flooded thoria bed concept should accomplish the objective of core retention in terms of preventing core-concrete interactions and, consequently, should mitigate containment pressurization, hydrogen and other combustible gas generation, and the potential for basemat melt-through.

The concept does have several potential disadvantages as is the situation with all of the choices examined. There are questions involving iron oxide-thoria interactions and the response of the rubble bed to jet impingement and fuel-coolant interactions. Another important problem concerns possible hydrogen generation following contact between molten iron and water in the flooded bed. However, hydrogen control in the containment would probably be required for other reasons. It should be noted that hydrogen will not be generated by interactions between water and iron if the iron can be cooled below 550°C by water.

Considering the above potential disadvantages, some experimental research is needed to confirm the viability of the concept proposed here. Information is needed on the interactions between thoria and core debris materials. In particular, it would be desirable to know whether thoria and iron oxide can form a low-melting eutectic. It is of interest to determine if jet impingement or melt streaming will disrupt either loose gravel of a certain pre-determined shape, or gravel held in place by a restraining structure. Also, the effect of a fuel-coolant interaction on a restrained and an unrestrained gravel bed is of interest. Another question concerns the potential for a melt trickling through a porous gravel bed. The rate of hydrogen and other gas generation should be determined from experiments with concrete cavities protected by a thoria gravel bed. The preceding experiments should be done with both dry and wet gravel beds. The tests should also examine the sustained interactions of corium melts with gravel beds to determine the heat flux partitioning that results.

With the performance of the above limited confirmatory research, it is concluded that the proposed flooded thoria bed concept is feasible and has the greatest potential for success of any concept considered for retrofitting the ZIP reactors. This concept, coupled with a hydrogen control system, will reduce significantly the radiological consequences and risks by preventing containment failure by basemat melt-through, and by mitigating the potential for failure by either overpressurization or hydrogen burning.

Future studies should include an examination of multi-layer systems. The utilities always have the option of proposing other effective means of mitigation.

The potential for hydrogen generation in a reactor system as well as in a flooded thoria rubble bed is also discussed separately in a later section of this report. It was concluded that there will be a temperature threshold for the generation of hydrogen from the oxidation of steel by water. Below 550°C, the amount of hydrogen generated by reactions between water and steel should be negligible. Additional study is desirable to determine the extent of hydrogen generation.

II. INTRODUCTION

This is an account of work performed for the Office of Nuclear Reactor Regulation (NRR) of the United States Nuclear Regulatory Commission. In the following sections, considerable emphasis will be placed on an examination of the feasibility of retrofitting the existing nuclear plants at Zion and Indian Point (ZIP) with a core retention device. This report is one of two reports prepared for NRR concerned principally with the potential for core retention at the ZIP plants. Other work is being performed at UCLA. The retrofitting problem is examined here principally from a materials point of view. The UCLA effort is concerned principally with the heat transfer aspects of core retention devices.

The discussion will commence with a postulated core meltdown event where penetration of the reactor vessel takes place. Consideration is given to the consequences of core debris attacking the concrete basemat. Various core meltdown scenarios are considered, including having the reactor cavity initially dry or flooded with water.

The report continues with a discussion of the available materials for core debris retention, given the constraints imposed by the conditions present in the ZIP reactors. This environment includes a high radiation level in the reactor cavity, which makes it impossible for personnel to work in the cavity for extended periods of time. Another problem is the presence of instrument guide tubes which occupy much of the available volume and cannot be moved without great difficulty since they are designed to operate at the same pressure as the reactor coolant system. These tubes come at one point to within nine inches of the reactor cavity floor. Also, the reactor cavity free volume is relatively small, especially in the Zion plant, thereby eliminating certain core retention concepts from further consideration.

Many of the materials options available for core retention in future nuclear power plants are not feasible for use in retrofitting the existing ZIP reactors for the reasons just discussed. Thus, the emphasis in one section of this report will be on the desirable characteristics of materials that can be used to retrofit an existing reactor with a core retention system. Those issues which were considered in the selection of a material for retrofitting the ZIP plants with a core retention device are listed below:

- Rapid installation time due to the radiation hazard
- Limited space for installation
- High density to avoid floatation
- High melting point
- Limited gas evolution to mitigate containment overpressurization
- High boiling point
- High specific heat and heat of fusion
- Thermal conductivity selection
- Miscibility of the fuel and refractory layer
- Chemical compatibility with core debris
- Low cost and high availability
- Resistance to thermal shock
- Resistance to mechanical shock
- Preaccident stability
- Toxicity
- Radioactivity

In a later section, various core retention devices are examined for the ZIP plants. All of the concepts considered include some provision for protecting the concrete basement with a refractory or sacrificial material and for water cooling. Calculations of the containment response show that core debris decay heat must be removed by a coolant, either directly or indirectly, and the reactor cavity concrete must be protected in some manner if overpressurization of the containment is to be avoided. Otherwise overpressurization will occur in approximately eight hours.

As a result of the above considerations, a flooded thoria rubble bed is considered to be the most promising candidate for implementation as a core retention device in the ZIP plants. The device can be installed quickly in a limited space without undue radiation exposure to personnel. It appears that it will achieve the goal of core retention. The concept is discussed in considerable detail in the following sections.

In addition to core retention concepts utilizing refractory materials, concepts have also been studied that employed low density materials, castable materials, and low temperature sacrificial materials. Therefore, our consideration was not limited exclusively to concepts employing refractories. The core retention device concepts discussed in the report include:

- Flooded thoria rubble bed
- ThO_2 over cooling coils
- UO_2 over cooling coils
- High alumina cement (HAC) with cooling coils
- MgO over cooling coils
- ZrO_2 over cooling coils
- Graphite over cooling coils
- Borax bath
- Iron oxide over cooling coils
- Lead bath over cooling coils

- Copper bath over cooling coils
- Uranium metal bath over cooling coils
- HAC/ThO_2/HAC (multi-layer system)

The preceding ZIP related activities constituted the effort made under Tasks II and III of this contract. The results are presented in Sections III through VIII of this report.

Another section (Section IX) of this report is concerned with the conditions required for the generation of hydrogen under core melt accident conditions in a reactor. This work was performed under Task IV of this contract. The existance of a temperature threshold for hydrogen generation is discussed.

An extensive trip report covering European travel to discuss core melt phenomena and protection against severe accidents is the subject of Appendix E. Other travel has been reported in our monthly letters and represents the effort under Task VI.

With regard to Task I, work will commence once further information on the Floating Nuclear Plant core ladle design is submitted by the applicant, Offshore Power Systems. Consequently, no effort has been made as yet under this task.

With regard to Task V, materials related information has been provided to the Brookhaven and UCLA programs on post-accident containment analysis as required. Additional activities will commence during the second contract year, as proposed under our program plan of June 1980.

A program plan for the second contract year appears in Appendix D. Work performed under a Supplement to this contract is the subject of Appendix C. The study was conducted by Dr. I. Catton. Additional work in the same area will be the subject of a report to be published by UCLA at a later time.

III. PENETRATION OF AN UNPROTECTED BASEMAT BY CORE DEBRIS

Reactor vessel failure resulting from a core melt can lead to molten fuel, clad material and steel entering the reactor cavity at high temperatures. The velocity of the molten material will be on the order of 10 meters per second when it impacts the cavity floor if it is falling only under the influence of gravity. The rate at which molten material flows out of the failed vessel is not known very well. In some core melt accident scenarios, a gradual melt through is expected whereas in others the entire molten core is dumped at once.

Various experimental investigators[1,2,3,4,5,6,7] have estimated that the penetration rate of molten urania and steel into concrete in an initially dry reactor cavity can be as high as 1-4 ft/hr. Penetration rates calculated by different computer codes for prototypic meltdown scenarios are considerably smaller but are still significant. Furthermore, there is evidence that a substantial fraction of the total penetration may occur after the molten core debris has solidified since the frozen core debris will persist at temperatures above the melting point of concrete for some time.

A major problem expected from the attack of molten core debris on concrete is the very large quantity of gas generation. When molten core debris attacks concrete, water vapor is driven out of the concrete and CO_2 is generated from the decomposiion of any limestone that may be present in the gravel, sand, or cement. Water vapor can oxidize metal in the molten core debris forming hydrogen. In the presence of molten iron, CO_2 will react with hydrogen to form hydrocarbons, such as methane, and will also form CO. These gases are undesirable because they contribute to the pressure level in the containment and because several of them (H_2, CO, CH_4) are combustible.

The extent of basemat attack and the quantities of gas generated are scenario dependent. Several reactor vessel failure modes are depicted in Figure 1 (from Reference (8)). In each of the different accident scenarios, there are substantial differences in the initial temperatures of the molten materials. The molten material interaction with the concrete during these initial stages will be a strong function of its temperature and the mode of failure. Another important factor in the interaction will be whether the cavity is initially dry or wet.

If the molten core dribbles out of the vessel, its temperature will be near the melting point of steel, not far above the solidus. Under these circumstances in a dry cavity, freezing will take place relatively soon after contact with the concrete. On the other hand, the sudden failure mode, described by Mayinger[9] and assumed by the Germans (FRG) for their core melt accident analyses, can result in higher temperatures with the molten pool existing for some time before freezing. During this initial period, rapid penetration into the basemat will occur and substantial quantities of gas will be generated.

If the molten core debris initially dribbles from the reactor vessel into a dry cavity, the initial penetration will probably be by solidified material. The solid fuel debris will penetrate the concrete more slowly than a molten pool would because the heat transfer downward is less efficient. Nevertheless, solid core debris will decompose the concrete if its temperature exceeds approximately 1200°C, the solidus temperature of concrete. Preliminary experiments[10] at Sandia Laboratories indicate

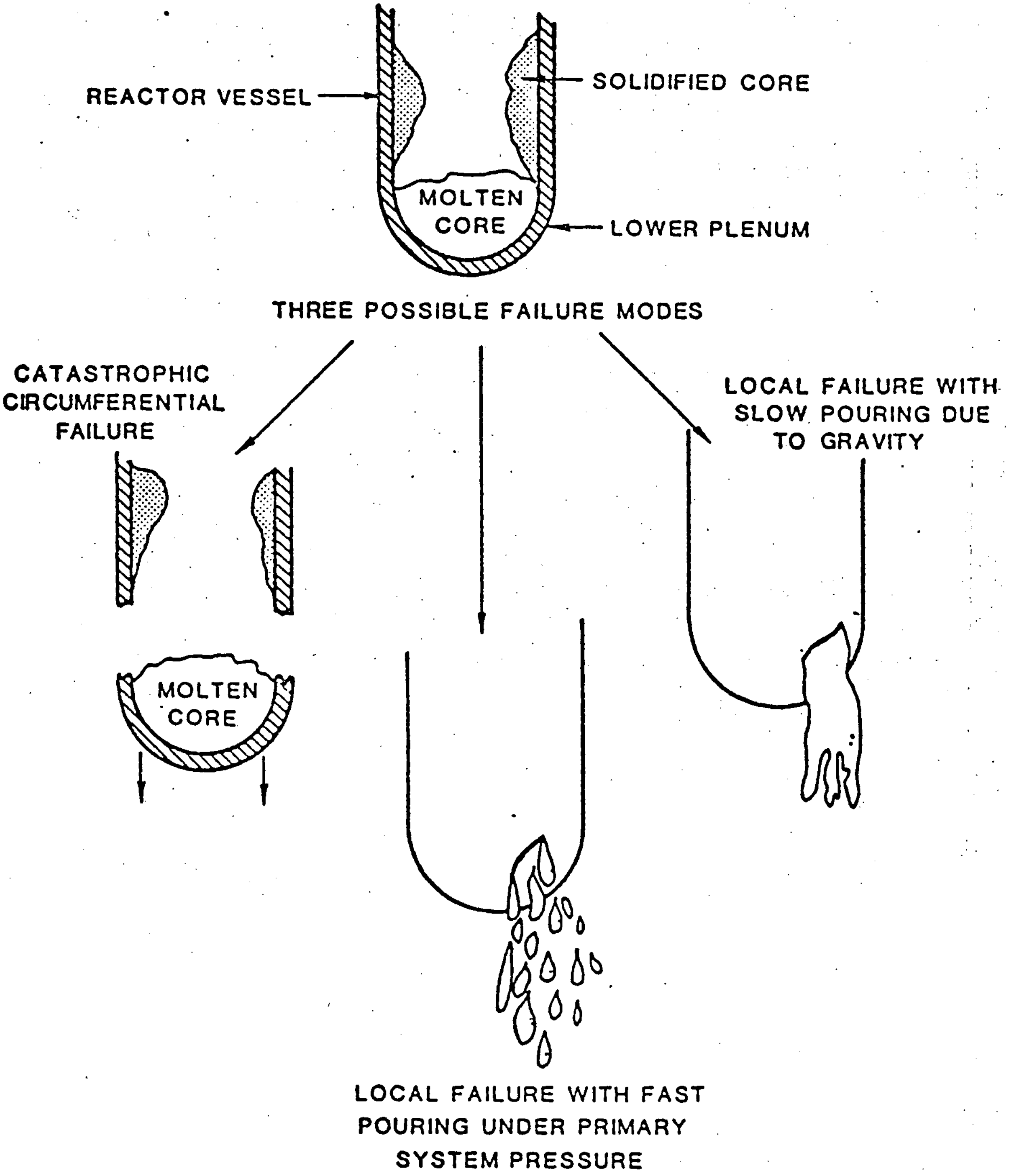


Figure 1. Various Possible Modes of Lower Plenum Failure

that solids will decompose the concrete and sink into the resulting mixture of liquids and gases. Once this process has initiated, the heat transfer to the concrete will increase. The foamy mixture will ooze through the debris bed and freeze on top forming a thick crust. This is a serious situation because water entering the cavity at a later time will serve no useful purpose in mitigating core-concrete interactions and basemat penetration will be much more likely.

The Sandia experiments with hot solids investigated two limiting cases. One of the experiments represented a solid chunk of fuel and employed a 3.125 inch diameter by 5 inch long cylinder at a temperature of approximately 700°C. The solid chunk of steel caused the concrete to decompose and be displaced as the solid mass sank into the concrete. When the cylinder was submerged, a crust formed and it continued on its downward path. The gas-liquid mixture resulting from the concrete decomposition process had the fluid properties of thin maple syrup.

In the other limiting case, a heated debris bed was represented by heated nails. This type of debris bed might result from the initial decomposition process or from debris bed dryout. As in the preceding experiments, the mixture of gas and molten concrete oozed through the debris bed and the nails eventually were submerged and covered by a crust formed by the melting concrete.

Preliminary experimental results obtained by Peehs (FRG) show that water has virtually no effect on either molten pool penetration into concrete or on the associated gas production. This is attributed to the formation of a thick crust at the upper surface of the molten pool and the creation of a void between the melt and crust. The void acts as insulation between the melt and crust and results in isolating the molten pool from the water available for cooling.

If the reactor vessel fails suddenly and a large molten pool forms in the cavity, the attack on the concrete will be much more vigorous. During the early stages of attack, the escaping gases will induce substantial splashing which will lead to attack on boundaries above the pool. The high initial heat transfer will probably cause freezing of the molten pool, perhaps into a solid chunk or porous mass. In any event, decomposition of the concrete will follow with the molten concrete-gas mixture flowing around the solid chunk or around and through the porous mass, as depicted in Figure 2 (from Reference [8]). The initial penetration rates will be much higher than in the previously discussed case and, depending on the molten pool cross-sectional area, may be more or less during subsequent times. Late entry of water into the cavity will probably not be helpful because the molten concrete will form a thick crust over the core debris with an intervening void space that inhibits heat transfer as discussed above.

If water enters the cavity before reactor vessel failure, the molten core debris will be fragmented. Water may be present initially in the reactor cavity, or may be dumped into the cavity by the accumulators. The resulting fuel-coolant interaction will probably generate relatively small particles (mean dia. ~200 μm). The extent of dispersal throughout the reactor cavity is uncertain. Even if the debris spreads out over the entire floor of the reactor cavity, a relatively deep bed (1 to 2m) will still be formed. There is insufficient data on debris fragment sizes, extent of dispersal and deep bed dryout phenomena

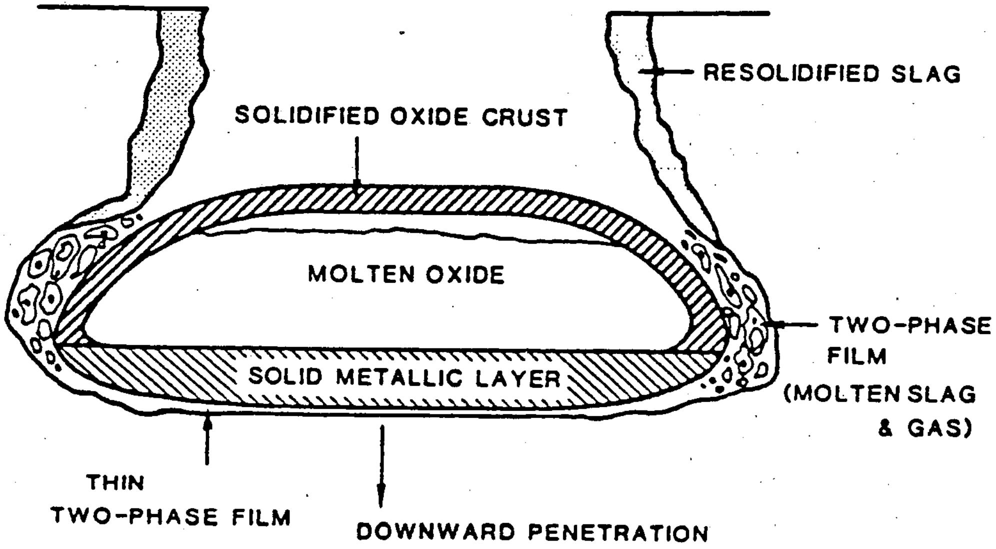


Figure 2. Solidified Core in Concrete After Meltdown

to guarantee that the core debris would be permanently coolable under a wide spectrum of core melt scenarios, particularly those involving the complete loss of AC power and the ultimate heat sink. Therefore, dry out and remelting of the core debris, accompanied by subsequent molten core-concrete interactions, has to be considered.

Penetration of the containment basemat by core debris cannot definitely be established or ruled out at this point in time. Clearly, more experimental information on the long-term behavior of core-concrete interactions is required before accurate estimates of the time required for melt-through of the containment basemat can be made.

Predictions have been made in this country and in Germany for both basemat penetration and containment overpressurization. In calculations performed by Murfin,[11] the principal parameters of interest were gas release, aerosol formation, and penetration time for the melt. It was found that if water precedes the melt, penetration may not immediately occur, although local or temporary dryout could lead to an irreversible situation where penetration into the concrete basement could eventually result. Once penetration commences, 3 or 4 days will be required to reach the bottom of the basemat.

Rates of 2-3 cm/hr were observed in the Sandia experiments when the debris temperature reached 1200 -1300°C. The decay power has an important influence on the rate and extent of penetration. Also, the heat required to decompose concrete is a weak function of the type of concrete and, as a result, the solid penetration mode is probably a function of concrete type. An important factor in the upward heat transfer is the extent of molten pool crust formation on the top surface. At present, it appears that either molten or solid debris can penetrate the basemat. Further, addition of water will probably not improve the situation if the attack on the concrete has already commenced. Finally, even if water enters the cavity before vessel failure, fragmentation and subsequent debris bed dryout can lead to situations where penetration of the concrete basemat will result.

Although penetration of the containment basemat is of concern, particularly for those plants that have unfavorable or adverse liquid pathway conditions, the primary concern is the threat to the above-ground containment failure caused by the interactions of molten core debris with concrete. Overpressurization failure of the containment building due to the gases generated through core-concrete interactions is expected to occur in approximately 18 hours.

Consequently, basemat penetration and containment overpressurization can only be precluded with confidence if concrete attack is not allowed to occur. This requires protection of the basemat from solid or molten core debris if reasonable assurance is to be given.

Since core-concrete interactions can cause serious threats to containment failure because of basemat melt-through, overpressurization, and hydrogen burn, core retention devices are being examined for the Zion and Indian Point (ZIP) plants. These devices would retain a molten core within the confines of the containment building and would prevent the interaction of core debris with concrete, so that non-condensible and combustible gas generation would be precluded. Although there are serious consequences from core-concrete interactions in the above-ground containment failure modes, the public confusion and potential hysteria that would develop if the reactor owners and Government officials could not guarantee that the core debris would be contained within the confines of the containment building should not be underestimated.

IV. RETROFITTING CONSIDERATIONS

Core retention systems that can be installed into existing nuclear plants are much more difficult to design than systems for new or proposed plants, such as the Floating Nuclear Plant (FNP). Most designs that have been suggested in the past for core retention devices involve fairly massive structures. The restrictions on retrofitting preclude installation of such massive structures. Smaller structures involving active device cooling present problems both because of their complexity and because the loss of onsite power for periods of hours could interfere with device cooling. Retrofitting is most difficult because:

(1) The volume of the reactor cavity is relatively limited.
(2) Existing hardware in the reactor cavity is not easily moved or altered. The hardware severely limits access to the cavity and reduces the space available for a core retention device.
(3) Alteration of the cavity configuration is neither easy, inexpensive, nor welcome by the utilities since it could compromise the seismic rating of the plant.
(4) Radiation levels in the reactor cavity are high enough that restrictions on worker access to the cavity preclude complex installations.

In this report, consideration will be given specifically to the Zion and Indian Point (ZIP) reactors. These plants are of interest because of their close proximity to Chicago and New York, respectively.

In the ZIP reactors, most of the problems associated with retrofitting these plants with core retention systems are ultimately traceable to the radiation environment. If the radiation levels were lower, alteration or modification of cavity hardware would be feasible. Most of the radiation in the cavity arises from activation of the concrete. Consequently, the installation of shielding under the reactor vessel will not significantly reduce radiation levels.

The radiation hazard greatly reduces the number of available options for retrofitting the ZIP plants. Since workers cannot remain in the radiation environment of the cavity for extended periods of time, it is necessary to eliminate from consideration an entire category of materials that are otherwise very attractive. Low density refractories, such as MgO and ZrO_2, would be the preferred choices for this application in the absence of the radiation problem. These materials are resistant to penetration by molten core debris, do not generate gas, and are inexpensive. Unfortunately, they must be mechanically restrained because of their low density in order to prevent loss of the refractory layer by floatation. This restraint could be provided by forming bricks of the refractory into an inverted arch, including interlocking of bricks by a double tongue-in-groove design. However, construction of an inverted arch structure requires that people work for long periods in the cavity and this is not possible due to the radiation environment. Furthermore, these systems would require relatively large volumes of material and their installation, consequently, would necessitate relocation of the instrument tubes.

In the Zion and Indian Point reactors,[11] physical access is limited by the presence of a large number of instrument tubes in the reactor cavity. The tubes occupy a large volume and come to within nine inches of the floor. Accordingly, there is very little space available in which in to install a core retention device. The available information suggests strongly that, as a practical matter, these tubes cannot be moved. Welding new instrument conduit to the reactor vessel is probably impossible due to the need to work on the inside of the vessel. If, instead, the existing conduit is cut apart, it is estimated that at least two hours would be required to weld each new conduit to the existing conduits. However, it should be possible to change the method of support for the conduit. Currently, the instrument tubes are supported by metal structures attached to the floor of the reactor cavity. It should be possible to modify the instrument conduit supports so that they are attached to the sidewalls and ceiling rather than to the floor. The tubes do not require cooling and, consequently, it may be possible to pack material around them. However, the tubes cannot be locked into place rigidly as space must be allowed for thermal expansion.

Since the available space is limited, dilution of a molten core in a large volume of material is presumably impractical. Consequently, the emphasis here will be on sacrificial bed core retention systems rather than dilution systems utilizing a sacrificial bed.

A special class of low density materials, the castable concretes, which form a monolithic layer, could conceivably be used. The material could be pumped in and would be inexpensive. For reasons to be discussed later, these materials are regarded as unsuited for this application. The concrete would lock the instrument tubes in place and thus would not allow for thermal expansion. In addition, the concrete evolves water and gas on being heated and melts at a relatively low temperature.

V. DESIRABLE CHARACTERISTICS OF CORE RETENTION MATERIALS

Many of the options available for molten core retention in newly constructed plants are not feasible for use in retrofitting the existing ZIP reactors. The emphasis in this section will be on the desirable characteristics of materials that can be used to retrofit a reactor with a core retention system. Some of the concerns which must be considered in the selection of materials for retrofitting the ZIP plants are indicated below:

Rapid installation. The high radiation level severely limits the the length of time that each person can work in the reactor cavity. Consequently, it is necessary to limit the selection to materials that can be rapidly installed. This requirement precludes the construction of complicated structures (such as an inverted arch) requiring precise placement of materials.

Limited space. The selection of a core retention concept is also complicated by the restricted space available. Much of the cavity is filled with instrument guide tubes which are constructed to operate at a high pressure and cannot be moved without great difficulty. In addition, the limited volume of the cavity precludes the use of materials that would contain the molten core by dilution. Thus, core containment must be achieved through the use of refractory materials.

High density. The molten core debris density may be as high as 8.0 g/cc. Consequently, it is desirable to employ core retention materials with densities greater than 8.0 g/cc in order to prevent floatation of the core retention material which would leave the concrete floor unprotected. The rapid installation requirement seems to eliminate the use of low density materials in an inverted arch configuration. Low density materials that can be cast as monolithic layers are a possibility providing they do not introduce other problems.

High melting point. This reduces the rate of advance of the melting front of the core debris into the sacrificial bed. It is desirable to select a core retention material with a melting point greater than that of the corium to reduce penetration since the volume of material that can be used is limited by space considerations. If a lower melting material is used, then any cooling system must be able to withdraw heat quickly from the sacrificial material to prevent penetration of the layer.

Limited gas evolution. Combustible gaseous reaction products, such as CO, H_2, and CH_4 should not be generated by chemical reactions between the molten core debris and the core retention material. These combustible gases, plus CO_2, also increase the pressure in the containment and can increase the sparging of fission products from the core melt debris.

High boiling point. If the boiling point of the core retention material is substantially less than the melting point of the molten core debris, substantial quantities of vapor will be generated which can contribute to containment pressurization.

High specific heat and heat of fusion. These properties reduce the rate of advance of the melting front by absorbing heat from the molten core debris.

Thermal conductivity. A low value will reduce the rate of advance of the melting front by slowing the rate of transmission of energy into the sacrificial bed material. In some concepts, a high thermal conductivity may be desired so that the decay heat can be removed by a cooling system.

Miscibility of molten fuel and sacrificial layer. If the fuel and sacrificial material are miscible, the volumetric heating rate of molten fuel will be decreased by dilution.

Chemical compatibility with molten core debris. The core retention material should not react chemically with molten UO_2 or steel. It is especially important that no exothermic reactions occur with the molten core debris.

Low cost and availability. The material should be relatively inexpensive and it must be possible to fabricate it into mechanically stable structures, such as bricks or pellets, at reasonable cost.

Resistance to thermal shock. A desirable characteristic of the sacrificial material is that it be sufficiently resistant to thermal shock so that it does not fail mechanically when exposed to the rapid heating transient of a core meltdown accident.

Mechanical shock. The material should be resistant to mechanical shock from falling objects.

Preaccident stability. The core retention material should be stable in the environment of reactor cavity for the expected life of the reactor. In particular, the material selected must be stable in water.

Non-toxic. The material selected should not be chemically toxic.

Nonradioactive. Ideally, the material selected should be non-radioactive or mildly radioactive.

VI. MATERIAL SELECTION

A. Methodology

As we have seen, the radiation environment in the ZIP cavity limits the time that personnel can work in the reactor cavity and restricts the potential for modifications of hardware. Clearly, it is desirable to select a material that can be poured in granular or chunk form onto a reactor room floor to retrofit a reactor with a sacrificial layer. Such a layer might be economically installed with minimum exposure of personnel to radiation, if a suitable material is available at reasonable cost. The disadvantage of this kind of layer, as compared to individually placed bricks of a refractory material, is its susceptibility to the channeling of molten core debris between the granules or chunks. Also, since the material is unrestrained, either jet impingement or a steam explosion might sweep it away unless preventive measures are employed.

Since these granules or chunks would be unrestrained, the first priority in the selection of a material is a density at least as high as that of the UO_2 fuel. It is impractical to construct a layer of unrestrained low density material that could reliably withstand the mechanical and thermal shocks of a core disruptive accident without floating away in the molten UO_2.

UO_2 has a maximum density of 10.96 g/cc at room temperature and 8.7 g/cc at its melting point of 2860°C. When dilution of molten uranium dioxide with other molten core constituents, such as zironium oxide and iron oxide, is considered, a maximum probable maximum density for the molten core debris is approximately 8.0 g/cc. Consequently, the initial examination of core retention materials will focus on materials with densities greater than 8.0 g/cc. Other factors considered in the selection of a containment material were discussed earlier. These include melting point, chemical compatibility with the various materials in the core debris, suitable mechanical properties, stability in the normal reactor operating environment, and availability at reasonable cost.

The CRC Handbook of Chemistry and Physics[10] was the source of the initial list of inorganic and organic substances and provided the chemical and physical properties for all of the materials examined. None of the organic compounds listed in the Handbook were found to have the required thermophysical properties. Of the 4123 inorganic compounds listed, 124 materials were found to have the required density. These materials consist of 38 metallic elements and some of their compounds. These substances are listed in Table 1. Certain other materials which can be cast into monolithic structures may also be acceptable even if they have a low density. These materials will be considered separately in a later section.

An inspection of Table 1 shows that some elements and compounds have properties exceptionally well suited to containment of molten core debris. Rhenium, for example, has the required high density, most probably will not react with the core debris and is easily formed into structural shapes such as bricks. It is, however, a rare element and, as a result, is prohibitively expensive.

In estimating the relative costs of the elements and compounds in this table, the natural abundance of the

Table 1: Materials With Densities Greater Than 8.0 g/cc

Element or Compound	Density (gm/cc)	Melting Point (°C) if Known
Ac_2O_3	9.19	
AmF_3	9.53	
AmO_2	11.68	
Bi	9.80	271.3
Bi_2O_3	8.2-8.9	704-860
BiOBr	8.082	dec. 600
Cd	8.642	320.9
CdO	8.15	1500
Co	8.9	1495
Co_2As	8.28	950
$CoWO_4$	8.42	
Cu	8.92	1083
Cu_3Sb	8.51	687
Cu_3As	8.0	830
Cu_3B_2	8.118	
Dy	8.55	1412
Er	9.006	1529
Er_2O_3	8.64	
Au	19.3	1064
AuI	8.25	dec. 120
Au_2S_3	8.754	dec.197
Hf	13.31	2227

Table 1: Materials With Densities Greater Than 8.0 g/cc

Element or Compound	Density (gm/cc)	Melting Point (°C) if Known
HfC	12.20	3890
HfO_2	9.68	2758
Ho	8.7947	1474
Ir	22.421	2410
IrO_2	11.665	dec.1100
IrS_2	8.43	dec. 300
Ir_2S_3	9.64	dec.
$IrTe_3$	9.5	
Pb	11.3437	327.5
PbF_2	8.24	855
PbO_2	9.375	dec. 290
PbO	8.0-9.53	886
Pb_3O_4	9.1	dec. 500
Pb_2O	8.342	dec.
PbSe	8.10	1065
PbTe	8.164	917
$PbWO_4$	8.23	
Lu	9.8404	1663
Lu_2O_3	9.42	
Hg	13.5939	-38.87
Hg_2F_2	8.73-8.95	570
Hg_2O	9.8	dec. 100

Table 1: Materials With Densities Greater Than 8.0 g/cc

Element or Compound	Density (gm/cc)	Melting Point (°C) if Known
HgO	11.1	dec. 500
$HgCl_2 \cdot 2HgO$	8.16	
HgSe	8.266	Sub
HgS	8.10	Sub 583
Mo	10.2	2617
MoB	8.65	
Mo_2B	9.26	
MoC	8.20	2692
Mo_2C	8.9	2687
Np	20.45	640
NpF_3	9.12	
NpO	11.11	
Ni	8.90	1453
NiSe	8.46	
Nb	8.57	2468
NbN	8.4	2573
OsO_2	11.37	dec. 500
Os	22.48	3045
OsS_2	9.47	dec.
Pd	12.02	1552
Pd_2H	1076	dec.
PdO	8.70	870

Table 1: Materials With Densities Greater Than 8.0 g/cc

Element or Compound	Density (gm/cc)	Melting Point (°C) if Known
Pt	21.45	1772
$PtAs_2$	11.8	dec. 800
PtO	14.9	dec. 550
PtO_2	10.2	450
PtP_2	9.01	1550
PtS	10.04	dec.
Pu	19.84	3232
PuF_3	9.32	1425
PuN	14.25	
PuO_2	11.46	
Po	9.4	254
Pa	15.37	1600
Re	20.53	3180
ReO_2	11.4	dec. 1000
Re_2O_8	8.4	145
Rh	12.4	1966
Rh_2O_3	8.20	dec. 1100
Ru	12.30	2310
Sm_2O_3	8.347	
Ag	10.5	962
Ag_F	8.57	dec. 90
AgSe	8.0	880

Table 1: Materials With Densities Greater Than 8.0 g/cc

Element or Compound	Density (gm/cc)	Melting Point (°C) if Known
Ag_2Te	8.5	955
Ta	16.6	2996
TaB_2	11.5	3000
TaC	13.9	3880
TaN	16.30	3360
Ta_2O_5	8.2	1872
Tb	8.2294	1360
Tl	11.85	303
TlF	8.23	327
Tl_2O	9.52	300
Tl_2O_3	9.52-10.19	717
Tl_2S	8.46	448
$Tl_4V_2O_7$	8.21	454
Th	11.0	1750
ThC_2	8.96	2655
ThO_2	9.86	3220
Tm	9.3208	1545
W	10.35	3410
WB_2	10.77	2900
WC	15.63	2870
W_2C	17.15	2860
WO_2	12.11	1500

Table 1: Materials With Densities Greater Than 8.0 g/cc

Element or Compound	Density (gm/cc)	Melting Point (°C) if Known
WP	8.5	
WSi	9.4	900
U	19.05	1132
UB_2	12.70	2365
UC_2	11.28	2350
UH_3	10.95-11.4	
UN	14.31	2630
UO_2	10.96	2878
U_3O_8	8.30	dec. 1300 to UO_2
US	10.87	2000 +
Yb_2O_3	9.17	

elements in the earth's crust has been employed. The natural abundances of the materials under consideration are presented in Table 2.

It should be noted that the cost of some elements is influenced by other considerations. Some elements are by-products of certain refining processes and as a result are less expensive than their natural abundance would suggest. Others may require elaborate, expensive separation techniques or other processing.

Of the elements found in Table 1, the transuranic elements (Np, Pu, Am) are totally unsuited for the application considered here for obvious reasons. Actinium, polonium, and protoactinium are far too rare to be considered. Also, the last nine elements in Table 2 are far too expensive. Toxicity considerations also eliminate osmium and its compounds. Cost and availability considerations, therefore, eliminate the 15 rarest of the 38 elements considered. This removes 43 of the 124 candidate materials from consideration.

Another desirable characteristic for a containment material, particularly in the ZIP reactor environment, is a high melting point. One important constituent of the core debris is iron and the core retention material at minimum should be able to contain molten steel. If one takes the melting point of iron, 1535°C, as the absolute minimum temperature the containment material must withstand without melting or decomposing, many of the remaining candidate materials are eliminated. Table 3 lists the 28 materials which meet the requirements of high density (at least 8.0 gm/cc), a melting point of at least 1535°C, reasonable cost, and low toxicity.[8,9]

Table 2: Abundance of Elements as Found in the Earth's Crust

Element	Abundance (ppm)	Cost of Std. Brick (est.)
Nickel	80	
Copper	70	
Tungsten	69	
Niobium	24	
Cobalt	23	
Lead	16	
Molybdenum	15	
Thorium	12	$1000
Samarium	6.5	
Hafnium	4.5	>$1000
Dysprosium	4.5	
Uranium	4	
Ytterbium	2.7	
Erbium	2.5	
Tantalum	2.1	
Holmium	1.2	
Terbium	0.9	
Lutetium	0.8	
Thallium	0.6	
Mercury	0.5	
Bismuth	0.2	
Thullium	0.2	
Cadmium	0.15	
Silver	0.1	$5000
Palladium	0.01	
Platinum	0.005	
Gold	0.005	
Rhodium	0.001	
Rhenium	0.001	
Iridium	0.001	
Osmium	0.001	
Ruthenium	0.001	

When chemical compatibility with water and molten core debris at a high temperature is considered, a number of materials in Table 3 can be eliminated from further consideration. It has been observed that the carbides listed will react with UO_2 at high temperatures producing CO. The nitrides are similarly unstable at high temperature in the presence of steam and UO_2. The borides and sulphides are also believed to decompose under these conditions, and the hydrides are also unstable. The metals listed are all fairly reactive and will oxidize. Tungsten, in particular, forms volatile oxides on contact with molten UO_2 and reacts with vigor. After taking chemical compatibility into consideration, only four potential candidates remain. These are listed in Table 4.

When considering the merits of the four remaining candidates, tantalum oxide is least desirable since its melting point is approximately 1000°C less than that of the other three. The remaining three materials are considered in considerably greater detail below.

B. Leading Candidate Materials

1. Hafnium Dioxide

Hafnia is chemically similar to both zirconia and thoria. Of these three, hafnia is the rarest, most costly, and least well described in the literature. All three materials share the same crystalline structure, and have similar thermochemical properties. Even the phase diagrams of hafnia and zirconia are similar with phase transitions from monoclinic to tetragonal occurring in the same temperature range, 1400°C. and 1200°C., respectively. The foregoing comparisons of hafnia and zirconia suggest that

Table 3: Materials With Densities Greater Than 8.0 g/cc and Melting Points Above 1535°C

Element or Compound	Density (gm/cc)	Melting Point (°C)
Hf	13.31	2227
HfC	12.20	3890
HfO_2	9.68	2758
Lu	9.84	1663
Mo	10.2	2617
MoC	8.20	2692
Mo_2C	8.9	2687
Nb	8.57	2468
NbN	8.4	2573
Ta	16.6	2996
TaB_2	11.5	3000
TaC	13.9	3880
TaN	16.30	3360
Ta_2O_5	8.2	1872
Th	11.0	1750
ThC_2	8.96	2655
ThO_2	9.86	3220
Tm	9.32	1545
W	10.35	3410
WB_2	10.77	2900
WC	15.63	2870
W_2C	17.15	2860
UB_2	12.70	2365
UC_2	11.28	2350
UH_3	10.95-11.4	
UN	14.31	2630
UO_2	10.96	2878
US	10.87	2000+

Table 4: Candidate Core Retention Materials

Element or Compound	Density (gm/cc)	Melting Point (°C)
HfO_2	9.68	2758
Ta_2O_5	8.2	1872
ThO_2	9.86	3220
UO_2	10.96	2878

reactions of hafnia, which are not found in the literature, can be inferred from those of zirconia, which are known. No phase diagrams for systems of hafnia with urania and iron oxides have been found in reviewing the literature.

The phase diagram of hafnia with zirconia shows a liquidus curve which various monotonically from the melting point of zirconia, 2690°C., to that of hafnia, 2900°C. The phase diagram[12] of zirconia with urania shows a similar monotonic curve between their respective melting points. In the absence of better data, it is inferred from this that the hafnia-urania phase diagram would resemble the urania-zirconia phase diagram. The effect of FeO on hafnia is expected to be the same as that on zirconia with liquid mixture temperatures occurring as low as 1300°C.

The availability of hafnia in large quantities may present a problem. Furthermore, it would probably be necessary to construct a facility to manufacture the material into bricks or other suitable shapes. Hafnia has one advantage over thoria and urania in that it is nonradioactive (one isoptope, ${}_{72}Hf^{174}$ which represents 0.18% of naturally occurring hafnia, has a half-life of 2×10^{15} years). Since hafnia is considerably more expensive than urania and thoria and since its behavior with molten core debris is not well characterized, it is not felt that it offers any particular advantage over the other materials under consideration.

2. Uranium Dioxide

Uranium dioxide was developed as a reactor fuel when it was found that it had excellent resistance to corrosion in high pressure water at temperatures up to 300°C and superior resistance to radiation damage. It will dissolve in iron oxide slag at 1300° and is generally relatively chemically reactive. One difficulty in the development of uranium dioxide fuel elements is its tendency to fracture and turn to dust under various conditions.

Uranium forms a number of stoichiometric oxides, such as UO, UO_2, U_4O_9, U_2O_5, U_3O_8, UO_3, UO_4, and others. Of these, UO_2 is the most stable and is preferred for use in fuel elements. Although the formula UO_2 implies that it is a stoichiometric compound, this is not accurate because the oxide can absorb additional oxygen. Consequently, the composition of UO_2, at room temperature, ranges from UO_2 to $UO_{2.25}$. In the range between UO_2 and U_3O_8, there are some two dozen non-stoichiometric oxides. The system uranium-oxygen has been identified by some as the most complex in inorganic chemistry.

The composition of uranium dioxide depends not only upon temperature but also on oxygen pressure, rate of heating, and previous thermal history. The chemical changes involved in transformations from one oxidation state to another result in density changes subjecting the solid to severe stresses which cause disintegration.

UO_2, as commercially delivered, is most frequently $UO_{2.05}$. The oxidation state increases to approximately

U_3O_8 when heated in dry air to 600°C. When heated further, oxygen is given off and, at 1100°C, the state of oxidation is UO_2. When heated beyond 2000°C, the state of oxidation goes to $UO_{1.85}$. When this material is cooled to room temperature, it is found that a mixture of uranium metal and $UO_{1.96}$ is formed.

Uranium dioxide is prepared from uranyl nitrate hexahydrate, which is treated with hydrogen peroxide to precipitate UO_4. The UO_4, in turn, is reduced in a hydrogen atmosphere at 700°C to form UO_2. Uranium dioxide can also be formed by the reaction of steam on uranium metal at 350°C.

Solid uranium dioxide is formed from the powder by cold pressing and sintering, hot pressing, extrusion, and slip casting. The preferred technique is that of cold pressing and sintering, which produces a material with a density of 95% of theoretical. Cold compacting of the powder is carried out with steel dies at pressures of approximately one ton per square inch. Organic binders, such as polyvinyl alcohol, are added to the UO_2 to increase the green strength of the cold pressed compact. This binder is volatilized during sintering at approximately 2000°C.

The density of the material produced after sintering is strongly dependent on the properties of the original powder, the oxygen pressure, and the non-stoichiometry of the starting material.

The chemical and density changes in the oxide while cooling from 2000°C produce enormous strains in the solid. These strains produce cracking which reduces the strength of the material. If the solid uranium oxide is thermally cycled in air, it normally fails. For this reason, uranium dioxide cannot be used as a structural material. Its use in a core retention application would probably require its encapsulation in another material.

In conclusion, urania suffers from several disadvantages: high chemical reactivity, susceptibility to slag attack by iron oxide, high cost ($40/lb), and a tendency to revert to a powder. It should be noted that the cost of depleted urania is purely artificial since there is no current market for the material; the price of the material is set by the government.

3. Thorium Dioxide

a. Physical and Chemical Properties

Thorium forms one of the most stable oxides known. It has been estimated[13] that the dissociation constant of thoria into thorium and oxygen is 10^{-27} at 2000°C. and 6×10^{-6} at 3000°C. The ThO_2 phase is the sole form known in the Th-0 system[14] (even though at high temperatures a slightly hypostoichiometric composition has been reported by Benz[15]). Thorium dioxide has a CaF_2-type crystalline structure and is isostructural with uraniua as well as zirconia, and hafnia.

Thoria has both the highest melting point and stability of any oxide. At temperatures of approximately 2000°C., in an inert atmosphere, no reaction has been reported between thoria and molybdenum, nickel, niobium, titanium, or zirconium[16]. A slight reaction has been reported with beryllium. In vacuum at this temperature, graphite reduces thoria with the formation of carbides and carbon monoxide. In vacuum, slight reactions are also reported with tungsten and molybdenum. No chemical reactions have been observed with alumina, zirconia, or urania.

Thoria is chemically stable in an oxidizing atmosphere up to its melting point of 3300°C. Under strongly reducing conditions thoria will be reduced, especially if the reducing element forms a highly volatile oxide.

The stability of thoria in high temperature water has been demonstrated in one application where it is used to coat urania pellets[17]. At a pressure of 50 kg per cm^2 and a temperature of 400°C., these pellets showed a negligible corrosion rate.

Thoria dissolves in a number of materials including thorium[15]. Thoria is soluble in all proportions in UO_2 with a eutectic temperature only slightly below the melting point of UO_2. Similarly, thoria dissolves in zirconia with a eutectic temperature at the melting point of zirconia. The only major problem in use of thoria in a core retention system appears to be a possible low melting solution formed with iron oxide. It should be noted that this same problem is also encountered with other candidate core retention materials such as magnesia, zirconia, urania, and high alumina cement. Although an extensive literature survey has been conducted, no thoria-iron oxide phase diagrams have been found. Preliminary data from recent experiments at Sandia Laboratories do not show complete melting of thoria-iron oxide mixtures at temperatures as high as 1460°C. It is hoped that future experiments will provide additional information in this area.

The industrial applications of thoria include its use in free standing crucibles for the fusion of zirconium, thorium, uranium, platinum, osmium, rhodium, and radium. Thoria is also used for coating tungsten filaments in electron tubes. In solid solutions with urania, thoria has been used as a structural material in atomic power applications. The most familiar application of thoria is its use in incandescent mantles in gasoline lanterns where it survives in both oxidizing and reducing conditions.

b. Availability of the Raw Material

Supplies of unprocessed thoria appear to be adequate, on the assumption that 110 tons will be needed for each reactor. Although thoria is a fairly common material, there are few commercial uses for it. Although the yearly consumption of thoria in the U.S. is only 33 tons per year, there is no lack of raw material. Proven reserves of thorium in this country amount to 192,000 tons (equivalent to 218,000 tons of thoria). The worldwide proven reserves of thorium, as a metal, are 904,000 tons. Over 4200 tons of thorium are immediately available in the form of a residue in a pond in Tennessee.

Most of the thoria powder used in this country is supplied by Rhone-Poulenc, Inc. of France. There are no American vendors of thoria powder. Rhone-Poulenc operates two processing facilities for thoria and other materials, at La Rochelle, France and at Freeport, Texas. The La Rochelle plant has a total capacity of 5000 metric tons per year and produces about 500 tons of thoria per year. The new Freeport plant has a capacity of 4500 metric tons per year, but only extracts La, Nd, Pr, and Ce. The remaining ore is then shipped back to France where the other constituents, including thorium, are separated. Eventually, the Freeport plant may have the capability to separate thorium from the ore. The thoria produced by Rhone-Poulenc is extracted from monazite sand from India and Brazil. The ore constituents are converted to nitrates and are separated by liquid-liquid extraction.

Rhone-Poulenc, Inc., currently quotes prices for thoria in 1000 kg lots. Their current selling prices are $8.35 per pound for 99 percent pure material and $17.40 per pound for a grade which is 99.9 percent pure.

c. Fabrication Facilities

An important consideration in the selection of a material is the existance of fabrication facilities. It appears that there are only a few concerns in this country with any thoria fabrication capability. The capabilities of several of these concerns are indicated below.

The Ceramic Products Division of the Corning Glass Works in Solon, Ohio, has fabricated objects from thoria in the past. Corning has allowed its government licenses to work with thoria to expire because of the costs and problems in coping with environmental regulations. Their old facility for thoria fabrication was relatively small, and it would be necessary for them to construct a new facility in order to fabricate thoria in the quantities needed for a core containment system. Corning estimates that 2-3 years would be required to license and construct any new facilities. Corning has no facilities for testing or melting thoria. Their capability is limited to making specimens from thoria.

Another vendor of fabricated thoria is Cerac Inc. of Butler, Wisconsin. Their current capability is limited to the manufacture of granules of thoria (-8 to +140 mesh) and washer-sized components. The granules and washers are prepared by arc-melting thoria powder in a graphite crucible. Cerac has no facilities for fabrication of material into other structures, such as bricks, and has a very limited capacity (a few hundred kg. per year). They cautioned that it has been their experience that it takes a year to obtain thoria powder from the only supplier, Rhone-Poulenc, Inc.

A third possible vendor of fabricated thoria is Trans-Tech, Inc. of Gaithersburg, Md. Trans-Tech indicated that it would prepare thoria by sintering the material at 1500-1600°C in tunnel kilns as part of a continuous manufacturing process. They are able to form brick size objects with dimensions as large as twelve inches and can achieve a density of 95% of theoretical. Their facilities have a capacity of approximately 10-15 tons of ceramic oxides per year. However, any substantial thoria production would reduce their capacity for other ceramics. In two years, they will occupy facilities at Adamstown, Md did express a willingness to manufacture thoria in the quantities required provided there is sufficient time to construct or acquire facilities.

Finally, the Eagle Pitcher Energy Products Co. in Oak Ridge, Tenn. indicated that they are able to convert several high speed presses used in the past for fabrication of highly enriched urania to thoria pellet production. They have a capacity in the area of 200 tons per year. They have estimated that they could provide 100 tons of thoria fabricated into pellets or spheres for approximately $5 million.

In addition to the three concerns mentioned earlier, thoria is used in this country by the Coleman Co. in the manufacture of gasoline lantern mantles. Thorium metal is also used in the manufacture of alloys. The alloys are used principally in incandescent electric light bulb filaments which contain tungsten and from 0.5 to 10.0% thorium.

Some capability is known to exist in Europe for fabrication of match-box size specimens of thoria by hot-pressing techniques. It appears that match-box sized bricks of thoria have been considered in the past for use in the SNR-300. It has been suggested that the most probable supplier of thoria in Europe is Degussa of Germany, a large manufacturer of ceramics.

There does not appear to be any technical reason why thoria cannot be fabricated at reasonable cost in the quantities needed for a core retention system. The facilities and processing required are not particularly extensive. The radiological hazards associated with the fabrication of thoria are less than those associated with urania. The other principal hazard is heavy metal poisoning (lead poisoning is an example), which industry commonly copes with successfully. The lack of commercial demand for thoria by industry seems to be the principal reason for the paucity of thoria fabricators.

C. Conclusion

It should be noted again that although the materials listed in Table 4 are regarded as the best ones available for the special case of the ZIP application, consideration has also been given in the following sections to some selected low density materials.

In conclusion, it is felt that thoria is an excellent choice for a molten core retention system material for existing reactors, such as the ZIP plants, which have a high radiation environment and restricted space available in the reactor cavity. Thoria has a high density so that it will not float in molten core debris and is chemically inert. Compared to UO_2, it also has the advantages of having a higher melting point and a greater ease of manufacture into structural shapes such as bricks. High temperature crucibles are regularly made from thoria. It appears that sufficient supplies are available and a fabrication capability exists that is adequate for the proposed use.

VII. CORE RETENTION CONCEPTS

A number of core retention concepts have been examined as a part of the ongoing ZIP effort. All of the concepts considered include a provision for cooling the molten core debris and protecting the concrete basemat. Calculations of the containment response indicate that the core debris from an accident must be cooled and the reactor cavity must be protected in some manner if overpressurization of the containment is to be avoided. Overpressurization is expected after a day in the absence of cooling. Unfortunately, the radiation hazard in the cavity below the reactor, which arises from activation of the concrete, and the space restrictions severely limit the range of mitigation methods.

In the following sections, some of the concepts suggested over the years are summarized and reviewed. In addition to the materials discussed in the previous section, concepts employing low density materials, castable materials, and low-melting materials as well as refractory materials are included. It should be emphasized that the lack of experimental data for molten core-material interactions in the temperature regime of interest for these materials and concepts is a serious problem and greatly complicates the selection of a core retention concept.

A. Flooded Thoria Rubble Bed Core Retention System With a Water Supply

1. Description

A flooded rubble bed (Figure 3) is a core retention scheme that appears feasible, is simple and easy to put in place and does not in any way compromise the safety

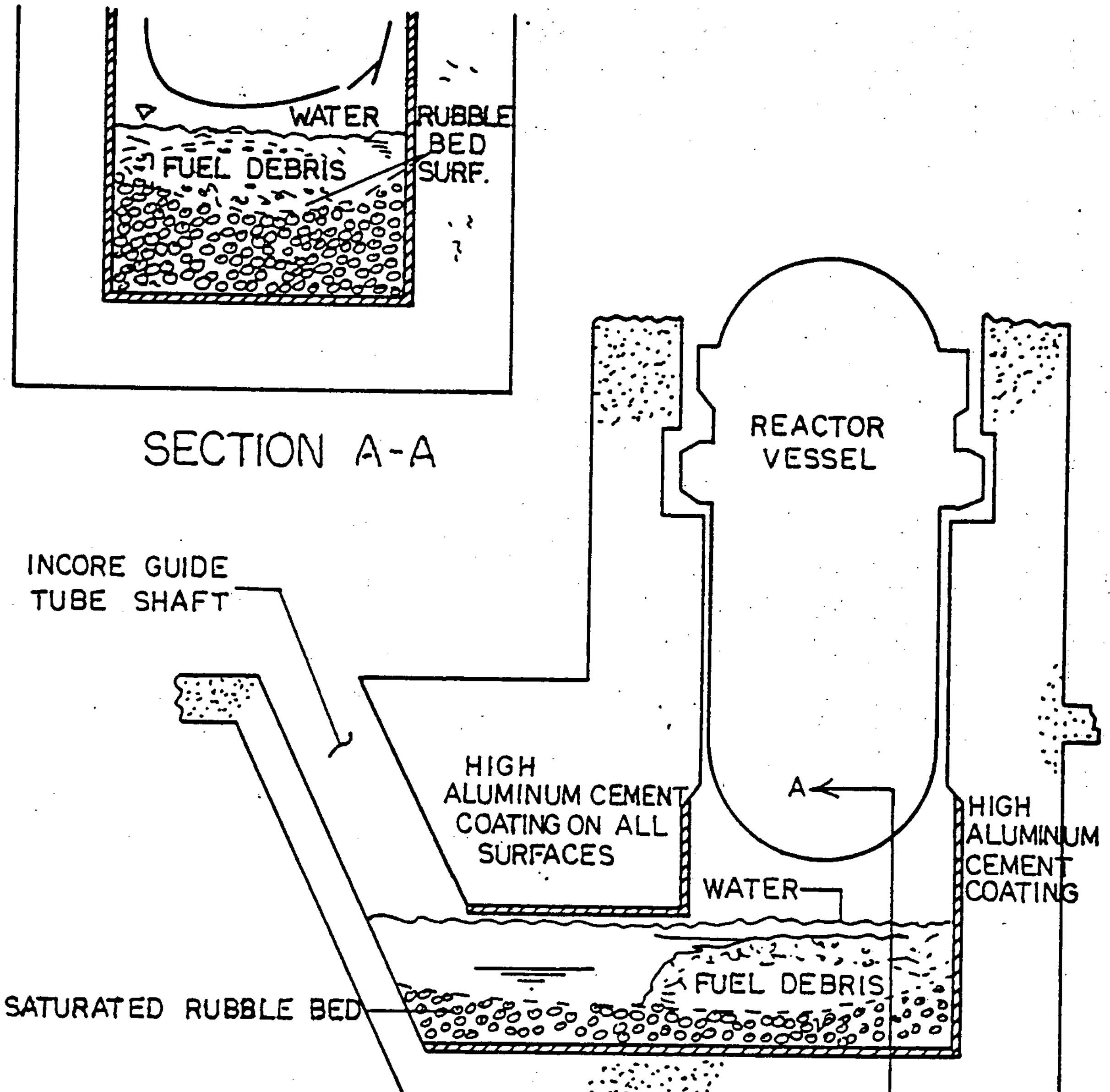


Figure 3. Rubble Bed Core Retention Concept (not to scale)

of the existing plant. The core retention system is simply a bed of thoria pebbles on the floor of the reactor cavity. The depth of the bed, particle sizes and shape need further examination before they can be fully specified. The bed would be limited to a depth of 6-9 in. under the instrumentation conduit although the thickness could be greater at other locations.

A rubble bed system has the advantages of being relatively easy to install and involving minimal radiation exposure to personnel since the thoria could be simply poured into the cavity through the 8 foot square access opening, or through the instrument tube shaft. Only about 68 gpm of water would be required to remove the estimated 30 MW (or about 1% of full power) of decay heat after a day, although for certain sequences, several times more could be necessary initially.

A temperature sensor would cause the bed to be flooded with water in a passive, but controlled manner, if a core melt accident was imminent. Water would be brought to the bottom of the cavity through a pipe and could be used to flood the bed and cavity to a depth of several feet. In the event of an accident, the water in the bed would remove heat from the molten core debris by evaporation. The evaporated water would be condensed by equipment located elsewhere in the containment and the water would be returned to the thoria bed through coolant pipes. Passive water sprays would also operate to help cool the molten core debris. Steam in the containment would assist in hydrogen control and would contribute to the maintenance of a non-explosive mixture.

2. Sidewall and Ceiling Protection

As long as the molten core debris is covered by water, there should be no danger to the upper cavity sidewalls and ceiling from either outgassing or melting as a result of radiation heat transfer from the core debris. However, additional protection can be obtained by having water trickle down the sidewalls and sprayed across the ceiling of the cavity by passive means.

The reactor cavity walls can be given some protection against molten core debris by the application of high alumina cement. Two high alumina concretes which may be suitable for wall protection are the Shasta and Armor concretes. These materials have the highest operating temperatures of any concrete with a maximum operating temperature of 3200°F (1760°C). These concretes have a density of 170 pounds per cubic foot and cost about $650 per ton (more than 20 times the cost of ordinary concrete).

These concretes are suitable for gunniting (spraying onto) walls. Both Babcock and Wilcox and Combustion Engineering have experience with applying these concretes to walls.

If concrete is sprayed onto an old wall, it will not adhere without a support structure. It is first necessary to secure a steel grid or lath to the wall. This is most easily done by preparing panels of this lath (as large as can be brought into any access entry ways) and installing them with ramset concrete nails (explosive driven concrete nails). The lath is installed so that the lath will be between 1/2 and 3/4 of the way between the old wall surface and the new surface. The old wall has to be wetted for about 24 hours before application of the high alumina cement and the new cement has to be kept moist for a time after setting.

3. Effect of Water in the Cavity

Before continuing with a description of the rubble bed itself, the question of desirability of water in the cavity must be addressed. Without water in the cavity, the core debris could penetrate into the interstitial spaces in the rubble bed as well as attack the concrete sidewalls causing decomposition. Further, radiative heat transfer from the molten core debris (or hot solids) would cause decomposition of concrete surfaces above the cavity floor. This is to be contrasted with the situation in a water filled cavity. The core debris will be completely enveloped (above and below) with water. Consequently, the entire decay heat will be used in vaporizing water which can condense on surfaces away from the cavity, with the condensate returning by gravity. Therefore, with water present in the cavity, decomposition of the concrete can be controlled. There is some indication that the large surface area available for condensation may be sufficient to handle the heat load for a significant period before the containment atmosphere reaches limiting temperatures or pressures.

4. Reasons for a Rubble Bed

The rubble bed will be a layer of non-reactive, refractory gravel covering the bottom of the reactor cavity. Its purpose is to protect the concrete and to insure that there is a means by which water can flow over the cavity floor and protect the concrete from the core melt for the long term, as well as during early attack when there may not be water present, or when water may be ineffective in preventing core debris from interacting with the concrete floor of the reactor cavity. A high melting refractory material will provide protection against initial attack, which is important if the molten core debris pool forms suddenly. If the refractory material is to be effective, it must be chemically inert in molten core debris and must not float away.

The use a of refractory gravel also enables consideration of the use of an initially dry cavity, with water added later. It may be advantageous to be able to add water after the molten core debris arrives in order to reduce the possibility of a steam explosion.

A refractory gravel also will allow the entry of coolant water at the bottom and along the lateral boundaries of the molten pool. Consequently, cooling water will be able to perculate through the molten core debris and cool it more effectively.

A high melting point gravel is desirable to ensure that the gravel retains its shape, especially if a decision is made to flood the bed after the molten core debris has arrived. If the gravel does not maintain its shape, the passage of cooling water through the bed will be adversely affected. Since the gravel will be immersed in water, it is also desirable for the refractory to be inert in water.

The rubble bed must be thick enough and have a high enough permeability to allow sufficient amounts of water to flow beneath the fuel debris so that it will remain wet under the most conservative estimates of the downward heat transfer. The rubble bed must be designed to avoid being swept away by the forces associated with vessel failure and it must not allow molten core material to trickle through the interstitial spaces and reach the concrete.

5. Shape of the Rubble Bed

The rubble bed could be dished (Figure 3) so that the core debris is contained within the hollow formed by the dishing. This would further protect the walls. It is only directly beneath the instrument tubes that the rubble bed thickness is limited and this region is several feet away from the keyway -cavity interfaces (see Figures 1 and 2). Making the rubble bed thicker in the keyway and near the cavity boundaries will give further protection to vertical surfaces as the molten material will collect in the depression. The instrument tubes require no external cooling and could be partially buried in the porous rubble bed without interfering with their movement during reactor vessel heat up and cool down. This would allow a deeper bed everywhere and smaller particles could be used with fewer concerns about water flow and bed displacement.

6. Rubble Bed Particle Size

It is anticipated that the retention system would consist of a graded bed of thoria with particles as large as 0.5 inches in diameter. The coarse particles will allow water to circulate when the bed is flooded and they will be less readily swept away by the jet impingement of falling molten core debris. There is the possibility that thermal shock could break up some of the pellets and thereby make the debris bed particle size less controlled. This could affect water circulation through the bed. At present, there is insufficient data to evaluate this possibility.

At the bottom of the bed, finer thoria particles should be used. The layer in contact with the concrete floor should be a thoria powder which will protect the concrete from channeling of molten core debris through the spaces in between the large diameter particles, especially in the event of cooling system failure.

7. Wire Mesh Net to Restrain the Rubble Bed During Steam Explosions

The coarse thoria particles should be constrained so as to remain in place during the early stages of an accident. Protection against a steam explosion can be achieved through the addition of a mechanically strong, easily applied, penetration resistent covering which will not add a different material. A steel mesh net, as is commonly used in the construction industry has the requisite properties to contain the thoria rubble bed. These nets are well suited to containing high velocity solid missiles. They are also quite porous so that water and steam can readily pass through the net. The melting point of the steel is relatively high. Once melted, the net would not produce any products which would enhance attack on the thoria, since steel is a constituent of molten core debris. Also, such a net is relatively easily installed and is inexpensive.

It is possible to fabricate a net out of other metals, such as tantalum, that would be more resistant to melting during jet impingement. At present, these alternatives are being examined from the point of view of cost, availability, and fabricability of the materials. Other metallic restraints, such as a honeycomb structure, which will have greater strength should also be investigated.

8. Selection of Particle Shape to Increase Resistance to Jet Impingement

The addition of a steel mesh net or other structure as discussed above, should provide adequate protection against bed displacement during a steam explosion. However, a steel net would probably melt if it came into contact with a jet of molten material.

Additional resistance to displacement of the refractory gravel can be provided by designing the gravel pellets to have shapes that will tend to resist being moved. There may be substantial problems in fabricating gravel from refractories with particular shapes. With this in mind, it is interesting to examine the effect of particle shape on displacement.

It is well known that high velocity bodies of liquid such as jets and waves can displace granular solid material which has a density of several times that of the liquid. Familiar examples of this effect are hydraulic placer mining and shoreline erosion by wave action. Engineering studies of shoreline erosion have expended some effort in finding methods that can be employed to minimize the displacement of the solid material.

Some of the observations and conclusions reached by the marine engineering studies are applicable to the problem of minimizing the displacement of refractory materials by a corium jet; others are not. It has been observed that an increase in both the size and density of the solid particles decreases their tendency to be displaced by a liquid. These two parameters are limited for our purposes by the physical properties of candidate refractory materials and the space limitations beneath the reactor vessel.

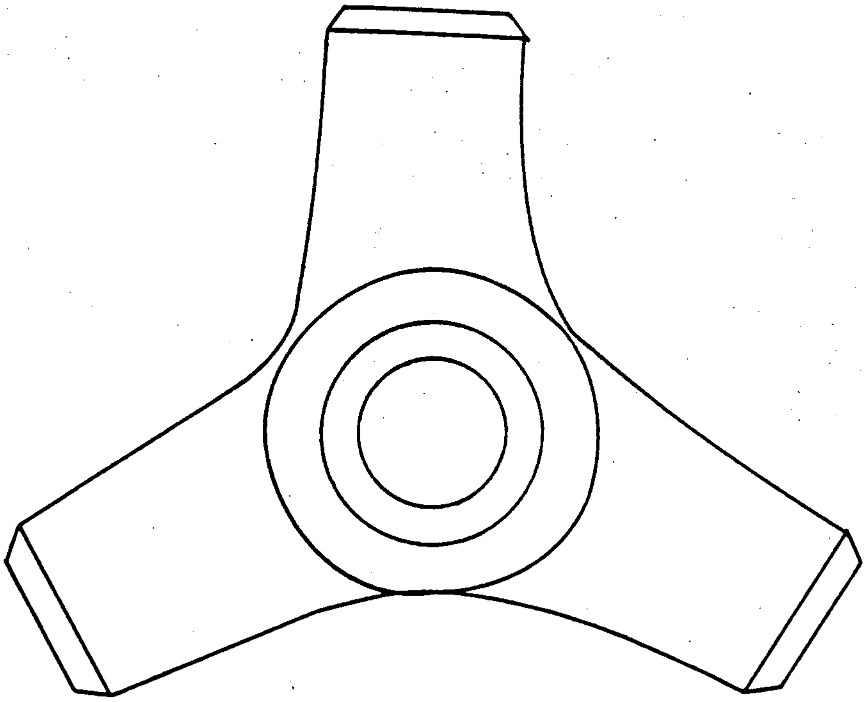
Given a maximum density and particle size for the solid material, other parameters are also found to affect the displacement of the material. The two most important of these parameters are both related to the shape of the particles. It should be readily perceived that anything which would increase the interparticle friction would increase the displacement resistance of the particles. At first, it was noted that rough angular particles were more displacement resistant than smooth rounded ones. In fact, spherical particles have the least resistance of any shape to displacement by fluids. For this reason in breakwater construction, rough quarried rock and concrete blocks are found to be more displacement resistant than rounded boulders found near the shore. Thus, quarried rock is employed despite the obvious cost advantage of using locally available round boulders.

The second shape related parameter is less obvious and is related to the back pressure developed within the particle bed as a result of the impinging liquid stream. If the particles have any significant cracks or spaces between them, the liquid penetrates between the particles exerting pressure in all directions including upward. This upward component of pressure lifts the surface blocks or particles and effectively reduces the inter-particle friction to zero allowing them to be swept away.

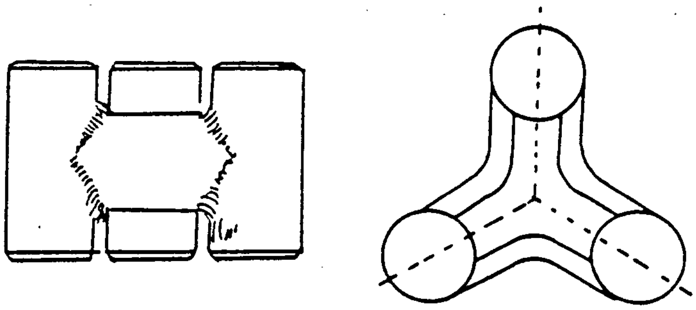
At this point, it should be noted that a very strong case can be made for the most effective means for preventing particle displacement. In marine engineering, a concrete grouting is used in some applications to cover the outside surface of the breakwater. This effectively makes the entire breakwater one large block, raises the inter-particle friction to a very high value, and finally prevents the back pressure problem entirely by preventing the entry of liquid between the cracks. For this reason, some though should be given to the possibility of using a two layer gravel bed system in retrofitting a reactor. A lower thin layer of thoria in contact with concrete could be held down with a small amount of cement to protect the concrete basemat from direct contact with a jet of molten core debris. The upper layer would consist of loose gravel, as proposed earlier, which would allow cooling of the molten core debris by transpiration.

There are a number of practical steps that can be taken to improve the displacement resistance of the gravel by increasing the inter-particle friction and reducing the back pressure. It is useful to examine initially the results of breakwater design studies. The first steps taken in breakwater design in reducing the back pressure were accomplished by drilling holes in the blocks used in breakwaters. The upward pressure was relieved by water jetting upwards out of these holes forming geysers.

More recent developments[17,18] in this field have led to the development of cast concrete shapes which combine the important features of high inter-particle friction and relief of back pressure. These shapes include the Tetrapod[19] (Figure 4(a); France), Tribar[20] (Figure 4(b); United States), Stabit[20] (Great Britain), Quadripod[20] (United States), and the Hollowblock[21] (Japan). Most of these designs are covered by patents.



(a) Tetrapod



(b) Tribar

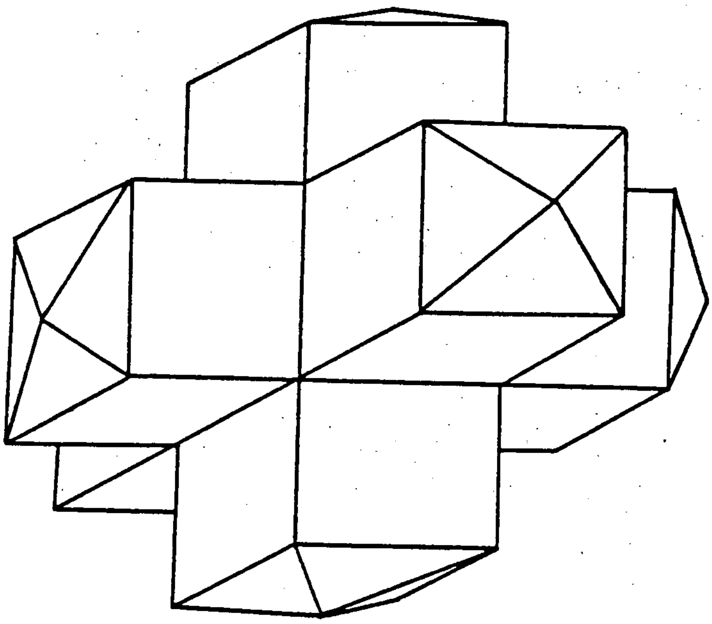
Figure 4: Refractory Pellet Shapes

These designs are all currently in use in breakwaters in various parts of the world. The tetrapod design has a high level of inter-particle friction and is perhaps the best design for this feature. These structures tend to interlock and will not slip past each other unless one or more of their arms are broken. The structures do not require careful placement and can be installed by simply dumping them in place. The other designs do not interlock quite as effectively but have other advantages. All the designs allow for relief of back pressure by virtue of a relatively large void space. Most designs are about 50% solid and 50% void when placed randomly. The only requirement, based on experimental observation, is that the layer of particles must be at least 2 particles thick if it is to remain in place.

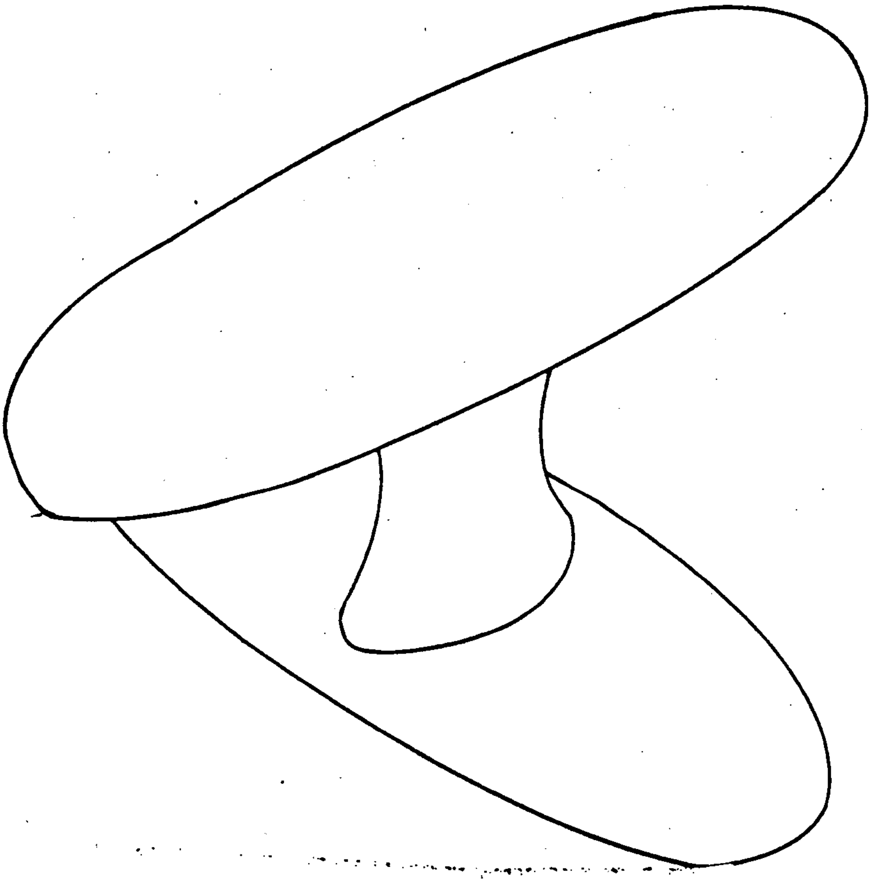
At this stage, the design of structures to resist displacement in fluids is more of an art than a science. Little is known and most of the available information is from empirical observations.

The designs of the breakwater shapes could equally well be used in much smaller sizes as a refractory gravel. However, some thought has to be given to their manufacture. The tetrapod, for instance, cannot be made using a two part mold since that shape cannot be removed from the mold without breaking either the mold or tetrapod. The other shapes lend themselves to manufacture using two part molds. This is an important consideration when one considers the number of such shapes required to cover a containment building floor.

The "Jacks" design (or Hexapod, see Figure 5(a)) and the twisted "H" (see Figure 5(b)) are also presented as



(a) "Jacks" Pattern



(b) "Twisted H" Pattern

Figure 5: Refractory Pellet Shapes

possible candidate shapes for a refractory floor covering. Both are readily manufactured using two part molds, both will interlock together, and both give back pressure relief.

Considerable difficulties may be encountered in attempting to manufacture gravel particles with complex shapes from a highly refractory material like thoria. As of this time, no effort has been made to assess the extent of the problems which may be encountered in the preparation of gravel particle with complex shapes.

9. Sintering by Molten Core Debris

The resistance of the rubble bed to jet impingement will be enhanced to some extent by sintering of the bed by molten core debris. Sintering of thoria is expected at a temperature of about 1800-1900°C. Recent experimental evidence obtained at Sandia Laboratories showed that iron oxide powder was not displaced by thermite falling from a height of one foot (which is substantially less than the height of fall expected in the ZIP reactors). There was preliminary evidence that sintering of the particles in the bed contributed to the lack of displacement.

10. Floatation Resistance

As previously mentioned, thoria was selected because of its very high melting point and low chemical reactivity. In addition, and most importantly, the material will not float in molten core debris due to its high density. Consequently, it does not require mechanical restraint to avoid floatation by molten core debris and can be poured in place.

11. Slag Attack

One potential problem with the use of thoria in a rubble bed containment system is its possible solubility in iron oxide. This possible solubility is one of the most important items of experimental data required to evaluate the material for use in core containment systems, as discussed earlier. However, it should be noted that every other candidate material for core retention is soluble in iron oxide and that thoria may, in fact, be much better, based on recent information from Sandia experiments. Regardless of its solubility in iron oxide, thoria is an attactive choice as a containment material because of its exceptional chemical and physical properties.

As in the case of the MgO lined ladle used in the steel industry, the expected region of dissolution by iron oxide, or slag attack, is at the sides of the containment cavity. Since the slag floats on the molten iron, no slag attack is expected at the bottom of the cavity until most of the iron has been converted to iron oxide.

The extent and rate of the conversion of molten iron to iron oxide slag is unclear. The steam from water in a flooded thoria bed will clearly be available to oxidize the iron, forming slag. However, there is the possibility that heat transfer from the core debris to the water will cause freezing which will stop the slag attack.

The reaction that generates slag also generates substantial quantities of hydrogen. Also, contact between hot urania, steel and water raises the possibility of a steam explosion. Water in contact with hot steel above 550°C. will generate substantial quantities of hydrogen.

While the steam that is generated will reduce hydrogen flammability to some extent, the success of the concept depends on the fraction of steam converted to hydrogen. If most of the steam bubbles through a molten layer after evaporating, a very large fraction of it will probably be converted to hydrogen with the oxygen forming iron oxide. The quantities of hydrogen produced might well exceed the capability of the water sprays to maintain a non-flammable mixture.

12. Containment Heat Removal

Installation of the proposed core retention system will not solve the problem of containment heat removal. Adding steam to the containment environment will cause the pressure to rise and, without its condensation, failure will eventually occur. The energy in the evaporated steam must find an ultimate heat sink, or all the retention system does is buy time. The proposed retention system will become a reflux boiler with condensation occuring on the containment walls. The decay energy will be thereby deposited as sensible heat in the containment structure and other equipment in the containment. A detailed discussion of these problems will be found in a report to be published by I. Catton Two methods of heat removal are discussed here.

One method of removing heat from the containment is to place a condensor inside the containment on a vertical surface as near to the cavity as possible. The condensor would be connected to a natural draft, natural circulation cooling tower outside containment, as depicted in Figures 8 and 9.

Another means (Figure 6) of providing a passive containment heat removal system involves the use of heat pipes. Commercially available heat pipes penetrating the containment could transfer a large amount of energy very effectively. The condensing surface inside the containment would be the heat pipe evaporating surface and a surface cooled by natural convection outside containment would provide the heat pipe condensing surface. The controlling heat transfer would be natural convection. Another loop connecting the heat pipe to a cooling water tower may be necessary to obtain the required surface area for passive buoyancy driven heat transfer.

Other passive means could probably be developed. The two above are discussed to indicate that passive heat removal or augmentation is probably feasible. The containment itself has some inherent heat removal capacity. A more detailed study is needed to establish the extent of the inherent heat capacity so that a supplementary passive system, if needed, can augment it.

B. Other Core Retention Concepts

A number of other core retention device concepts have been considered for the ZIP plants in addition to the flooded thoria rubble bed discussed previously. Some of these are discussed in the remainder of this section. It should be noted that these concepts are discussed from the point of view of their suitability for use in the ZIP environment. Many of these concepts are well suited for use in new construction, or in other reactor environments, even though they are not well suited for the specific case of the ZIP environment.

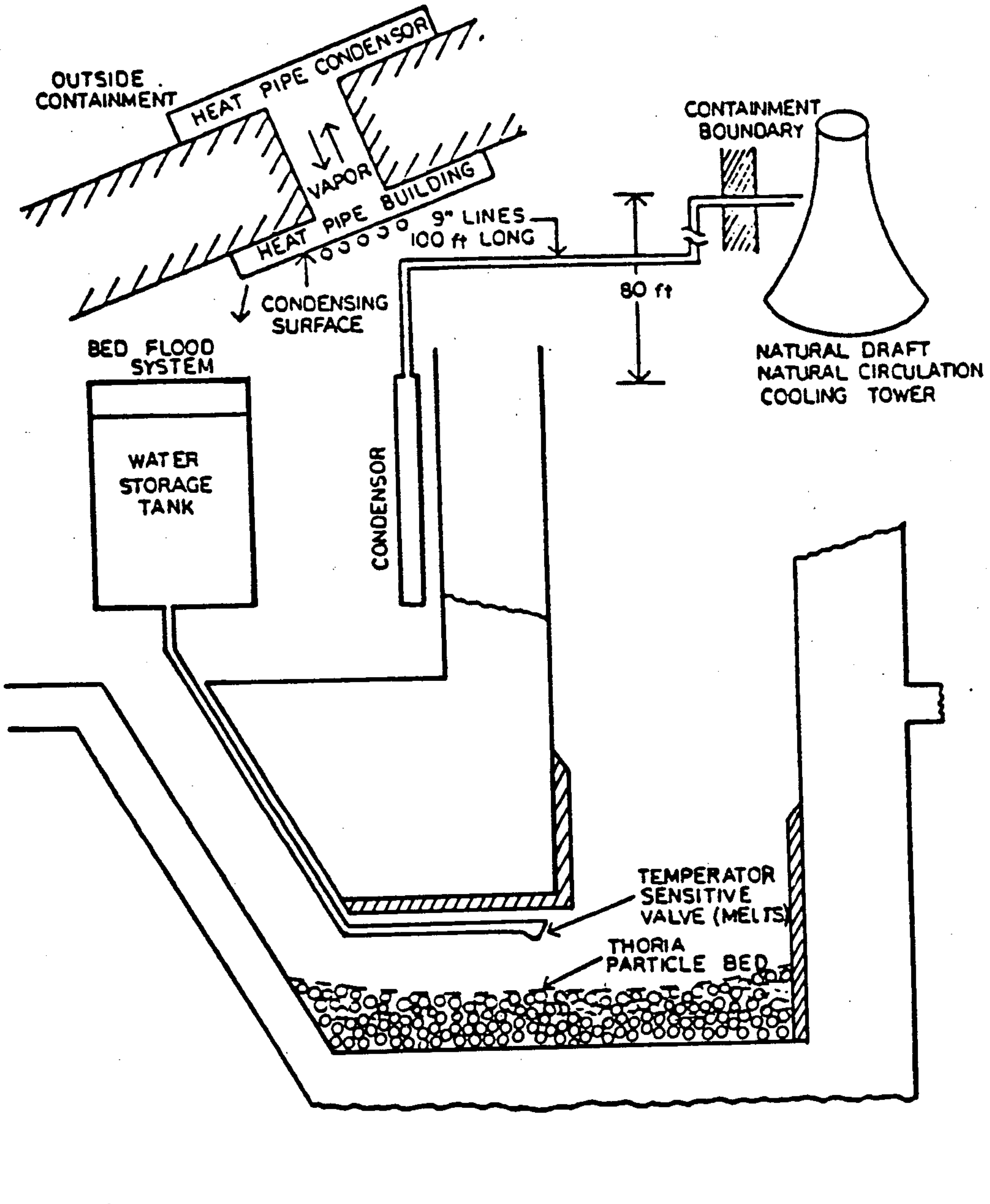


Figure 6: Thoria Rubble Bed Including Two Approaches to Containment Heat Removal

1. ThO_2 or UO_2 Covering Cooling Coils

In one concept, the 6 to 9 inch region between the instrumentation conduit and the cavity floor would be filled with a thoria or urania debris bed covering a cooling coil with a square cross-section. The coils would be prefabricated outside of the containment and assembled in the reactor cavity to reduce the radiation exposure to personnel. Connections between coils would be made outside of the high radiation area. Woods metal, or some similar low-melting temperature material would be used to insure good thermal contact. The coils would also be extended along the sidewalls of the cavity to protect the walls from molten core debris. Radiation shielding, consisting of several layers of a very thin (a few mills) reflective metal, such as tantalum or tungsten, would be installed to protect the roof and sidewalls of the cavity from radiation heating.

The region above the coils would be packed with several layers of increasingly coarse thoria (ThO_2) or urania (UO_2). The material would be packed around the instrumentation leads. Since molten core debris may channel through the spaces between the coarser grains, the bottom layer covering the cooling system would be a powder. Coarser material is recommended for the upper layers in order to provide some protection to prevent the bed from being swept away if a jet impingement of molten core debris should occur. In order to reduce the probability for excavation of the granules by a jet, it may be desirable to add a thin layer of refractory cement on top of the granules to hold them in place during the initial impact of molten core debris on the retention system.

Either thoria or urania might be employed. It should be noted that both are sufficiently dense so that they will not float away in molten core debris with the expected maximum density ($\rho = 8.0$ g/cc). Of these two, thoria seems to be a better choice due to its higher melting point, general chemical inertness and relative ease of fabrication. Thoria melts at 3220°C ± 50° as compared to 2878°C + 20° for UO_2. Thoria is one of the most chemically inert compounds known while urania has a very complex chemistry. Thoria can be fabricated into a relatively hard, durable structure. Urania is much more difficult to fabricate into useful structures and tends to disintegrate into powder. Neither material is very resistant to thermal shock although thoria is better than urania. Thoria's solubility in water at 24°C has been reported as less than 0.01 mg/liter H_2O. In tests, urania has lost 0.3% of its weight after exposure to 650°F water for 147 days. Thoria has about one-third of the radioactivity of urania.

An examination of the expected chemical interactions of thoria and molten core debris constituents indicates that molten UO_2 and ThO_2 form a eutectic composition at a temperature within a few degrees of the melting point of UO_2. The eutectic temperature of the ThO_2-ZrO_2 system is also very high. However, the behavior of thoria with iron oxide is unknown. Iron oxide will probably form in great quantities when water from the reactor cooling system comes in contact with molten iron from the reactor support structures. Unfortunately, molten iron oxide and molten iron tend to act as universal solvents. Molten iron oxide forms a low melting eutectic (1300-1400°C. or lower) with all oxides for which data is available. Some recent data suggests that thoria-iron oxide interactions occur at higher temperatures, if they do in fact occur at all.

One can speculate that thoria will be more resistant to molten core debris (even if it contains iron oxide) than some other lower melting materials, such as high alumina cement (HAC), although there is no direct experimental evidence. This will be especially true where the molten core debris is at a temperature far above the melting point of HAC. At such temperatures, it is expected that HAC will be quickly penetrated as compared to thoria.

Another factor contributing to a greater iron oxide penetration resistance for thoria might occur because a eutectic composition, rather than molten iron oxide, would be in contact with thoria after the formation of the initial eutectic mixture. In a quiescent pool, as more thoria dissolved, the melting point would rise quickly, following the eutectic diagram. In the absence of experimental data, it is not possible to predict the rate of dissolution of thoria.

One potential disadvantage of thoria, surprisingly, is its low thermal conductivity. In a new plant with a core retention system designed in from the beginning and with a refractory liner on the walls and ceiling of the cavity, a low thermal conductivity would be advantageous. Unfortunately, the radiation levels make it impractical to have personnel install a refractory liner on the walls and ceiling in the course of retrofitting plants like the ZIP reactors.

The problem arises because a low thermal conductivity may prevent heat from being withdrawn fast enough through the cooling system. In this event, the heat will go upward where it may melt the concrete ceiling of the cavity and the upper sidewalls. It should be possible to overcome this problem by suitable sizing of the layer thickness so that a sufficient quantity of heat can be removed through the cooling system.

2. High Alumina Cement (HAC) with Cooling

In this concept, the interior of the cavity would be filled with a monolithic layer of relatively low density, refractory high alumina cement (HAC) to a depth of approximately six inches. Although more HAC could be readily pumped in around the instrument tubes forming a thicker layer, the conduit cannot be locked into place due to the need to provide for thermal expansion. Cooling water would be pumped in, either through the space between the metal plate cavity liner and the concrete or through cooling coils.

The concept has the advantage of ease of installation, requiring low radiation exposure to personnel. In addition, the concept is probably the least expensive to implement because of the low cost of the concrete. Furthermore, the greater thermal conductivity of the high alumina cement will allow the cooling coils to remove heat more readily than in the case of thoria, thereby reducing the thermal load on the upper regions of the cavity.

One possible high alumina cement for this application has a density of 2.97 g/cc with the composition shown in Table 5. Although HAC has a relatively low melting point[22] (1800°C.) as compared with either thoria, urania, or magnesia, it has been suggested that the low melting point is not important because iron oxide will probably form a low melting eutectic with almost any refractory oxide at around 1300°C. While we agree that this is

Table 5. Composition of High Alumina Cement*

Cement

Constituent	Weight Percent
CaO	28.
Al_2O_3	69.
Fe_2O_3	0.11
MgO	0.13
Na_2O	0.46
K_2O	0.47
SiO_2	$>$ 0.4

Aggregate ("DURALUM")

Constituent	Weight Percent
Al_2O_3	98.0
SiO_2	$>$ 0.4
Cr_2O_3	0.05
Fe_2O_3	1.55

Mixing procedure

Constituent	Weight (pounds)
Cement	94
Aggregate	
#2 mesh	420
#8 mesh	140
#16 mesh	140
#20 mesh	140
Water	44
Air entraining agent	12.5 oz
Plastimet	2.5 oz

*HAC data courtesy of Dana Powers, Sandia Laboratories likely, high alumina cement may be worse than most oxides in this regard because it is a mixture of a number of constituents. Generally, each additional constituent will further depress the eutectic temperature of a mixture. Also, if the molten core debris is above the melting point of the HAC, it will rapidly penetrate the cement.

Any comparison of high alumina cement with refractories must take into account the rate of dissolution and the possibility of saturation of the iron oxide layer at a particular temperature. In the absence of experiments, the rate of dissolution is unknown. However, it is expected that the iron oxide layer would be more quickly saturated with more refractory materials (thoria, for example) than with high alumina cement due to expected steepness of the temperature - composition curve on the refractory phase diagram. This steepness is a result of the very high melting point of refractories such as thoria. It may be that the quantity of molten iron oxide would be so great that it could not be saturated by the available thoria. If the iron oxide layer cannot be saturated and the rate of thoria dissolution is high, then thoria will not be better than HAC and the low melting temperature of HAC will be irrelevant.

However, we would expect that thoria would be more resistant than HAC, especially to attack by molten iron oxide above 1800°C. Unfortunately, we cannot reach any conclusion about the relative resistance of thoria and HAC without experimental data.

Another important issue raised by the use of high alumina cement concerns water generation which will result in the production of hydrogen by the oxidation of iron.

From the data in the table, it is apparent that nine pounds of water, or more than a gallon, are used to make each cubic foot of concrete. Water constitutes 4.8% of the starting mix as compared to 5.7% by weight for limestone concrete and 3.5% for basalt concrete. In the absence of other data on HAC, we would not expect it to generate significantly less water than other concretes on being heated by molten core debris. Thermogravimetric analysis of HAC would be desirable to confirm this hypothesis. Clearly, this hydrogen generation is undesirable. If HAC offers no advantage over ordinary concrete in reducing either hydrogen generation or penetration by molten core debris, then it is necessary to question the need for it at all.

It has been suggested that the water content of the high alumina cement could be reduced by drying the concrete after installation. This seems impractical in the ZIP environment. The instrument tubes would get in the way of any drying effort and might be harmed by the heat. Further, drying the concrete would require that personnel work for extended periods of time in the reactor cavity. Finally, only the region near the surface of the concrete would be dehydrated unless the high alumina cement was heated for a long time.

Since high alumina cement has a density that is only one-half to one-third of that expected for molten core debris, there is concern about the possibility for excavation of the material by floatation in the event that extensive cracking or spallation occurs. In experiments conducted in Sandia Laboratories with molten steel at 1800°C., cracking did not occur in a way that permitted removal of the concrete and spallation was not observed.

However, although the experiments employed hundreds of pounds of molten steel, the scale of the experiments may not have been large enough to be conclusive regarding spallation. In addition, molten core debris from an accident may be a thousand degrees hotter than the molten steel in these experiments.

Given the lack of data on high alumina cement interactions with molten materials, it is not our first choice, in our initial evaluation, as a core retention material for the ZIP plants. In view of the high water content of high alumina cement, it is hard to see how it offers any improvement over the basalt concrete floor in the area of hydrogen generation. However, in situations where very large quantities of HAC can be used, the concept may deserve further attention.

3. MgO Covering Cooling Coils

In this concept, cooling coils would be installed in the reactor cavity and would be covered with a layer of MgO refractory bricks. MgO, which as been selected for a core retention system for the proposed Floating Nuclear Plant(FNP)[23,24], has a number of advantages and a few disadvantages. Figure 7 shows the MgO ladle designed for the FNP. For retrofitting purposes, the ladle could be made thinner and placed over cooling coils. The interactions between MgO and UO_2 and steel have been studied in a number of experiments[1,2,3,4,5].

The advantages include a high specific heat (0.31 cal/g°C), a high melting point (2800°C), excellent stability with respect to molten UO_2 and steel, no gas generation, ease of fabrication and relatively low cost. Its disadvantages include a low density so that it must be

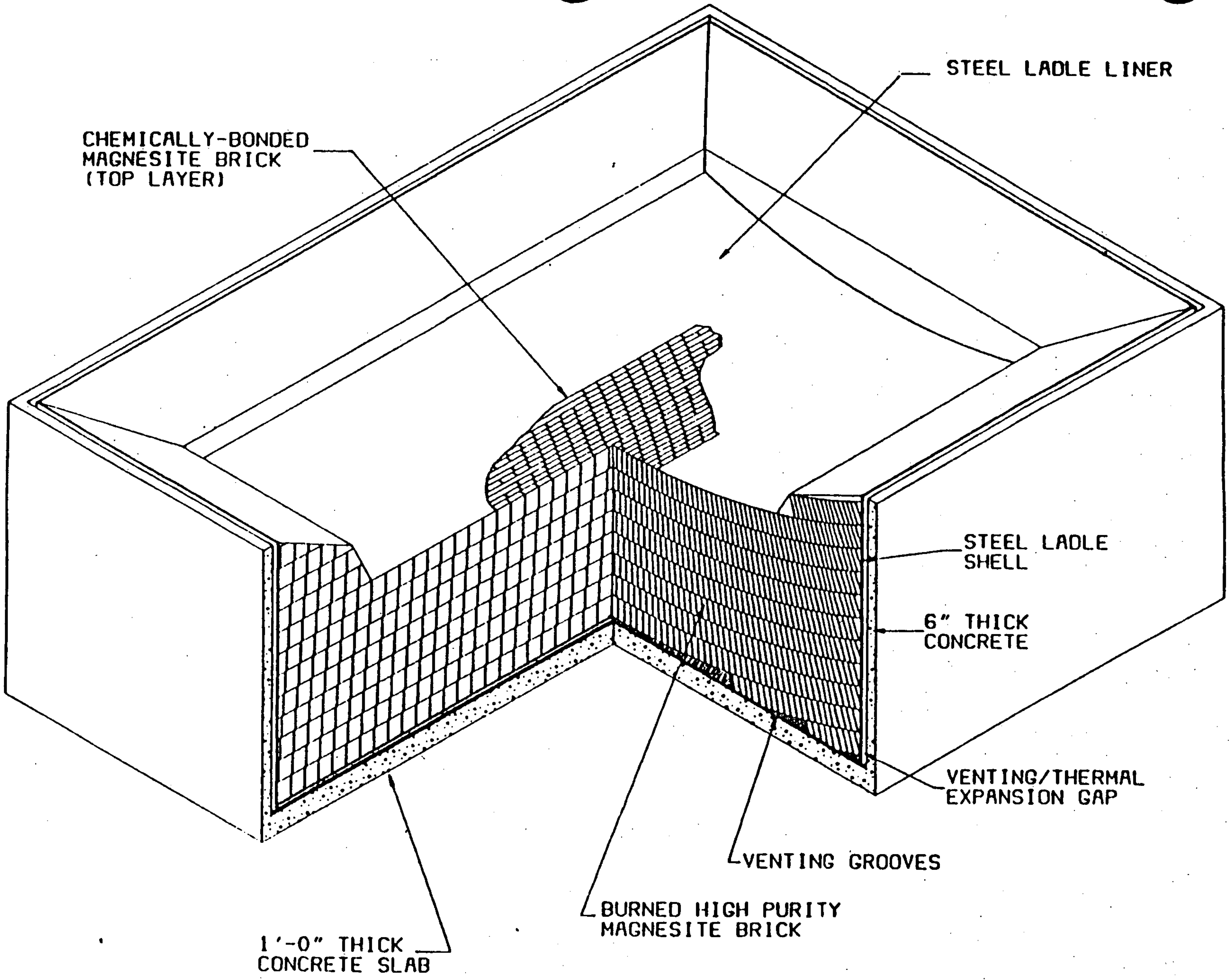


Figure 7. MgO Ladle Proposed for the Floating Nuclear Plant

mechanically restrained to avoid floatation. It also forms a eutectic with the iron oxide slag expected in the molten core debris, at 1391°C.

In the ZIP application, installation of an MgO layer would require relocation of the instrument conduits to provide adequate space. This relocation, we have been advised, is totally impractical, given the radiation environment in the reactor cavity. In addition, the bricks would have to be keylocked and formed into an inverted arch in order to prevent brick floatation. This would be time-consuming and consequently difficult in the radiation environment of the cavity.

Given the installation problems in the radiation environment of the ZIP plants, this concept is not recommended. MgO remains an outstanding choice for use in core retention systems in new plants.

4. ZrO_2 Covering Cooling Coils

This concept is similar to the preceding suggestion for the use of MgO over cooling coils. Although the density of ZrO_2 ($\rho = 5.6$ g/cc) is greater than MgO, it will still float in molten core debris and must be mechanically restrained by being used in an inverted arch structure.

The thermal conductivity of the material is much less than for MgO. Although the melting point of zirconia is 2715°C, zirconia forms a low melting eutectic with iron oxide at 1350°C. It is chemically stable with respect to molten core debris. Its disadvantages include structural instabilities due to phase changes and relatively high cost, compared to MgO. However, there is a large reservoir of industrial experience from the steel industry relating to its use at high temperatures.

Although this material should be considered for core retention devices in new plants, it is impractical for use in the radiation environment in the ZIP plants due to excessive installation time requirements and a lack of adequate space for a sufficiently thick layer under the instrument guide tubes.

5. Graphite over Cooling Coils

Graphite has also been suggested as a possible candidate material. Its advantages include a high sublimation temperature (3652°C), easy fabrication and low cost. It has several disadvantages however. It acts as a neutron moderator and thus increases the potential for recriticality. The material can react with UO_2, forming UC, UC_2, CO, and CO_2, thus generating gas and adding to the heat load because the reactions are believed to be exothermic. Finally, graphite has a very low density ($\rho = 1.5 - 2.25$ g/cc) and consequently must also be mechanically restrained to prevent floatation. We have previously alluded to the difficulties in effecting restraint of low density materials in the ZIP environment. Consequently, this concept is not recommended for retrofitting reactors.

6. Borax Bath

The borax bath system might employ a thick layer of borax bricks sealed in stainless steel cans covering the bottom of the reactor cavity. Its advantages include high neutron absorption, low cost, and high heat capacity and solubility with UO_2, thereby diluting both the decay heat and temperature of the core debris.

There are several concerns about a borax system. Its low boiling point (1600°C) makes substantial vapor generation a certainty if it comes into contact with molten UO_2 (melting point, 2750°C). In addition, it has a low density (ρ = 2.5 g/cc), so that it could float on top of a molten UO_2 layer initially although it would eventually go into solution with UO_2. Also, it might be particularly susceptible to attack by a jet of molten UO_2 or steel due to its low melting temperature (800°C). Finally, the volume of the reactor cavity and the limited space under the instrument tubes (allowing a layer 6 inches thick) is insufficient to contain the quantity of borax required to retain a molten core without penetration. A layer of borax approximately 12 feet in depth would be required for the concept to work successfully. For these reasons, this concept cannot be recommended for retrofitting the ZIP reactors although it may be acceptable in other reactor environments.

7. Heavy Metal Bath (Lead, Uranium, or Copper)

The heavy metal bath concept consists of a layer of lead bricks covering cooling coils. The advantage of this system is that the molten UO_2 cannot sink through the lead once it has melted because of its high density (ρ = ~11.3 g/cc). If the protective layer is swept away by jet impingement or a steam explosion, it will flow back into place due to its density and is thus self-healing. However, it should be noted that, unfortunately, only a thin layer (9 in) of lead can be used in the ZIP environment. Consequently, a large hot chunk of core debris could sink through the thin layer. This problem would not exist in situations where thicker layers can be used.

Another advantage is that the high thermal conductivity of lead makes heat removal much more effective. The material has a low melting point (327°C) and should freeze in the vicinity of the cooling system. If molten lead slightly above its melting point comes into contact with concrete, the damage to the concrete will be minimal. The lead bricks are easily fabricated and the softness of solid lead will tend to absorb the shock of the impact of the molten core debris.

The disadvantages of using this concept include a low boiling temperature for lead (1740°C), which will lead to vapor generation, and the toxicity of lead. In addition, in the event of a cooling system failure, this system would provide little protection to the concrete floor from a high heat flux transmitted from the molten core. The heat flux could cause concrete decomposition.

Another material suggested for use in a heavy metal bath system is uranium metal (melting point: 1132°C). Again, the high density of the metal ($\rho = 19.0$ g/cc) would keep it in place or cause it to flow back in place after a steam explosion or jet impingement. Since the boiling point is higher (3818°C), there is less prospect of a problem due to vapor generation.

The disadvantages include the radioactivity of the uranium and the lack of protection in the event of cooling system failure, since uranium metal is a good conductor of heat. In addition, some thought should be given to the maintenance of a controlled environment for this system, since the metal is pyrophoric and easily corroded. If the molten uranium metal comes into contact with oxygen, the result could be a metal fire.

Copper (melting point: 1083°C) could also be employed in a heavy metal bath and could be poured in to the cavity as copper shot. Although its density (ρ = 8.9 g/cc) is less than lead or uranium, it does have a greater density than the expected molten core debris. It is a good conductor of heat with a high boiling point (2567°C) so that vapor generation should not be a problem. It does tend to form an alloy with iron and dissolve into molten iron to a limited extent.

All of the metal bath concepts present implementation problems in the ZIP reactor environment and none can be recommended without further study, which would probably have to include an experimental program.

8. Iron Oxide

The use of a layer of iron oxide (melting temp: 1350-1600°C) as over a cooling system has also been proposed. The material is inexpensive, soluble in UO_2 and ZrO_2, and already present in the molten core debris so that chemical compatibility is not a problem.

There are a number of disadvantages to the use of iron oxide. One difficulty with the use of this material is its low density (ρ = 5.2 g/cc) which makes it impractical for retrofitting applications although the concept may have merit in future reactors. Since the density of the material is low, it will float on top of molten core debris unless it is mechanically restrained, as by forming bricks of the material into an inverted arch. As we have seen, the radiation hazard and restricted space make it impractical to construct a structure in the reactor cavity.

Another problem with the use of iron oxide in retrofitting is its low melting point. In order to contain a molten core with a low melting material, there must be space for a large quantity of the material so that the corium can be sufficiently diluted. Unfortunately, the available space for retrofitting is quite limited and only a relatively small amount of iron oxide can be placed in the cavity.

It should also be noted that the boiling point of iron oxide is unknown and could be a problem if it is significantly lower than the melting temperature of the corium mixture.

VIII. SELECTION OF A CORE RETENTION CONCEPT FOR THE ZIP REACTORS

During the course of this study, a number of core retention materials and concepts have been examined. Some important considerations in material selection, as noted earlier, were the material density, to avoid floatation, whether gas would evolve during contact with molten core debris, melting point, boiling point, and cost. Other considerations included compatibility with water, air and molten core debris and toxicity, availability and the existence of fabrication facilities.

When the particular requirements imposed by the environment in the ZIP reactor cavity are considered and used to eliminate candidate materials, remarkably few choices remain. Even if the criteria are relaxed, there are few suitable materials. None are really outstanding, in that all of the materials have drawbacks. Table 6 and Appendix A compare a number of materials from the point of view of their suitability for use in molten core retention under the specific constraints of the ZIP reactor only. Table 6 presents a one page summary of the material comparisons and the table in Appendix A amplifies on Table 6 at considerable length.

The best of the available materials for use in retrofitting the ZIP plants, in our view, is thoria. It will not float in molten core debris and has a high melting and boiling point. The material does not evolve gas and is the most chemically inert material of any studied in a molten core debris environment. Its disadvantages include a moderately expensive raw material ($16.40/kg) and substantial fabrication costs. It is very mildly radioactive

Table 6. Comparison of Materials for Molten Core Retention
- Summary -

	Floatation	Low melting point	Low boiling point*	Gas Evaluation (H_2O, CO_2)	Time for Installation (radiation hazard)	Cost	Availability of fabrication facilities	Mechanical stability	Chemistry (water)	Chemistry (molten core debris)*	Thermal, mech. shock	Toxicity	Additional radioactivity	Pyrophoric
ThO_2						X	X						X	
UO_2						X		X			X		X	
MgO	X				X				X					
HAC	K	X		X						X	X			
Pb		K	X									X		
Cu		X												
Fe	K	X			X				X					
FeO	X	X			X									
Ni		X								X				
Mo						X			X	X				
U		X				X		X				X	X	X
Graphite	X			X	X				X	X				
ZrO_z	X				X	X		X	X					
Borax	X	X	X	X	X			X	X		X			

X = problem

K = known problem, but not relevant in this application

* = area where more information is needed

but is in the hands of the public in the form of gasoline lantern mantles.

In selecting a concept for retrofitting the ZIP reactors, it is desirable to reduce the exposure of workers to radiation to a minimum. This means that the installation time must be as short as possible. Other problems include a very limited space available which contains high pressure instrument guide tubes that cannot be moved. It is also desirable that the concept selected offer some protection in the event of a cooling system failure. It should be resistant to jet penetration and offer as little prospect as possible of a vapor explosion and the generation of gas, particularly hydrogen. Table 7 and the Table in Appendix B compare some possible core retention concepts for the ZIP reactors. Table 7 is a one page summary of the concepts and Appendix B provides details and further information on each concept. The comparison of core retention concepts presented here is applicable only to the specific environment of the ZIP reactors. In other environments, the concepts would rank and compare differently.

The best concept examined, in our opinion, is the flooded thoria rubble bed with a water supply. It can be rapidly installed by pouring pellets of the material in place with minimum radiation exposure to personnel. Water can be forced through the bed with minimal difficulty. There are adequate supplies of the thoria and facilities are available for its fabrication. The expected cost of implementation is not unreasonable.

It is important to note that the system will provide protection against concrete penetration for a time even if

Table 7: Comparison of the Suggested Core Retention Systems for the ZIP Reactors - Summary (1)

	Cost	Installation Time Required (affects radiation dose)	Resistance to Corium Jet Penetration	Vapor Explosion Potential (Cooling system intacts)(2)	Hydrogen Production (Note (3))	Protection Provided Against Corium if Cooling System Fails Temporarily
Flooded Thoria Bed	High	Short	Fair	Depends	High (Requires control measures)	High
ThO_2 or UO_2 over cooling coils	High	Moderate	Fair	Low	None	High
HAC over cooling coils	Low	Moderate	Good	Low	High	Moderate
MgO Bricks over cooling coils	Low	High	Good	Low	None	High
ZrO_2 Bricks over cooling coils	Moderate	High	Good	Low	None	High
Graphite Bricks-coils	Low	High	Poor	Low	Depends	Unknown

<u>Notes</u>

(1) Please note that this table only applies to the specific conditions found in the ZIP reactors, as discussed in the text and as amplified in Table IV. Note that all concepts, with one exception, require cooling either by cooling coils or by transpiration. In other environments, this comparison would yield different results.

(2) A potential for a vapor explosion exists where the material boiling point is substantially less than the temperature of molten urania (2750°C.). It should be noted that an explosion is regarded as possible but not probable. The available information is insufficient to make any further judgement.

(3) The question asked concerns whether the concept contributes to hydrogen generation in the cavity. It is assumed that cooling coils are intact. The reaction between molten core debris and reactor cooling system water to produce hydrogen is not considered in this column.

Table 7: Comparison of the Suggested Core Retention Systems for the ZIP Reactors - Summary (1) (Cont.)

	Cost	Installation Time Required (affects radiation dose)	Resistance to Corium Jet Penetration	Vapor Explosion Potential(Cooling system intact)(2)	Hydrogen Production (Note (3))	Protected Provided Against Corium if Cooling System Fails Temporarily
Borax Bath	Low	High	None	High	None	None
Pb Bath over cooling coils	Low	Moderate	Poor	High	None	Questionabl
Cu Bath over cooling coils	Low	Moderate	Poor	Low	None	None
U metal bath over cooling coils	High	Moderate	Poor	Low	Possible	None
FeO over cooling coils	Low	High	Questionable	Low	Low	High
HAC/ThO_2/HAC over cooling coils	High	Moderate	Good	Low	Low	High

the arrival of water is temporarily delayed. Most importantly, the flooded thoria bed concept should accomplish the objective of core retention in terms of preventing core-concrete interactions coupled with the concommittant mitigating effects on containment pressurization, hydrogen generation, and basemat melt-through.

The concept does have several disadvantages as is the situation with all of the choices. There are questions involving iron oxide-thoria interactions and the response of the rubble bed to jet impingement and fuel/coolant interactions. Another important issue concerns possible hydrogen generation from the water in the flooded bed. We presently consider that hydrogen control in the containment would be required even for the flooded thoria rubble bed system. However, it should be noted that hydrogen will not be generated by interactions between the water and iron if the iron can be cooled below 550°C by the water.

It should be noted again that the choice of concepts has been made for the specific environment of the ZIP reactors. In other reactor environments, or in new construction, the concept selected would probably be different.

Some additional confirmatory research is needed to confirm the viability of the concept proposed here. Information is needed on the interactions between thoria and core debris materials. In particular, it would be desirable to know whether thoria and iron oxide form a low-melting eutectic. It is of interest to determine if jet impingement or melt streaming will disrupt loose gravel of a certain pre-determined shape or gravel held in place by a restraining structure. Also, the effect of a steam explosion on a restrained and an unrestrained gravel bed is of interest. Another question concerns the potential for a melt trickling through a porous gravel bed. The rate of hydrogen and any gas generation should be determined from concrete cavities protected by a thoria gravel bed. The preceding experiments should be done with both dry and wet gravel beds. The tests should also examine the sustained interactions of corium melts with gravel beds to determine the heat flux partitioning that results.

With the performance of the above limited confirmatory research, it is concluded that the proposed flooded thoria bed concept is feasible and has the greatest potential for success of any concept considered for retrofitting the ZIP reactors. This concept, coupled with a hydrogen control system, will reduce significantly the radiological consequences and risks by preventing containment failure by basemat melt-through, overpressurization and hydrogen burning.

There may be other, better ways of containing a molten core beside the concepts considered here. Multilayer systems, for example, should be considered further. Additional study of these possibilities is desirable.

IX. A THRESHOLD FOR HYDROGEN GENERATION IN CORE MELT ACCIDENT SEQUENCES

The conditions required for the generation of hydrogen from steam in contact with hot metal are of considerable interest in studies of possible accident mitigation measures. In the Zion and Indian Point plants, water is expected to be present in the reactor cavity during normal operation, as the plants are currently designed. The utilities operating these reactors have proposed flooding the reactor cavity after a postulated accident has occurred. Water would be introduced either before or after the molten core debris enters the cavity in order to cool the molten material. Core retention concepts, such as the flooded bed of rubble, would also bring core debris and water into contact. Also, water might be introduced into a core retention system after the molten core arrives in the cavity to provide additional cooling. Even if water is not intentionally introduced, steam will be generated as the concrete floor of the reactor cavity is heated and dehydrated, unless the concrete is protected in some manner.

In any of these situations, steam will come in contact with hot metal (steel) from the molten core debris, causing corrosion or oxidation of the metal. Hydrogen is a product of the oxidation of metal by water and its generation can lead to containment failure if conditions arise which permit it to burn.

The quantity of hydrogen that will be generated if steam comes into contact with hot metal depends on a number of factors. One factor is the temperature of the metal. If steam comes into contact with molten metal, oxidation of the metal will occur rapidly accompanied by the generation of hydrogen. If the metal is solid, the extent of oxidation is less certain. An initial oxide film on the surface of the metal may serve to inhibit further oxidation. Also, the constituents of metal alloys may effectively inhibit oxidation. As a consequence of these factors, there will be a minimum temperature which must be exceeded before substantial oxidation of the metal, and hydrogen generation, can occur.

In the course of the following discussion, the oxidation of steel in air will be discussed initially because there is more data available for this situation. In general, the relative corrosion resistance of various steels to steam is the same as it is in air.

One of the simplest forms of corrosion is the oxidation of metal in air which occurs by a diffusion of oxygen inward and alloying elements outward. In plain carbon steel, the amount of oxidation in air is negligible below 1000°F. Above this temperature, the oxidation rate rapidly increases.

The oxidation resistance of carbon steel above 1000°F can be increased by alloying it with chromium. Chromium will oxidize preferentially over iron forming a tightly adhering layer of a chromium-rich oxide on the surface of the steel. The layer serves to inhibit further oxidation by retarding the inward diffusion of oxygen. The oxidation resistance can also be increased by adding other elements, such as aluminum and silicon, which are oxidized preferentially over iron.

Figure 8 illustrates the progress of oxidation for carbon steel and 5 percent chromium steel, containing molybdenum, at 1100°F. Although both steels oxidize rapidly initially, the oxidation rate of the chromium steel decreases rapidly as a chromium-rich oxide layer forms. It is generally true that the corrosion resistance of steel in various atmospheres will increase with chromium content. Figure 9 indicates the maximum amount of chromium required for freedom from oxidation at temperatures from 1000° to 2000°F.

In steam, the chromium content of the steel is also a primary factor in determining its resistance to oxidation. Figure 10 compares the relative corrosion of various steels in steam. At temperatures below 560°C, there is little corrosion of the metal by steam and consequently little hydrogen will be generated. Some steels containing large quantities of chromium will not corrode significantly at temperatures as high as 900°C and, consequently, the hydrogen evolution rate should also be minimal below this temperature for these alloys. Clearly, chromium has a profound effect on the corrosion resistance of steel in steam. Although silicon is beneficial in improving the oxidation resistance of steel in air, this element does not improve oxidation resistance in steam.

It should be noted that high temperature hydrogen can attack steel by diffusion into the metal where it reacts with any carbon present to form methane. Methane generation could add to the flammability problem if significant quantities are generated. As indicated in Figure 11, this reaction will not constitute a problem with the relatively low hydrogen pressures expected in the reactor containment.

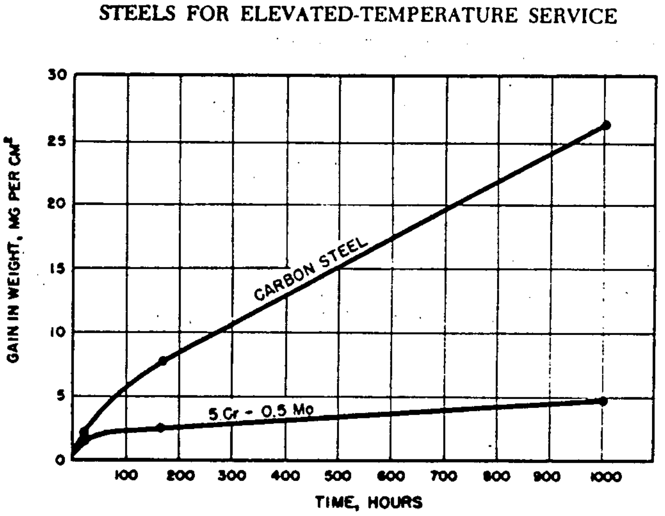


Figure 8: Oxidation of Plain Carbon and 5 Percent Chromium - 0.5 Percent Molybdenum Steel at 1100°F. (Ref 25)

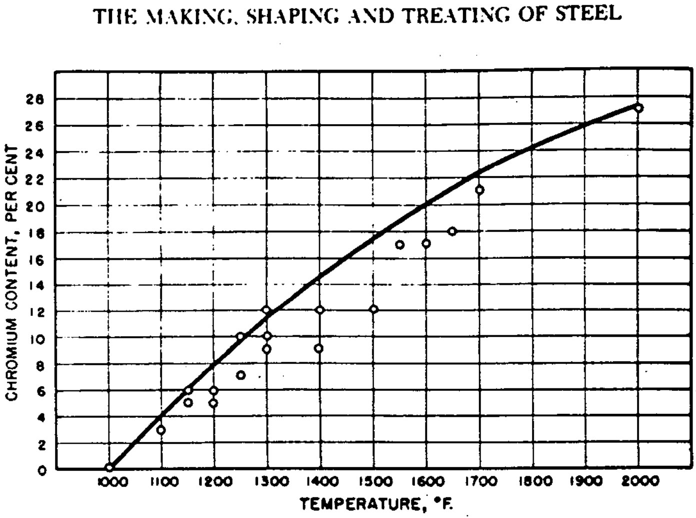


Figure 9: Maximum Amount of Chromium Needed for Freedom from Oxidation at Temperatures from 1100°F to 2000°F. (Ref 25)

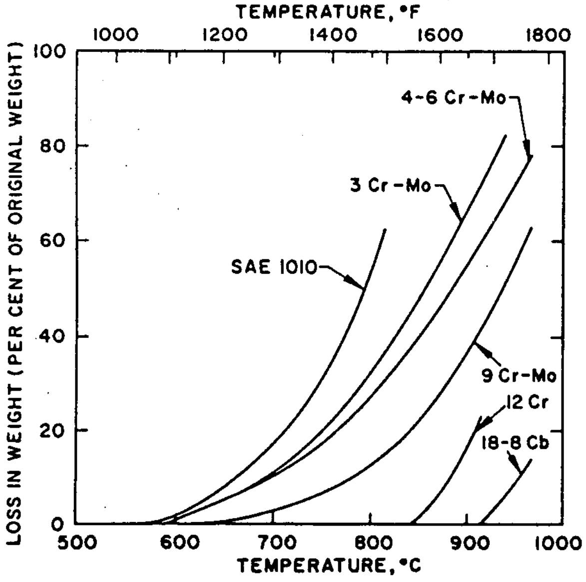


Figure 10: Corrosion of Steel Bars in Contact with Steam for 500 Hours at Various Temperatures. (Ref 26)

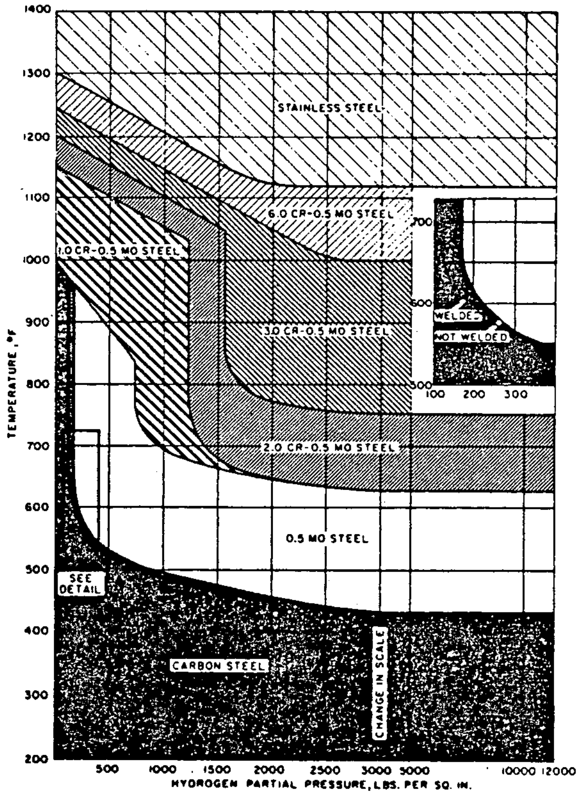


Figure 11. Safe Operating Zones for Steel in Hydrogen. (Ref 27)

In considering the cooling of core debris in the Zion and Indian Point reactors, the reaction of water with molten steel is frequently mentioned in terms of the hydrogen explosion hazard. The upper and lower flammable and explosive limits for hydrogen are generally taken to be 4% and 74% by volume. The lower limit of oxygen concentration at which burning can take place has been reported to be 4.5 to 5% by volume.

Assuming an air atmosphere at the time of core disruption, several factors work towards reducing the explosion hazard. The generation of steam from contact between water and molten core debris by itself lowers the oxygen concentration by dilution. The molten core debris, especially molten metals such as steel, provide a strongly reducing environment. Molten steel reacts with oxygen gas even more readily than it does with water, further reducing the containment building's atmospheric oxygen concentration.

In the event that a pool of molten steel is formed with steam bubbling through it, the most efficient method envisioned for generating hydrogen, and in the event that the atmosphere above the steel still contains more than 4% oxygen by volume, it is most likely that the hydrogen will simply burn at the surface of the steel. The white hot steel provides the ignition. The hydrogen would be expected to continue burning until the oxygen concentration in the containment dropped to below 4 or 5%.

Under these strongly reducing conditions, the accumulation of oxygen will be less than might otherwise be expected. With the building at a pressure greater than one atmosphere, no additional oxygen from leakage of air into the containment is possible.

In conclusion, hydrogen generation should be insignificant for solid steel in contact with steam at a temperature below 560°F. At higher temperatures, the extent of hydrogen generation will be dependent on the amount of chromium in the steel alloy.

X. REFERENCES

1. D.G. Swanson, E.H. Zehms, C.Y. Ang, J.D. McClelland, R.A. Meyer H.L.L. van Paassen, Annual Progress Report. Ex-Vessel Core Catcher Materials Interactions, ATR-77(7608)-1, The Aerospace Corporation (October 1976).

2. D.G. Swanson, D.M. Goddard, E.H. Zehms, J.D. McClelland, R.A. Meyer and H.L.L. van Paassen, Annual Progress Report. Evaluation of Materials for CRBRP Core Retention, NUREG/CR-0076, The Aerospace Corporation (May 1978).

3. D.G. Swanson, E.H. Zehms, R.A. Meyer, J.D. McClelland and H.L.L. van Paassen, Annual Progress Report. Evaluation of Materials for Retention of Sodium and Core Debris in Reactor Systems, NUREG/CR-0900, The Aerospace Corporation.

4. Exploratory Study of Molten Core/Material Interactions, July 1975-March 1977, SAND 772042, Sandia Laboratories (February 1978).

5. D.A. Powers and F.E. Arellano, Large Scale Molten Steel Interactions With Concrete, Sandia National Laboratories, Albuquerque, NM, in press

6. L. Baker, Jr. et al. Post Accident Heat Removal Technology, Argonne National Laboratory, ANL/RAS-77-2 (January 1977)

7. S.A. Meacham, Molten Stainless Steel and Urania Compatibility with Refractory Materials Considered for CRBRP Parallel Design Ex-Vessel Core Retainers, WARD-D-0098, Westinghouse Advanced Reactors Div. (October 1975).

8. J.B. Rivard, Review of In-Vessel Meltdown Models, SAND80-0455, Sandia National Laboratories, 1980

9. F. Mayinger, Untersuchung Thermohydraulischer Vorange Sowie Warmeaustausch in der Vienschelez, Abschlussberecht TUH R548/1, (September 1975)

10. Light Water Reactor Safety Research Program Quarterly Report, April-June 1980, SAND80-1304 (NUREG/CR-1509) Sandia National Laboratories (August 1980)

11. Report of the Zion Indian Point Study, NUREG/CR-1410, Sandia Laboratories, 1980.

12. C.R. Weast, CRC Handbook of Physics and Chemistry, CRC Press, Inc. Boca Raton, 1980.

13. Mellor, J.W., A Comprehensive Treatise on Inorganic and Theoretical Chemistry, Longmans, Green and Co., London (1960).

14. Peterson, S. and Curtis, C.E., Thorium Ceramics Data Manual, Oxides, ORNL-4503, 1, (1970).

15. Benz, R.J. Journal of Nuclear Materials, North-Holland Publishing Co, Amsterdam, 29, 43 (1969).

16. Samsonov, G.V., The Oxide Handbook, IFI/Plenum Data Corp., New York, (1973).

17. Barbesino, F., Brutto, E., Di Pietro, R., Gioia, A., Perona G. and Sesini., R., UO_2 Fuel Elements Shaped as Small Spheres, Proceedings of the Third International Conference on The Peaceful Uses of Atomic Energy, 11, 547, United Nations, New York, (1964)

18. A. DeF. Quinn, Design and Construction of Ports and Marine Structures, McGraw-Hill, New York, 1961.

19. P. Daniel, E. Chapus, and R. Dhaille, Tetrapods and other Precast Blocks for Breakwaters, J. Waterways Harbors Div. Am. Soc. Civil Engrs., 86, 1 (1960).

20. Design of Breakwaters and Jetties, in U.S. Army Engineering Manual, EM 1110-2-2904, U.S. Army Corps of Engineers, Washington, DC, 1963.

21. S. Nagai, Stable Concrete Blocks on Rubble Mound Breakwaters, Proc. Amer. Soc. Civil Engrs., 88, 85 (1962).

22. E. McLevin, C.R. Robbins and H.F. McMurdie, Phase Diagrams for Cements, American Ceramic Society, Columbus, Ohio, 1975.

23. Final Environmental Statement, Floating Nuclear Plants, NUREG-0502, USNRC, 1978.

24. FNP Core Ladle Design and Safety Evaluation, Offshore Power Systems, Topical Report 36A59, Jacksonville, 1979.

25. The Making, Shaping, and Treating of Steel, edited by H.E. McGannon, U.S. Steel Corporation, Pittsburgh (1971)

26. G.A. Hawkins, J.T. Agnew, and H.L. Golberg, Trans, ASME 66, 291 (1944).

27. G.A. Nelson, Corrosion Data Survey, Shell Development Co. (1960).

APPENDIX A: Comparison of Materials for Core Retention

Table 8: Comparison of Materials

	Density (gm/cc)	Floatation Resistance*
ThO_2	9.86	Yes
UO_2	10.96	Yes
MgO	3.5	No
Al_2O_3	3.97	No
Pb	11.34	Yes
Cu	8.92	Yes
Fe	7.86	No (Marginal)
FeO	5.7	No
Mo	10.2	Yes
Ni	8.9	Yes
U	19.05	Yes
Graphite	F3.51	No
ZrO_2	5.7	No
Borax	1.73	No

* The density of molten core debris is assumed to be less than 8.0 g/cc

Table 8: Comparison of Materials (cont.)

	Melting Point (°C.)	Melting Penetration Resistance
ThO_2	3220	Excellent
UO_2	2878	Excellent
MgO	2852	Excellent
Al_2O_3	2072	Good
Pb	327	Poor
Cu	1083	Good (Because of high conductivity if covering a cooling system)
Fe	1535	Fair (Lower conductivity)
FeO	1369	Poor (Lower conductivity)
Mo	2617	Excellent
Ni	1453	Fair
U	1132	Fair (Due to high conductivity if covering a cooling system)
Graphite	J3550	Poor; Sublimes; (Dissolves in molten iron and reacts with urania)
ZrO_2	2715	Excellent
Borax ($Na_2B_4O_7 \cdot 10H_2O$)	Dehydrates at 60, 320, 741°C	Poor
($Na_2B_4O_7$)		Poor

Table 8: Comparison of Materials (cont.)

	Boiling Point (°C.)	Vapor Explosion Potential*
ThO_2	4400	None
UO_2	DecJ3000	None
MgO	3600	None
Al_2O_3	2980	None
Pb	1740	Moderate
Cu	2567	Slight
Fe	2750	None
FeO	Unknown	None
Mo	4612	None
Ni	2732	None
U	3818	None
Graphite	4827	None
ZrO_2	3000	None
Borax		
($Na_2B_4O_7 \cdot 10H_2O$)	Dec. at 60, 320°C	Moderate (Tends to foam)
($Na_2B_4O_7$)	Dec. 1575°C	

* A potential for a vapor explosion exists where the material boiling point is substantially less than the temperature of molten urania (2750°C). It should be noted that an explosion is regarded as possible, but not probable. The available information is insufficient to make any further judgement.

Table 8: Comparison of Materials (cont.)

Gas Evolution

ThO_2	None
UO_2	None
MgO	None
Al_2O_3	None
Pb	None (except Pb vapor at high temperature)
Cu	None to slight (very slow production of H_2 with water possible)
Fe	Produces H_2 in contact with water (rapid when molten)
FeO	None to slight
Mo	Produces H_2 in contact with water (rapid at high temp) Also forms a volatile oxide
Ni	Produces H_2 in contact with water when molten but not as rapidly as Fe. Volatile nickel carbonyl forms if CO is present
Graphite	Water gas reaction generates CO and H_2
ZrO_2	None
Borax	$Na_2B_4O_7$ 10 H_2O liberates water of hydration at a low temperature

Table 8: Comparison of Materials (cont.)

Time Required for Installation

ThO_2	In gravel or lump form, thoria could be easily poured in.
UO_2	Urania could be poured in but since it exists either in powder or pressed powder form some problem is anticipated with radioactive dust. If it is used, it should be encapsulated.
MgO	Because of its low density, magnesia would most likely have to be installed in the form of anchored bricks. This would require a considerable installation time.
Al_2O_3	A high alumina concrete could be poured, but this would melt slightly above the melting point of iron.
Pb	Could be poured in if shot is used.
Cu	Could be poured in in the form of shot.
Fe	If poured in in the form of shot, it could float in molten core debris. If installed as plate, a a long installation period would be required.
FeO	Could be poured in in the form of gravel but this would float in core debris. Consequently, it must be installed so that it is mechanically restrained and this will require a considerable time.
Mo	Could be poured in.
Ni	Could be poured in in the form of shot.
U	Could be poured in.
Graphite	A substantial installation time is required as it must be anchored to prevent floatation.
ZrO_2	Would require installation in the form of anchored bricks.
Borax	Could be poured in, but should be encapsulated and and set in the form of brick shaped cans due to its solubility in water.

Table 8: Comparision of Materials (cont.)

	Cost
ThO_2	\$7.00-14.00/lb, depending upon purity, plus fabrication costs
UO_2	\$45/lb (natural urania; the price of depleted urania is artificial)
MgO	\$0.15/lb (harklase brick)
Al_2O_3	\$0.17 to \$0.30/lb
Pb	\$0.45/lb
Cu	\$1.01/lb
Fe	\$0.10 to \$0.30/lb (depending upon purity, etc.)
FeO	\$0.10 and up (function of purity)
Mo	\$15.00/lb (in large quantities)
Ni	\$0.25 to 0.50/lb (depending upon purity)
U	\$45.00/lb
Graphite	Varies greatly
ZrO_2	\$3.00/lb (depends upon purity, whether stabilized, etc.)
Borax	\$0.50 to \$1.00 (may be cheaper in large quantities)

Table 8: Comparison of Materials (cont.)

Availability

ThO_2	Mineral sources of thorium are widespread with important deposits in Brazil, India, and the United States.
UO_2	Depleted UO_2 available in the USA primarily as a by-product of the nuclear industry.
MgO	Magnesia products are readily available from several manufacturers in the USA. Products developed for the metals refining industry are well suited to core catcher technology.
Al_2O_3	Mineral sources are plentiful and advanced technology exists in the USA for producing alumina products.
Pb	Lead is readily available in a variety of forms including lead shot at a cost only slightly above the bulk material cost.
Cu	Copper is readily available in a variety of useful shapes.
Fe	Iron is the most readily available and the least expensive of all metals.
FeO	Available as a naturally occurring mineral or as a by-product of the steel industry (slag). Its price is less than that or iron.
Mo	Readily available in the USA. The current annual production is approximately 108 lb/yr.
Ni	Readily available from Canada, which produces 60% of world supply.
U	See UO_2 above
Graphite	Readily available; used for electric arc electrodes in steel mills.
ZrO_2	Readily available in refractory grade purity in various forms.
Borax	Readily available in high purity form.

Table 8: Comparison of Materials (cont.)

Availability of Production Facilities

ThO_2	Not available in the USA at present although reactor fuel fabricators have expressed a willingness to convert facilities designed for urania to process thoria.
UO_2	Currently available
MgO	Currently available
Al_2O_3	Currently available
Pb	Currently available
Cu	Currently available
Fe	Currently available
FeO	Currently not available as no need exists, but facilities could readily be set up.
Mo	Currently available
Ni	Currently available
U	Currently available
Graphite	Currently available
ZrO_2	Currently available (Corning)
Borax	The raw material is readily available and facilities for encapsulation could readily be set up.

Table 8: Comparison of Materials (cont.)

Mechanical Stability

ThO_2	Good	Used in free-standing crucibles
UO_2	Poor	Available in powder form only
MgO	Good	Used in free-standing crucibles
Al_2O_3	Good	Used in free-standing crucibles
Pb	Fair	Melts easily; low yield strength
Cu	Fair-Good	Modest yield strength
Fe	Good	High yield strength
FeO	Fair-Good	Can be cast
Mo	Good	High yield strength
Ni	Good	High yield strength
ZrO_2	Good (if stabilized)	Used in free-standing crucibles
U	Poor	Undergoes two crystalline phase changes
Graphite	Good	
Borax	Poor	Requires encapsulation

Table 8: Comparison of Materials (cont.)

Reactions with H_2O

ThO_2	None	Inert
UO_2	None	Inert
MgO	Slight	Dissolves in H_2O as a function of the temperature and purity of the MgO
Al_2O_3	Slight to None	
Pb	None	Essentially Inert
Cu	None	Slight oxidation depending on conditions
Fe	Is Oxidized and forms H_2	Reaction rapid above melting point
FeO	Slight-None	Depends on H_2 concentration
Mo	Is Oxidized and forms H_2	Also forms volatile MoO_3
Ni	Is Oxidized and forms H_2	Reaction rapid above melting point.
U	Is Oxidized	
Graphite	Forms H_2 and CO	Water gas reaction
Borax	Sol. in H_2O	Hydration water removed at 320°
ZrO_2	Slight-none	Some hydration

Table 8: Comparison of Materials (cont.)

Compatibility with Molten Core Debris

ThO_2	Compatible with all components of core debris with the possible exception of FeO. If it is soluble in FeO, the eutectic would probably be J1450°C.
UO_2	Compatible with everything except FeO with which it forms a eutectic near 1300°C.
MgO	Forms a eutectic with UO_2, ZrO_2 and especially with FeO (at 1300°C for FeO).
Al_2O_3	Dissolves most oxides, especially FeO, near 1300°C.
Pb	Chemically compatible with all core debris but has a relatively low boiling point.
Cu	Chemically compatible with a possible slight reaction with water.
Fe	Reacts with water to form hydrogen and produces FeO which dissolves almost all other oxides at 1300°C.
FeO	Forms a eutectic with nearly all oxides, including UO_2 and ZrO_2 at 1300°C.
Mo	Reacts with water to form hydrogen and a volatile oxide. Also readily dissolves in molten iron.
Ni	Reacts with water to form hydrogen. Also readily dissolves in molten iron.
U	Can be readily oxidized and is pyrophoric in air, especially when present as fine particles.
Graphite	Reacts with urania and is dissolved by molten iron.
ZrO_2	Dissolves in FeO near 1300°C.
Borax	Dissolves in water. Dissolves all oxides, including concrete, at temperatures of approximately 800°C.

Table 8: Comparison of Materials (cont.)

Thermal and Mechanical Shock Resistance

ThO_2	Good
UO_2	Poor
MgO	Good
Al_2O_3	Good
Pb	Excellent
Cu	Excellent
Fe	Excellent
FeO	Fair
Mo	Excellent
Ni	Excellent
ZrO_2	Fair to Good (if stabilized)
Borax	Poor to Fair

Table 8: Comparison of Materials (cont.)

Thermal Conductivity
(W/cm-°C.)

		Comments
ThO_2	0.13 at 0°C to 0.015 at 2000°C.	
UO_2	0.008 at 0°C. to 0.2 at 2000°C.	Powder
MgO	0.6 at 0°C. to 0.05 at 2000°C.	
Al_2O_3	0.003	Powder
Pb	0.4	
Cu	4.0	
Fe	0.8 at 0°C. to 0.3 at 1545°C.	
FeO	0.06 ?	Value is not well known
Mo	1.2 to 1.0	
Ni	0.90	
U	0.275	
Graphite	0.06-21.0	(Depends on manufacturing process; anisotropic)
ZrO_2	0.015 to 0.020	
Borax	0.012 (estimated; data is not available)	

Table 8: Comparison of Materials (cont.)

Toxicity

ThO_2	Radiation Hazard only. No chemical toxicity reported.
UO_2	Radiation Hazard, inhalation, ingestion.
MgO	Irritant; Ingestion-slight; Inhalation-moderate
Al_2O_3	Irritant; Ingestion-slight; Inhalation-moderate
Pb	High: acute and chronic by ingestion and inhalation
Cu	Slight irritant and allergen
Fe	Slight; acute systemic by inhalation
FeO	Slight; acute systemic by inhalation
Mo	Slight; acute local irritant
Ni	Slight; acute local irritant and allergen by inhalation and ingestion
U	Heavy metal poisoning is possible
ZrO_2	No chemical toxicity reported
Borax	Moderate local irritant by ingestion and inhalation

Table 8: Comparison of Materials (cont.)

	Additional Radioactivity Produced
ThO_2	Slight to moderate
UO_2	Moderate
MgO	None
Al_2O_3	None
Pb	None
Cu	None
Fe	None
FeO	None
Mo	None
Ni	None
U	Moderate
Graphite	None
ZrO_2	None
Borax	None; reduces chances of recriticality

Table 9: Comparision of Core Retention Systems for the ZIP Reactor

Flooded Thoria Bed with Water Supply

Cost: Moderately high. The basic cost of thorium and fabrication of thoria pellets are both high.

Installation Time: Short. Thoria pellets could be poured in or blown in place using a hose and high pressure air.

Resistance to Floatation: Yes. Thoria has a density greater than any component of the corium and consequently does not require a restraint to prevent floatation.

Vapor Explosion Potential: Varies. If the water level remains below the top of the bed, there is no problem. If the water level is higher than the top of the bed, a steam explosion becomes possible. The bed can be held in place during a steam explosion through the addition of a wire mesh grid on top of the bed.

Hydrogen Production: Yes. Steam from the flooded bed is forced through the molten steel in the core debris and readily produces hydrogen and iron oxides. Hydrogen control measures are required.

Resistance to Jet Penetration: Fair. A jet of molten corium, steel or urania could displace the thoria particles.

Temporary Failure of Cooling System: Good. Protection is provided by the high melting temperature of the thoria if the bed dries out temporarily. Thoria is not attacked by urania, steel, or zirconia below 2000°C. Subsequent formation of iron oxides by the reaction of steel with water may lead to solution of thoria in FeO at temperatures J1450°C.

Table 9: Comparison of Core Retention Systems for the ZIP Reactors (cont.)

Thoria or Urania Pellets over Cooling Coils

Cost: Moderately High. Both thoria and urania have moderately high costs and fabrication is expensive.

Installation Time: Moderate. Both thoria and urania pellets could be blown in place using a hose and high pressure air. However, a prefabricated cooling coil system will require installation.

Resistance to Floatation: Yes. Both thoria and urania have densities greater than that expected for corium.

Vapor Explosion Potential: Low. Unless the cooling coils are penetrated, there is no vapor explosion potential as long as the cavity remains dry.

Hydrogen Production: No. Neither urania or thoria react with any component of corium to produce hydrogen. The concept does not add to the hydrogen generation problem.

Resistance to Jet Penetration: Fair. Both thoria and urania pellets might be displaced by a jet of molten corium or falling solid debris, thereby exposing the cooling coils.

Temporary Failure of the Cooling System: Good protection against temporary failure is provided by the high melting points of the materials above 2000°C. Neither urania nor thoria are soluble in corium. Subsequent formation of FeO, if water is introduced, would cause solution at J1450°C.

Table 9: Comparison of Core Retention System for The ZIP Reactors (cont.)

High Alumina Concrete (HAC) Over Cooling Coils

Cost: Low. The cost of HAC is quite modest, especially when compared with urania and thoria.

Installation Time: Moderate. The cement can be pumped in but time is required for the installation of the cooling coils.

Resistance to Floatation: Yes. Although the density of concrete is less than that of corium, it forms a monolithic layer. Only that concrete thermally decomposed or mechanically disloged is expected to float.

Vapor Explosion Potential: Low. Exposure of the HAC to molten corium is expected to produce an evolution of steam which reduces the thermal contact between the corium and HAC. No explosion is expected, especially since HAC has been used to contain molten steel, providing the cavity is free of pools of water.

Hydrogen Production: Yes. Water will be produced by heating high alumina cement. The water will react with hot iron in the core debris forming substantial quantities of hydrogen.

Resistance to Jet Penetration: Fair. A jet of molten corium is expected to decompose the low melting cement surface and remove it by ablation. The situation is helped somewhat by the relatively high thermal conductivity of the concrete which will allow withdrawl of heat by the cooling system.

Temporary Failure of the Cooling System: Moderate. If the molten core debris temperature exceeds the melting temperature of the concrete, then the concrete will be rapidly penetrated.

Table 9: Comparison of Core Retention Systems for the Zip Reactors (cont.)

Magnesia or Zirconia Over Cooling Coils

Cost: Magnesia-Low; Zirconia-Moderate. Both materials are commercially available for applications similar to this one.

Installation Time: Long. Either material floats in molten core debris and consequently, bricks of the material require mechanical restraint, as by being made into an inverted arch. Pellets of the material would float away and would consequently be unsuitable. A cooling system is required in the ZIP application to remove heat and would have to be installed from prefabricated panels.

Resistance to Floatation: No. Both magnesia and zirconia in granular form would readily float in molten corium.

Hydrogen Generation: No. Neither magnesia nor zirconia react with corium to produce hydrogen or any other flammable gas.

Resistance to Jet Penetration: Yes. A brick inverted arch structure would be resistant to jet impingement. Neither material is soluble in urania below 2000°C. Later formation of FeO would cause solution down to 1300°C.

Temporary Failure of Cooling System: Good protection would be provided by either high melting refractory material.

Table 9: Comparison of Core Retention Systems for the ZIP Reactors (cont.)

Borax (Anhydride) in Steel Cans over Cooling Coils

Cost: Low. The cost of borax very low; the only signficant expense is that of canning the borax.

Installation Time: Long. The borax cans must be anchored to prevent floatation. If uncanned borax is used, it will dissolve in water and float in molten core debris.

Resistance to Floatation: No. Both borax and borax in cans would readily float unless anchored.

Vapor Explosion Potential: High. Since the melting point of molten core debris is 2000°C higher than the boiling point of borax, a vapor explosion must be assumed to be possible. Whether one is also probable is unknown.

Hydrogen Production: No. Borax (anhydride) produces no hydrogen when in contact with molten corium.

Resistance to Jet Penetration: No. The initial concept of the borax bath required a layer some 3 meters thick (not 9") which would dissolve and dilute all of the oxide constituents of corium. The borax bath forms liquid systems with oxide components of corium down to 800°C.

Temporary Failure of Cooling System: No protection is afforded due to the low melting point of borax. Note that this result is a consequence of the limited volume available in the ZIP systems. With other conditions (a large volume of borax), no cooling would be needed.

Table 9: Comparison of Core Retention System of the ZIP Reactors (cont.)

Graphite Over Cooling Coils

Cost: High. The cost of graphite varies considerably depending on the grade used; it must be anchored.

Installation Time: High. Since graphite will float in corium, it must be mechanically restrained. A cooling coil system must also be employed.

Resistance to Floatation: Poor. Mechanical restraint is required to prevent floatation.

Vapor Explosion Potential: None in a dry cavity.

Hydrogen Generation: Depends. In a dry cavity, no hydrogen will form. However, if water is present, the water gas reaction occurs, forming CO and H_2.

Resistance to Jet Penetration: Poor. Although graphite has a high melting point, it reacts with urania forming uranium carbide and dissolves in molten iron.

Temporary Failure of Cooling System: Unknown. Although graphite has a high melting point, it dissolves in molten iron and reacts with urania. Its conductivity is highly anisotropic, along the molecular planes.

Table 9: Comparison of Core Retention Systems for the ZIP Reactors (cont.)

Heavy Metal Bath (Lead, Copper, or Uranium) over Cooling Coils

Cost: Moderate for Lead ($.45/lb) and copper ($1.00/lb) and high for uranium metal ($45.00/lb)

Installation Time: Moderate. Lead, copper, or uranium shot could be poured into place. Cooling coils would have to be installed from prefabricated panels.

Resistance to Floatation: Depends on the bath thickness. A thin layer metal bath, consisting of either lead, copper or uranium, will melt and be able to support only thin layers of hot corium; a thick lump of corium would sink through a thin bath. In order for the bath concept to work, the bath must be relatively thick (J1m).

Vapor Explosion Potential: Lead, High; Copper, Uranium; None. Since the melting point of corium is considerably greater than the boiling point of lead (1740°C), a vapor explosion is possible though probably unlikely.

Pyrophoric Nature: Uranium is pyrophoric and fine particles will burn in air.

Hydrogen Production: No. Neither lead nor copper are expected to produce hydrogen in contact with water. Uranium reduces water readily at temperatures as low as 350°C to form hydrogen and thus the cavity should be kept dry if this metal is used.

Resistance to Jet Penetration: No. A jet of molten corium could easily displace the thin metal bath layer at point of contact and destroy cooling coils. However, once the jet ceases, the molten metal would flow back in place so that the heavy metal bath is self-healing.

Temporary Failure of Cooling System: Poor protection is provided by copper and uranium metal because the heat from the molten core debris will simply be conducted through into the concrete. With lead, the situation is different because of its lower boiling point. As long as any lead remains, most of the heat from the molten core debris will be used to boil lead and the maximum temperature of the bath in contact with concrete is 1740°C.

Table 9: Comparison of Core Retention Systems for the ZIP Reactors (cont.)

Iron Oxide Over Cooling Coils

Cost: Low. The cost of iron oxide is quite low.

Installation Time: High. As iron oxide has a low density and will float in corium, it must be anchored in some manner. A cooling system must also be installed.

Resistance to Floatation: None. The material must be mechanically restrained.

Vapor Explosion Potential: None if the cavity is dry. The boiling point of iron oxide is unknown.

Hydrogen Production: No. It should be noted that water and lower oxides might react to form higher oxides, generating hydrogen.

Resistance to Jet Penetration: Questionable. The degree of resistance will depend on the way that the material is mechanically restrained.

Temporary Failure of Cooling System: Good protection will be be provided by this oxide as it is an insulator.

Table 9: Comparison of Core Retention System for the ZIP Reactors (cont.)

Thoria Sandwich: HAC/Thoria/HAC/Cooling Coils

This concept consists of high alumina cement (HAC) covering a layer of thoria. These layers cover a layer of HAC which in turn covers the cooling coils so that the result is a layer of thoria sandwiched between HAC.

The purpose of the top layer of HAC is to restrain the thoria gravel against both a steam explosion and jet impingement. If the top HAC layer is thin, little water vapor will be generated to produce hydrogen. This sandwich does introduce HAC on top of the thoria where it will presumably form a eutectic with thoria.

Appendix C: Supplement To The Contract - Thermal - Hydraulics of Rubble Beds

By Dr. Ivan Catton

In this appendix a few designs for core catchers are proposed and their characteristics are discussed. The designs are chosen such that they can be implemented in the existing system without excessively taxing the resources or increasing the radiation exposure of the personnel. Though the concepts are very general, the base calculations are limited to the Zion and Indian Point 2 and 3 nuclear power plants.

Flooded Rubble Bed

Before going into details of this concept, it is essential to determine whether or not a debris bed formed after a core meltdown accident will dry out in the situation where a sufficient amount of water exists in the cavity to fully submerge the debris bed. If there exists a possibility that the debris bed could dry out, then one has to devise a concept that will eliminate the debris bed dryout and thereby inhibit the penetration of concrete by the debris.

At the present time a concensus seems to exist between various investigators that a fuel-coolant interaction can occur in light water reactors, either in the vessel or when the molten material falls into water contained in the cavity. However, it is realized that the containment will be able to withstand the maximum pressure pulse generated as a result of the fuel-coolant interaction. If an energetic fuel-coolant interaction occurs, the core debris will be finely divided and spread all over the containment. If the fuel coolant interaction is mild, the debris will be coarse and will most probably be confined to the cavity region. In the present work, the emphasis is focused on the latter situation.

The debris bed heights, as a function of the fractional area occupied by the debris and porosity of the debris, are listed in Table I for Indian Point reactors 2 and 3. The amount of steel in the core debris is arbitrarily taken to be about 50% of the weight of the lower dome of the reactor vessel. For the bed heights listed in Table I, the heat flow at which the debris bed totally submerged in a pool of water will dry out can be limited either by the inability of the liquid to penetrate the porous layer or by the hydrodynamic instability of vapor jets (peak heat flux on a flat plate) at the top surface of the bed. A given particulate bed will dry out by a mechanism which yields the smaller heat flux.

The dimensionless dryout heat flux controlled by resistance of the bed to the downward flowing liquid has been given in reference [1] as

$$\frac{q_{vd}}{q_Z} = \frac{Q_v h(1-\varepsilon)}{q_Z} = 7.5\times10^{-4} K_1 (\bar{d}'_1)^2 \Gamma^{1/2} M \qquad (1)$$

where q_Z is peak heat flux on a flat plate given by Zuber as

$$q_Z = \frac{\pi}{24} \rho_v^{1/2} h_{fg} [\sigma g(\rho_\ell - \rho_v)]^{1/4} \qquad (2)$$

while other dimensionless parameters in equation (1) are defined as

$$K_1 = \frac{\varepsilon^3}{(1-\varepsilon)^2}$$

$$\bar{d}' = \bar{d}\sqrt{\frac{g(\rho_\ell - \rho_v)}{\sigma}}$$

$$\Gamma = \rho_\ell / \rho_v$$

$$M = \left[\frac{\rho_\ell^{\,2} \sigma^3}{\mu_\ell^{\,4} (\rho_\ell - \rho_v) g}\right]^{1/4}$$

Table 1. Core Debris Bed Characteristics

		Zion 1		Indian Point 2		Indian Point 3	
Core Full Power, MW		3.250		3,030		2.760	
Fuel UO_2, lbm		217,000		217,000		216,000	
Zircaloy, lbm		44,500		44,500		42,000	
Steel, lbm		58,000 (half of lower dome)		58,000		58,000	
Total Core Debris		319,500		319,500		316,000	
Debris Bed Depth, ft	Fractional Area of Cavity	With Steel	Without Steel	With Steel	Without Steel	With Steel	Without Steel
ε = 0.4							
	100%	2.7	2.1	3.1	2.4	3.1	2.4
	50%	5.4	4.1	6.1	4.7	6.1	4.7
	150%	1.8	1.4	2.1	1.6	2.1	1.6
ε = 0.6							
	100%	4.1	3.1	4.7	3.5	4.7	3.5
	50%	7.4	6.2	9.3	7.0	9.3	7.0
	150%	2.5	2.1	3.2	2.4	3.2	2.4

For a given bed porosity, the particle size above which the heat flux would be limited by the hydrodynamic instabilities of the vapor jets rather than by the resistance offered to the coolant by the particles themselves can be obtained by setting the ratio $q_{vd}/a_Z = 1$. Substituting the thermophysical properties of saturated water at one atmosphere pressure into equation (1) yields a particle diameter of 1000 μm and 365 μm for bed porosities of 0.4 and 0.6 respectively. Since the particle size can be assumed to be greater than 1000 μm for a mild fuel coolant interaction, the maximum heat flux at the top of the particulate bed will be independent of bed height and will be limited to $q_Z = 3.5\text{x}10^5$ Btu/ft² hr.

Table 2 lists the maximum surface heat fluxes for Zion and Indian Point 2 and 3 when the debris bed is parametrically assumed to occupy 50%, 100% and 150% of the area of the cavity. From Table 2, it can be seen that debris dryout will occur if the area of the cavity occupied by the debris is 100% or less. If the debris is fully submerged in water and spreads out into the keyway, so that the area occupied by the debris is significantly greater than the cavity, the debris bed may not dry out. It should be noted that the heat loss from the lateral surface of the particulate bed has been ignored in calculating the surface heat fluxes. The maximum heat flux on a vertical wall submerged in a pool of saturated liquid is given in reference [2] as

$$q_{max} = 0.9q_Z \tag{3}$$

When the debris bed occupies 100% of the area of the cavity, the lateral surface area of the bed exposed to water will be the height of the particulate bed times the

Table 2. Maximum Heat Fluxes at the Top Surface of Debris Bed for 1% Full Power

Fractional Area of Cavity Occupied by Debris	Maximum Heat Flux, Btu/ft^2 hr Zion	 Indian Point 2	 Indian Point 3
100%	3.17×10^5	3.14×10^5	3.45×10^5
50%	6.34×10^5	6.28×10^5	6.90×10^5
150%	2.11×10^5	2.09×10^5	2.34×10^5

Maximum Heat Flux on a Horizontal Plate = 3.5×10^5 Btu/ft^2 hr width of the keyway. Accounting for the heat loss from this surface results in a reduction of about 8% in the heat flux at the top of the surface. When the debris bed occupies only half of the cavity area, the convection arising from the heat loss from the lateral surface can be as much as 30%. Since the heat flux at the top of the bed is about q_Z for beds occupying 100% of the cavity area, and is $2q_Z$ for beds occupying 50% of the cavity area, no significant change in conclusions occurs even when heat loss from the later surface of the bed is considered. Thus, there is a good possibility of debris bed dryout if the debris occupies an area equal to or less than the area of the cavity. Debris bed dryout will result in heating of the particles and possible penetration of the debris into the basemat of the cavity even if the debris is fully submerged in water.

Next, it is pertinent to determine if debris bed dryout can be prevented by forcing liquid through the debris. This requires information about two phase pressure drop through the debris. At this time, no correlation based on experimental data exists for the pressure drop through a volumetrically heated layer. The experimental work done at UCLA [3] shows that Vasiliev et al.'s [4] second pseudo fluid model predicts pressure drops higher than the observed pressure drops. Thus a conservative estimate of the pressure drop through the porous layer can be made by using the second model proposed in reference [4]. In this model, the kinematic viscosity of the two phase mixture and specific volume of the mixture are defined as

$$v_m = v_v(x) + v_\ell(1-x) \qquad (4)$$

$$v_m = v_v(x) + v_\Pi(1-x) \qquad (5)$$

The two phase pressure drop through the bed is written as

$$\frac{\Delta P}{\Delta Z} = \frac{150(1-\varepsilon)^2}{\varepsilon^3 \bar{d}^2} {}_{m}G + \frac{1.75(1-\varepsilon)}{\varepsilon^3 \bar{d}} G^2 v_m \qquad (6)$$

For a particulate bed configuration such as that shown in Figure 1, with saturated water entering at the bottom and saturated steam leaving at the top, the required mass flux, G, for debris beds occupying 100% and 50% of the cavity area (Indian Point 3) are calculated to be 355 and 710 lbm/ft² hr, respectively. Figure 2 shows the pressure drop as a function of axial height when the particle diameter is taken to be 1000 μm and bed porosity is taken to be 0.4. It is noted that for a debris bed occupying 100% of the area of the cavity (bed height = 6.1 ft), a pressure drop of about 1 psia would occur if all the heat generated in the debris bed was used in vaporizing water to saturated steam. However, if the debris bed occupies only 50% of the area of the cavity (bed height = 3.1 ft), a 5 psia pressure drop would occur through the bed.

The physical model shown in Figure 1 and used in calculating the pressure drop is only possible if a core catcher is used to constrain the debris bed with rigid bounding walls. In the existing designs of the reactor cavity, the possible configuration of the debris bed is shown in Figure 3. For such a configuration, forced flow cooling of the debris bed cannot be established as water will try to bypass the bed (region of high resistance). Thus, for the existing cavity designs, forced cooling is of no value and pool boiling conditions in the debris bed will exist when the debris is fully submerged in water. However, as discussed earlier, the debris bed will dry out

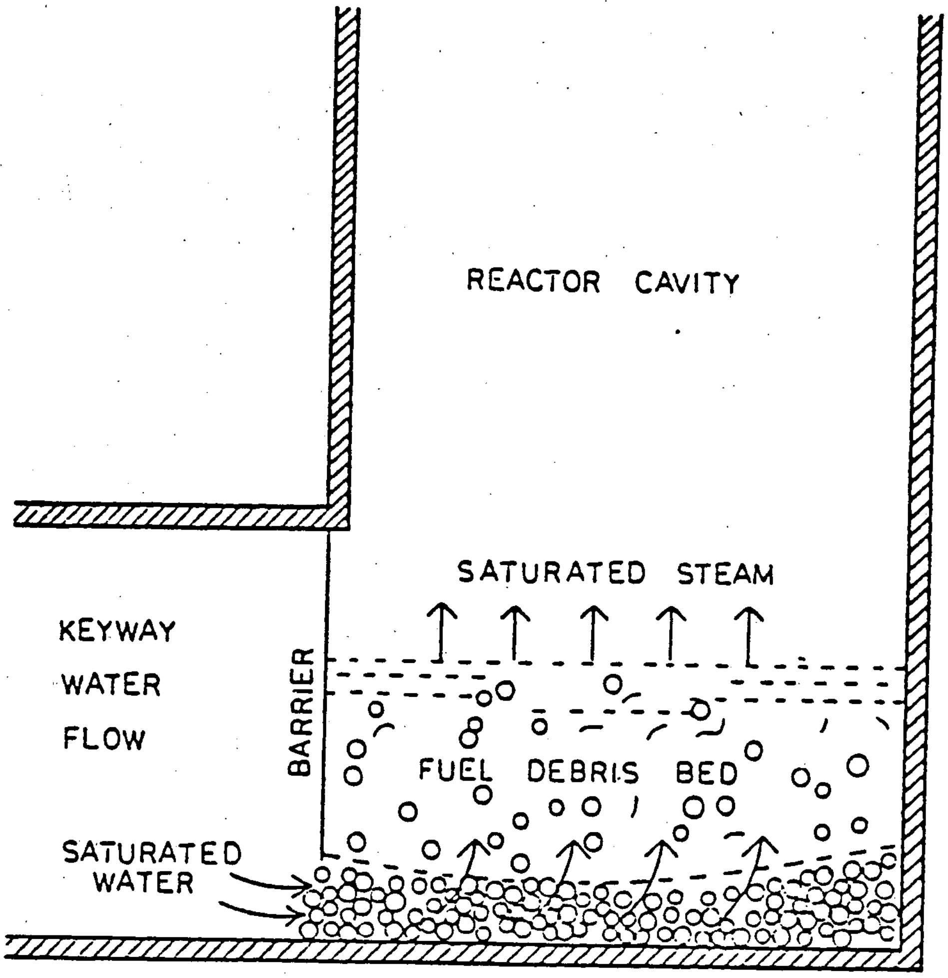


Figure 1. Physical Model Used to Estimate Forced Flow Pressure Drop in Debris Bed

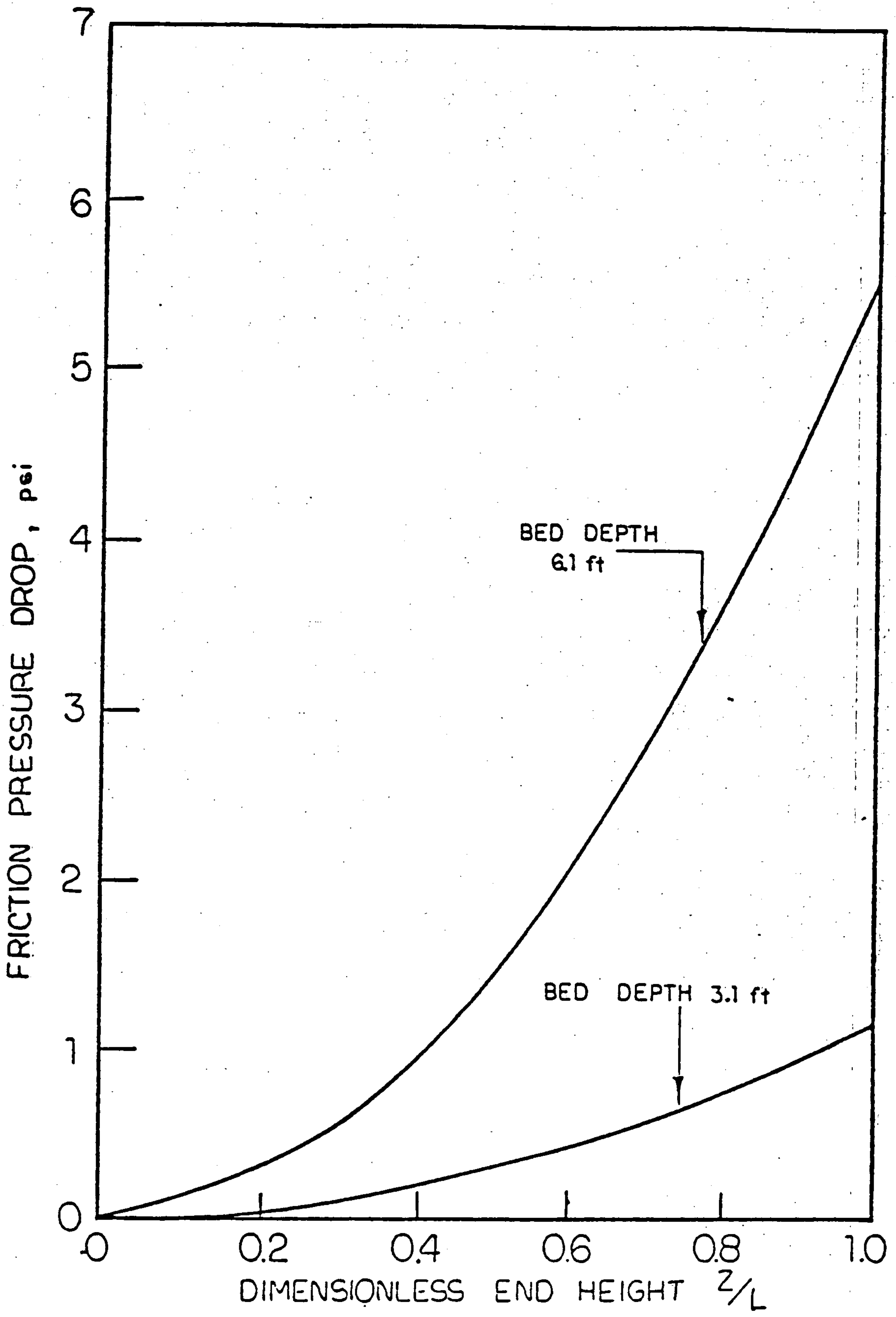


Figure 2. Pressure Drop for Forced Flow Cooling of Debris Bed

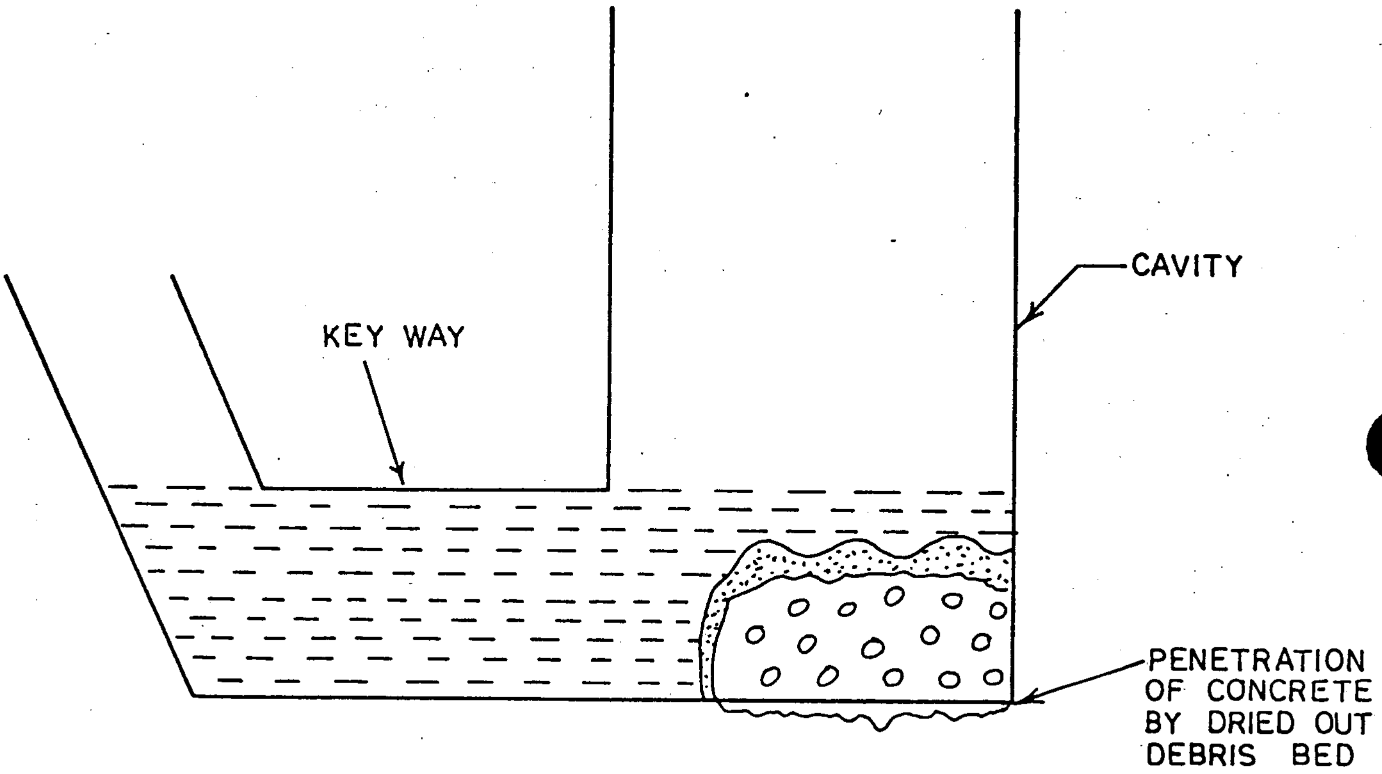


Figure 3. A Possible Configuration of the Debris in the Cavity

if it occupies less than 100% of the cavity area and if the heat generation rate in the debris is equivalent to 1% of the full power of the reactor.

To minimize the possibility of penetration of concrete by the dried out debris bed, it is suggested that panels consisting of square cooling tubes embedded in a 3 inch thick layer of alumina (Al_2O_3) be placed on the floor of the cavity. The panels could then be covered with a 6 inch deep layer of thoria particles with a diameter of approximately 1.5 inches. This arrangement will take only 9 inches of the vertical space in the cavity and will not interfere with the space occupied by instrument tubes. Figure 4 shows the proposed concept and details of the panels and cooling tubes are shown in Figure 5.

Thorium dioxide has been selected as the first layer of sacrificial material contacting the debris bed because of its very high melting point (see Table 3) and stable thermophysical properties. Thorium dioxide will also serve as a good sacrificial material even if the cavity is dry and a molten fuel-steel mixture falls on the cavity floor after the core melt down accident. If the molten fuel penetrates the upper layer, which is improbable if cavity is flooded with water, the alumina panels should be able to arrest further penetration. Aluminum oxide was selected because of its high thermal conductivity and high latent heat of fusion. The alumina panels could be cooled either by natural convection or forced convection. Under convection. Under forced flow cooling, calculations indicate that the upper surface temperature of the alumina panel will be much less than the melting temperature of Al_2O_3 if 30 MW of energy is removed from the panels. Under natural convection conditions only, 0.75 MW of

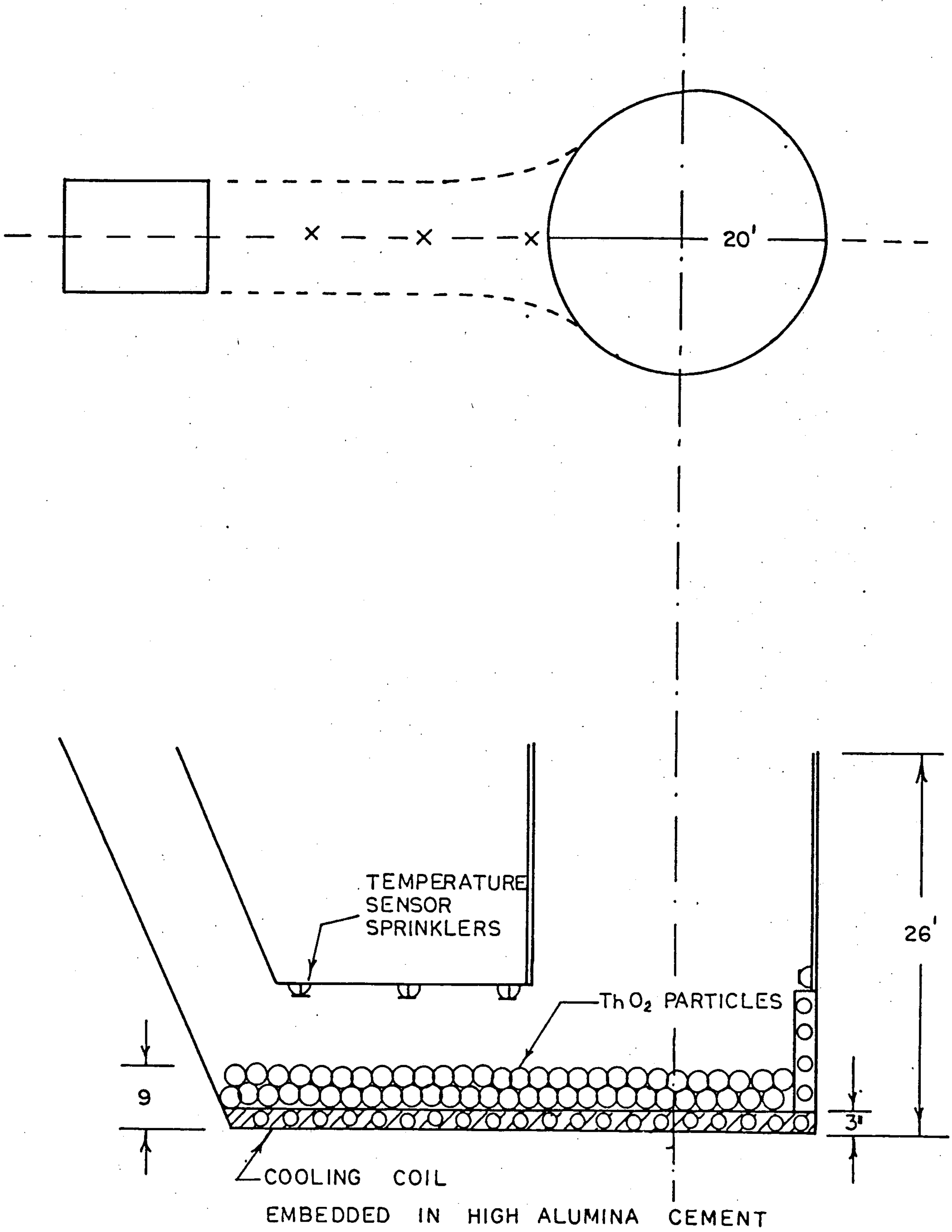


Figure 4. Proposed Concept

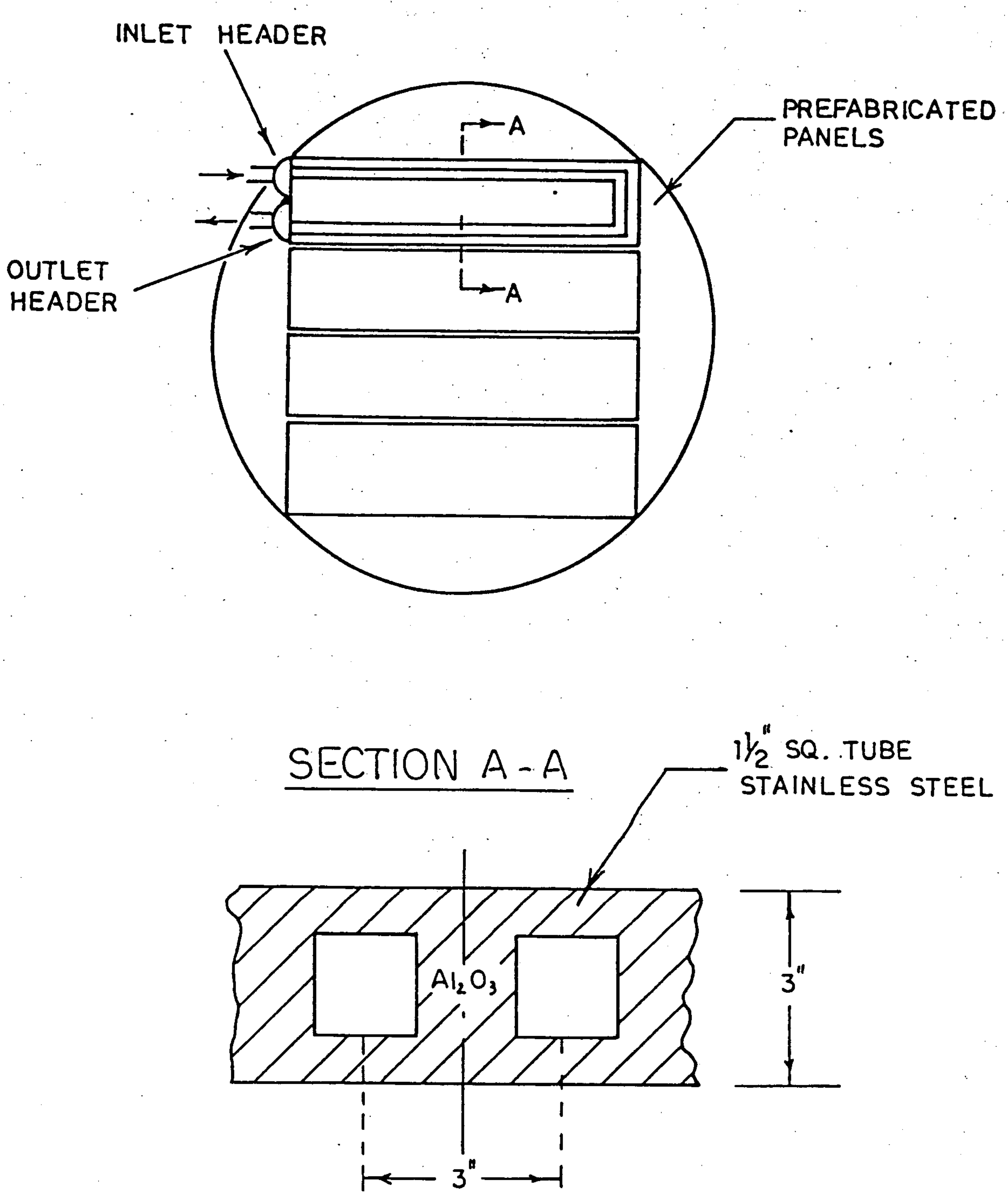


Figure 5. Details of the Panels and Cooling Tubes

Table 3. Heat Absorbing Capacity and Thermophysical Properties of Sacrificial Bed Materials*

Material	Density (lbm/ft^3)	Specific Heat (BTU/lbm-°F)	Thermal Conductivity (BTU/hr-ft-°F)	Melting/Eutectic or Point Lowest Solidus Point (F°)	Heat of Fusion (BTU/lb)	Heat Capacity (Melting/Eutectic) (10^3 BTU/ft^3)
MgO	212	0.32	4.11	5070/4170	826	512/451
Al_2O_3	243	0.30	4.60	3720/3500	500	385/369
ZrO_2	374	0.14	1.33	4850/4700	304	362/354
UO_2	624	0.09	1.69	5160	121	360
ThO_2	580	0.079	1.45	5920/5200	146	351/318
Basalt	162	0.24	0.968	2190	126	102

* Material properties evaluated at about one half their melting point.

energy could be removed. In calculating the maximum heat that can be removed by natural convection, the pipes connecting the inlet and outlet headers to the cooling tower are assumed to be 100 feet long and 9 inches in diameter. The temperature of the water at the inlet and outlet of the cooling tubes is taken to be 160 and 80°F, respectively. The static height of the 80°F water in the cooling tower is taken to be 80 feet, while the height in which water is evaporated is taken to be only 5 feet.

To assure that the cavity remains flooded after core debris or molten core materials fall in the cavity, it is suggested that a temperature sensing sprinkler system be installed in the cavity and the key way. This sprinkler system should be automatically initiated when the temperature of the environs of the cavity reaches approximately 500°F.

REFERENCES

1. V.K. Dhir and I. Catton, Study of Dryout Heat Fluxes in Beds of Inductively Heated Particles, NUREG-0262, 1977.

2. J.H. Lienhard and V.K. Dhir, Extended Hydrodynamic Theory of the Peak and Minimum Pool Boiling Heat Fluxes, NASA CR-2270, 1973.

3. A. Naik, D. Le and V.K. Dhir, Forced Flow Evaporative Cooling of a Volumetrically Heated Porous Layer, ANS Trans., Vol. 34, 1980.

4. L.L. Vasiliev and V.A. Mairov, An Analytical Study of Resistance, Heat Transfer and Skidding in Evaporative Cooling of a Porous Heat Producing Element, Int. J. Heat Mass Transfer, Vol. 22, 1977.

Appendix D: Technical Activity Plans for the Second Contract Year

Technical Activity Plans For The Second Contract Year

A schedule for the program is shown in Figure 12. Details of planned activities for the second contract year are discussed in the following sections.

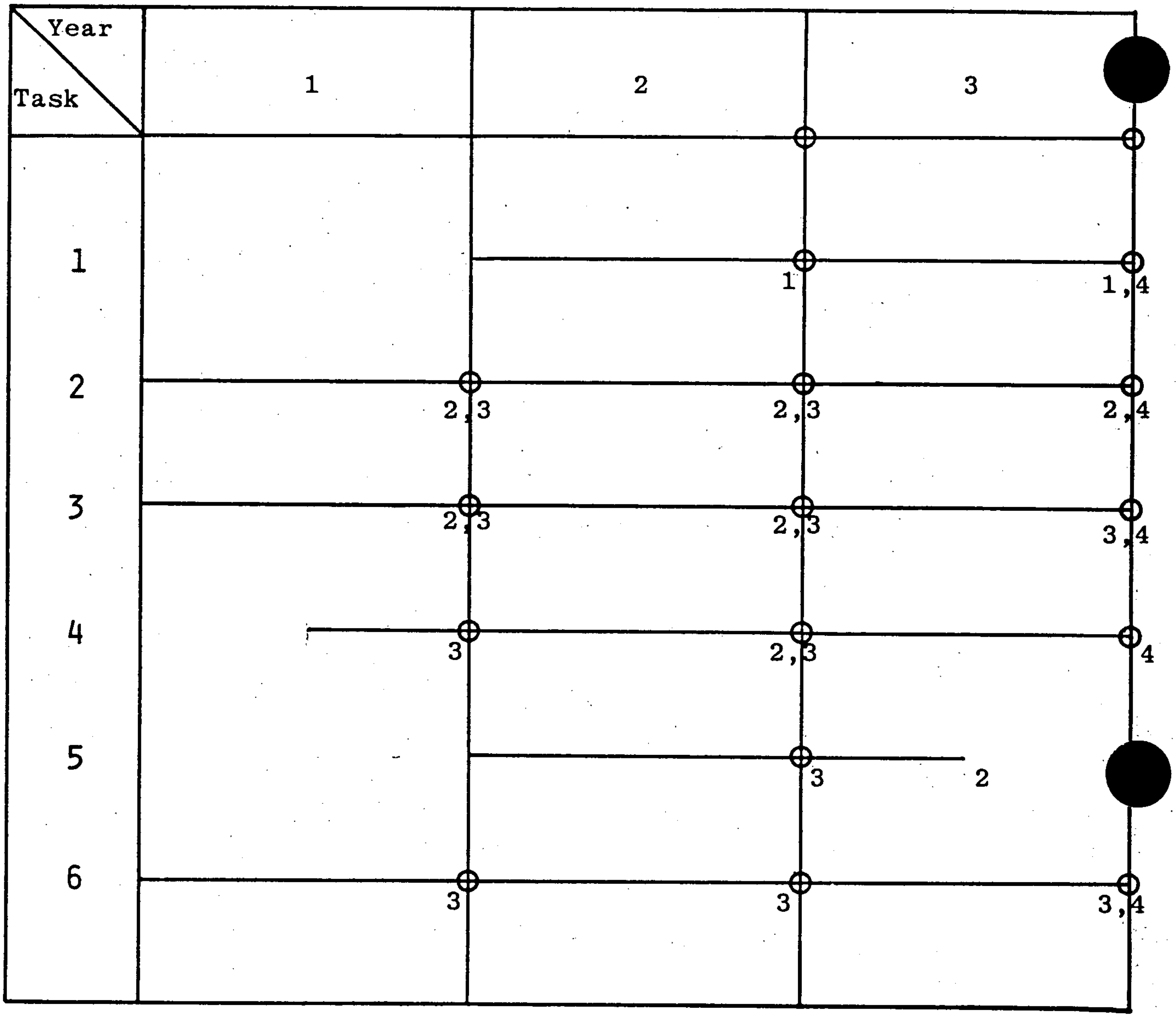
Task I: Technical Assistance on the Floating Nuclear Plant (FNP)

Review will continue of the final Floating Nuclear Plant (FNP) core ladle design (see NUREG-0054, dated February 1980) as new information becomes available. This task will necessitate an examination of phase diagram data, other experimental data and relevant industrial experience. Particular attention will have to be given to interactions between iron oxide slag and MgO. The potential for slag formation and probable extent of slag formation will be examined. Factors that may serve to mitigate the low melting eutectic between the iron oxide slag and MgO will be studied. The work will include an examination of the effect of the basicity of the brick and impurities on the expected interactions with the constituents of molten core debris, particularly iron oxide.

The test program at Sandia utilizing the large scale melt facility will be closely watched. In this facility, a large MgO crucible will be subjected to a corium melt which will be sustained in a molten condition for a time after test initiation. This experiment is planned for this summer.

Industrial experience is relevant and will be examined because it constitutes the only available body of technical information on the interactions between molten

Schedule



1. Review and evaluation of submitted technical reports (three over a three year period).
2. Report on task evaluation results
3. Annual Report on results of evaluations and status of experimental programs.
4. Final report.

Figure: Program Schedule

materials and large scale refractory structures like the FNP core ladle. Areas where industrial experience will be particularly useful include consideration of the mechanical and thermal shock resistance of the FNP ladle. Other areas include the extent of cracking and whether the proposed design is adequate to prevent floatation.

Task II: Evaluation of Candidate Core Retention System Materials

It has become evident that one of NRC's most pressing current needs is an examination of the feasibility and cost effectiveness of retrofitting the ZIP plants with a core retention device. Task II of the contract calls for an "evaluation of candidate core retention system materials". Consequently, our initial efforts under Task II focused on the feasibility of retrofitting the ZIP reactors. It is expected that a substantial effort in the area of conceptual designs will continue during the second contract year. This effort will be directed in support of hearings at Indian Point starting during the summer of 1981 and the Degraded Core Rulemaking activities which are scheduled later this year.

We will be concerned primarily with materials related issues in any proposed core retention system. The material selected should have a high melting point, high boiling point, high specific heat and heat of fusion. It should also have a high density, a low thermal conductivity and should be chemically compatible with the molten core debris. It should not generate gas on contact with the molten core debris and it must resist thermal shock. The material also must be stable in the expected warm, wet environment of the reactor cavity. Our initial examination indicates that these considerations will restrict the range of possible choices to only a few materials.

The most promising concept that has been examined is a layer of refractory gravel flooded with water. It is possible to add a core retention system to an existing plant by pouring a suitable high density refractory such as thoria, into the reactor cavity. This is a desirable option because it will result in minimal radiation exposure to personnel. The details of this concept have been examined at length elsewhere in this annual report. Further work on the flooded thoria refractory gravel bed is anticipated, particularly in establishing its cost/benefit ratio and competing risks.

Core retention concepts will be studied further. This work will require that we confer with NRC consultants at Brookhaven and Sandia. There will be consideration and discussion of the design details of the proposed rubble bed core retention concept. This investigation will include a consideration of what can reasonably be done in the limited space available in these plants in view of the existing high radiation environment.

The results of our work are being integrated with the results of heat transfer calculations at Brookhaven and other work at Sandia. It is anticipated that a substantial coordination effort will be required to integrate these results and develop a recommended course of action.

At some point, it may be desirable to visit the Zion and Indian Point plants. A detailed examination of the reactor cavity through personal inspection is highly desirable in order to avoid unexpected surprises, such as plant features not indicated on the drawings.

For containment types other than the large dry cavities, other concepts may be more attractive than the flooded rubble bed proposed here for the ZIP plants. A study similar to that performed for the ZIP plants is planned to investigate the feasibility and cost effectiveness of core retention devices for other existing containment types (BWR Mark I, II, and III, Ice Condenser, Sub-Atmospheric, etc.), including an examination of the available space in the lower part of the reactor cavity.

Task III: Core Melt Interactions with Concrete

Before the interactions between core debris and concrete can be ascertained, it is necessary to investigate the influence of water on the core debris. There is a controversy regarding the potential for the formation of a coolable debris bed when molten core debris and water come into contact. If a coolable bed forms, the concrete will not be attacked significantly by the core debris.

Additional issues include the strength of the crust that forms over core debris and the variation in crust strength with the quantity of core debris (scale effects). It is expected that the nature of the crust will change dramatically as concrete melts and enters the molten pool. An investigation of the nature and strength of this crust will be conducted based on available information. Proposals by Sandia National Laboratories for experiments in this area will be followed closely and tests will be suggested to support NRR safety evaluations.

Other factors affecting the expected extent of molten core penetration into concrete are the melting of constituents, chemical attack, formation of low-melting temperature eutectics, cracking, spallation, and gas evolution. The importance of the concrete constituents in the behavior of concrete will also be assessed.

With the substantial difference between the expected temperature of the molten core debris and the melting temperature of concrete, it is probable that concrete penetration will be dominated by the melting of the concrete. When molten core debris comes into contact with the heterogeneous concrete, the low-melting constituents are expected to melt preferentially. The formation of low melting eutectic mixtures will enhance penetration.

Chemical attack has not been observed in molten core debris-concrete interaction experiments and does not appear to be a significant contributor to the penetration rate, based on information available at present. Crack formation can provide a path for penetration of molten core debris into the concrete and thus will enhance the penetration rate by the removal of large chucks of concrete from the melt-concrete interface. The spalled chunks could float away into the molten core debris, exposing fresh concrete to attack.

The assessment under this task will include a review of the available literature on the interactions between core debris and concrete and a review of various experimental programs sponsored by DOE and NRC. The heterogeneous nature of concrete makes it necessary to carefully consider the behavior of each of the concrete constituents in a molten core debris environment. Consequently, the scientific literature and experimental information available on the interactions between the constituents of concrete and molten core debris will also be reviewed.

The assessment of gas evolution must consider several effects. On one hand, gas evolution signifies the decomposition of concrete constituents which should increase the rate of penetration. However, gas evolution also reduces contact between the molten pool and concrete and therefore reduces the rate of heat transfer into the concrete. The gas generated may also react with the molten core debris. The extent of gas formation is important because of the effect of the increased gas pressure on the containment and because of the potential for combustion.

The extent of non-condensible gas formation will be evaluated by a consideration of potential chemical reactions. The expected quantities of gas will be calculated as a function of the amount of concrete that reacts with the molten core debris. For a given quantity of concrete involved in reactions, the amount of gas that forms can be calculated from the chemical stoichiometry. This information, in turn, can be used in the gas law relations to calculate the pressure exerted by the gas.

Task IV: Generation of Hydrogen in LWR Systems

If water is initially present in the reactor cavity, hydrogen will be generated if hot core debris falls into the water. The existance of a temperature threshold for hydrogen generation from metal-water reactions has been noted elsewhere in this report and hydrogen will be generated until the core debris cools sufficiently. The threshold is a function of the metal constituents. Water driven from concrete by the hot core debris can also interact with the core debris to generate hydrogen.

On the other hand, if the cavity is initially dry when core debris comes into contact with concrete, the heat transferred into the concrete from the hot material will drive water from the concrete, causing dehydration. This water, in the form of steam, will generate hydrogen from a number of chemical reactions when it comes into contact with reactor materials at high temperatures. If water is added subsequently, hydrogen will be generated when water comes into contact with metal above the threshold temperature.

Another issue in a very complicated problem is the influence of crust formation. When a crust forms, it will reduce hydrogen production from water reactions because it will be cooler than the material inside the crust. Also, the crust will reduce contact between the water and metal. Issues include the strength of the crust, whether it breaks open readily, whether water can penetrate the crust, and interactions between the crust and water. The situation is further complicated by expected changes in crust composition as core debris-concrete interactions proceed.

The quantity of steam and other gases, such as CO_2, which will evolve when molten core debris comes into contact with concrete, will be dependent on the composition of the concrete used in the reactor and the environmental history of the reactor cavity (wet concrete for example, contains more water than dry concrete). Other factors include the composition, quantity, and temperature of the molten core debris.

The effect of the proposed rubble bed on hydrogen generation must also be considered. With a dry rubble bed, hydrogen can be generated from core debris contact with concrete if the core debris is molten and trickles through the rubble bed. With a flooded rubble bed, some hydrogen may evolve from contact between the core debris and water but this evolution will be reduced by crust formation and water cooling of the crust surface. If the water level is kept below the top of the bed initially, the probability of a steam explosion can be reduced.

The extent hydrogen generation will be calculcated from the known chemical reactions. Calculations will be performed for specified penetration depths on the assumption of a complete reaction. It will be assumed that chemical kinetics is not a limiting factor. From the stoichiometry, with a postulated penetration rate, the extent of hydrogen generation will be calculated using the gas laws.

The hydrogen program at Sandia will be closely followed. Experiments to support NRR safety evaluations will be suggested where appropriate.

Task V: Modeling Studies

Materials interaction information has been and will continue to be provided to the Technical Assistance Program at Brookhaven National Laboratory. This work is concerned with modeling post-accident containment analyses following postulated core meltdown events in light water reactors. Calculations will be performed to establish correlations for critical materials related parameters influencing the extent of (1) molten core debris interactions with either concrete or refractory sacrificial materials and (2) formation of non-condensible gases and combustible gases. The work will include modeling of phenomena related to hydrogen generation. These calculations will be based on experimental and other materials information.

Task VI: Interface with DOE and NRC/RES

The information derived from the RES Materials Interaction Confirmatory Research program at Sandia Laboratories will be reviewed. Tests, to be performed by RES contractors, will be recommended to support NRR safety evaluations where appropriate. Assistance will be provided to NRR in factoring these research results into the safety and environmental reviews associated with certain light water reactor licensing actions, such as the severe accident mitigation feature study for the Zion and Indian Point plants, the licensing of Floating Nuclear Plants, and the follow-up response to the TMI-2 accident. A close interaction will be maintained with DOE contractors in order to maintain awareness of programs and to provide technical guidance regarding the resolution of materials interaction safety concerns to NRR. This interaction will include work done by light-water reactor research groups and evaluation of European work where appropriate.

Appendix E: Trip Report: Foreign Travel

Report of Foreign Travel

by

David G. Swanson
Applied Science Associates, Inc.
Hawthorne, CA 90250

and

Andrew R. Marchese
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

Project Nuclear Safety Colloquium
US/FRG Core Melt Information Exchange
and
Visits to Laboratories in Germany,
France, and Italy

November 25 - December 8, 1980

Forward to Trip Report

In the following trip report, emphasis will be given to a presentation of the views of foreign engineers and scientists. Papers presented by Americans will not be discussed as it is assumed that American work either has been or will be adequately disseminated in this country.

I. Projekt Nukleare Sicherheit (PNS) 1980 Annual Meeting, Karlsruhe, Germany

The 1980 PNS annual meeting was held at Kernforschungszentrum Karlsruhe (KfK) in Karlsruhe, Germany on November 25, 1980. Eight papers were presented to a small contingent of Americans and a large number of German nuclear scientists and engineers. An agenda for the meeting appears in Attachment I. The authors of this trip report have copies of these papers available if desired. Highlights from some of these papers are presented in the following sections.

A. Human Failure: A Basic Problem of Nuclear Power Plants - Dr. Schmidt, KfK

Dr. Schmidt feels that there would be a period of 1-2 hours after an accident occurs before the core will be uncovered and a melt sequence is initiated. Consequently, it is believed that there will be sufficient time available to react in an appropriate way to mitigate the consequences.

Dr. Smidt indicated that he does not believe that human failures alone were responsible for the Three Mile Island (TMI) accident but rather that technical and design errors also contributed to the accident. It is necessary to take account of the possibility of human errors even though there are automatic systems. He indicated that there was a need for better instrumentation, particularly to indicate the water pressure and temperature in the core, the water level, and the availability of high pressure water. He felt that there was a need to employ computer systems to filter information in order to avoid confusing the operators so that they can concentrate on the proper signals and act accordingly.

He also felt that there was a need for better staff selections. In particular, there should be engineers in the control rooms of the plants, who are acting as supervisors, rather than the current situation where the supervisors are foreman without engineering degrees. He felt that further training was needed with different accident hypotheses using simulators, both for engineers and managers.

He concluded by observing that the man-machine interface and psychological considerations must be considered by those concerned with nuclear safety.

B. Recent Advances of Reactor Safety Research in the Nuclear Safety Project - H. Rinisland, KfK

Dissolution of iodine and cesium in water could reduce the expected dose by a factor of three following a core melt accident. The iodine and cesium combine forming cesium iodide. The distribution of cesium and iodine in the aqueous and dispersed phase is very important and will be a high priority item for study at the KfK.

Overpressure failure of the containment is predicted to occur in about 1-3 days for various core melt scenarios. The time of failure of the containment is important in terms of the amount of radioactivity released and evacuation requirements. The airborne aerosol concentration will continue to be high after one day. Consequently, if the containment fails after one day, there will be significant releases. If the containment fails after some days, there will be a significant reduction in the releases of radioactivity due to plating out of the aerosol.

C. Analysis of the Course of Hypothetical Core Meltdown Accidents - K. Hassmann, KWU and M. Reimann, KfK

The times calculated for the occurrence of significant events in a core meltdown sequence were discussed. About 1.75 hours after the initiation of a transient, 50 percent of the core will be molten. After 2.5 - 3.0 hours, the molten core will drop to the lower reactor head. Reactor pressure vessel failure is expected to occur at about 3.0 - 3.5 hrs. After 4.0 - 4.5 hours, core-concrete interactions begin. These results are similar to recent results from calculations in the United States for a TMLB' sequence for the Zion/Indian Point (ZIP) plants.

Significant attention will be devoted in the future to the 4th phase, where the core debris attacks the concrete basemat. The Germans have conducted experiments to study corium-concrete interactions with water flooding. The studies showed that corium crust formation in water will occur which separates the melt from the water. A void under the crust, between the crust and molten material, has also been observed.

In order to examine these interactions in greater detail, a large scale experimental facility, the BETA facility, is under construction and will be completed in 1982. With this facility, it will be possible to thermitically heat 600 kg of oxide while inductively heating the metallic layer.

Calculations indicate that the first few hours of the 4th phase core-concrete interactions are very important. During the first hour, there is significant gas generation. After three hours from the beginning of core-concrete interactions, the metal phase (iron) freezes at a temperature of about 1700° K. It is predicted that core debris will penetrate the 6m concrete basemat of a standard German PWR in about 14 days. An 8.5 bar overpressure, which is sufficient to burst the containment, will be generated after 4 days due to core-concrete interactions. The design pressure for the containment is 6.3 bars gauge.

Mr. Marchese questioned the Germans with regard to the studies that they are conducting on the influence of water in alleviating the penetration of core debris into concrete. He inquired about the extent to which the Germans had investigated this experimentally and analytically. They replied that, based on a few preliminary small-scale experiments, water will not prevent the core debris from penetrating concrete and generating gases. With a small amount of water present, film boiling occurs and prevents water from cooling the corium. With more water, a porous crust forms over the top of the melt. The crust is separated from the melt by a void with the result that water has virtually no effect on mitigating the melt attack on concrete. Since the crust is porous, any gas generated can readily penetrate it.

The Germans indicated a concern over whether or not the U.S. is conducting core melt studies of severe accident mitigation features for boiling water reactors (BWRs) in addition to those for pressurized water reactors. The response was that little has been done but that more attention will be paid to BWRs after the ZIP studies are completed.

II. US/FRG Core Melt Information Exchange Meeting - Karlsruhe, Germany

This meeting was held at Kernforschungszentrum Karlsruhe in Karlsruhe, Germany on November 26-27, 1980. An agenda appears in Attachment II. Mr. Sherry of the NRC's Office of Research, who acted as one of the chairman of the meeting, has provided a trip report (including all handouts), dated January 8, 1981, on this meeting. Therefore, the authors will only highlight those points that are of interest to the licensing staff.

A. On-Going Activities in FRG Core Melt Research - Dr. D. Lummerzheim, German Ministry for Research and Technology (BMFT)

Dr. Lummerzheim's comments will be discussed in detail under the sections covering our meetings with him.

BMFT is conducting risk analyses and studying the consequences of a core melt accident. The accident of TMI, emphasized the need to consider hypothetical core melt accidents and such accidents will be considered in future

FRG licensing actions. It is still an open question in their view as to whether it is better to (1) install features to mitigate the consequences of a core melt accident, or (2) improve the methods for preventing such accidents.

B. Steam Explosions: Theoretical and Experimental Investigations In Steam Explosions, Overview - Dr. Kottowski, Euratom (Ispra)

The processes promoting large vapor explosions are (1) large coarse mixing zones; (2) the absence of self-triggering explosions; and (3) a pressure trigger, which must be large enough to condense the vapor. On the other hand, vapor explosions are impaired by (1) freezing of particles during forced mixing; (2) non-condensible gas formation during the course of the explosion; (3) local entrapment during mixing; and (4) local quenching by the coolant.

The conversion efficiency of thermal energy to mechanical work is dependent on the mass of melt. With large melts, the mixing is less efficient and, consequently, the conversion efficiency is lower. For droplets of melt, the conversion efficiency is around 10%, but for 20 kg quantities, the rate drops to 1%.

The explosion generated by ten tons of melt releasing all of its heat and converting this heat to mechanical energy would be sufficient to fail the reactor vessel. However, a large scale vapor explosion of this magnitude is regarded today as much less probable than a few years ago. As the melt mass increases, the conversion efficiency is expected to decrease. This is different from the Sandia results where no mass dependence has been observed. In addition, at Ispra, explosions have been observed at high pressures without triggering. This is also contrary to the results obtained in experiments at Sandia.

The general consensus at the meeting was that the probability of a steam explosion failing containment is much less likely than assumed in WASH-1400 (by a factor of 100).

Even though steam explosions breaching containment are less likely, the core debris characterization is becoming more and more important. Issues include the effect of fragment size distribution and the extent of debris dispersal following a steam explosion on the subsequent accident evolution.

After several American presentations, there was some general discussion on steam explosions. Although steam explosions can occur, it was felt that it was unlikely that large melt masses would participate. Reactor pressure vessel failure may occur if large melt masses participate, but no large missiles will be generated that will fail the containment. The missiles that might form are generally expected to be small and will not penetrate the containment.

C. Analysis of MARCH-KESS Comparison Calculations - Dr. Hassman, Kraftwerk Union (KWU)

This paper began with a comparison of the features of the German KESS code and the American MARCH code, both of which are intended to calculate the transient conditions in the containment building following various core melt sequences. It was concluded that there was reasonable agreement between the two codes. Both codes predict the same pressure-time history with appreciable differences in the peak pressures for large LOCA accidents.

D. Melt-Concrete Interactions - Status of the BETA Experiments - Dr. Hosemann, KfK

Dr. Hosemann began by indicating that the BETA facility was intended for the study of core melt accidents, especially core-concrete interactions. The work is directed towards verifying the results of risk studies, including computer codes. The phenomena that will be studied in the BETA facility include the behavior of concrete under thermal load, heat transfer, gas release, aerosol formation, crust formation, solidification, etc.

Induction heating will be used to simulate the decay heat flux. All phases of the core-melt interaction cannot be simulated in a single experiment; multiple experiments are required. A steel and aluminum oxide mixture (300 kg of each) will be used as the corium simulant. The pool diameter will be 40 cm with a maximum net input power of 1.8 megawatts (8.7 MW total input). The induction system operates at a frequency of 1000 cycles/sec and couples to the steel only. The concrete crucible inner diameter is 40 cm. The crucible is 2-3 m high and weighs 5-10 tons. The melt can be maintained at 2000°C for 7 min. and at a temperature around the melting point of steel for 30 min. Very extensive instrumentation is planned in the concrete, the melt, and for gas and aerosol measurements.

The facility will be completed early in 1982 with initial experiments planned for the summer of that year. Experiments will be conducted with a frequency of about one per month. A total of 12 are planned thus far. The long term behavior of solid core debris penetrating concrete will be studied in this facility.

Following these presentations, there was some general discussion about whether core debris could penetrate through the entire concrete basemat. The general consensus was that it was not possible to state this definitely without further experimental information on the long term behavior of solid core debris attacking concrete. Even though present calculations indicate that core debris will penetrate through the basemat, the models break down for times significantly greater than a few hours.

E. WECHSL and KAVERN - Dr. Reimann, KfK

Dr. Reimann compared the German WECHSL and KAVERN computer codes for prediciing the penetration of molten core debris into concrete. Both have essentially the same physics. Calculations were described for siliceous concrete in which erosion was mainly in the downward direction in the initial phases of the corium-concrete interaction. After 4 hrs, a solid steel layer forms and continues to penetrate the concrete. Since gases must bypass the steel when it becomes solid, there is less cooling and the oxide layer begins to heat up again. This situation, following solidification, is not modeled by the codes. With Clinch River Breeder Reactor (CRBR) type concrete, containing very high proportions of limestone, the steel solidifies later in time due to the refractory nature of the concrete which maintains the melt at higher temperatures. A single German code for core melt-concrete interactions will be developed and incorporated into the KESS code for calculating the overall core melt sequence of events. They also plan to investigate further and improve models for sump water ingression on the melt.

F. EPRI Core Melt Activities - Dr. Ritzmann, Electric Power Research Institute (EPRI)

EPRI plans to conduct large scale testing directed at hydrogen combustion and control. A 75000 ft^3 spherical vessel with a 75 psig capability has been acquired for this purpose. Planned experiments include an examination of combustion in various mixtures of hydrogen, air and steam. They will also study controlled burning, flame suppressent techniques, and water sprays. Other work will examine steam spikes, corium-water interactions, debris bed coolability and dryout. A request for proposals has been issued for studies of containment performance during severe postulated accidents. EPRI expects to have major results within one year in order to influence degraded core rulemaking. It plans to spend \$3-4 million in the first year and \$20 million over 5 years.

G. Status of SASCHA Experiments - Dr. Albrecht, KfK

Dr. Albrecht is doing excellent work on fission product release under degraded core conditions. This work should be closely followed by NRC.

In the SASCHA experiments, a fuel pin specimen is inductively heated in a thoria crucible, using a high frequency power supply. The fission product gases released during heatup are separated by a system of filters and the radioactivity spectra are analyzed by Ga-Li detectors. Representative fission products which can be measured include I, Cs, Te, Sb, Ag, Ba, Mo, Ru, Zr, Ce, Nd, Cr, Mn, Fe, Ca, and Np.

Two important parameters are temperature and time; releases at low temperature and long times are equivalent to releases at high temperatures and short times. With a temperature increase of 400°C, the release rate increases by two orders of magnitude. Releases under higher system pressures are greater than under lower pressures, perhaps due to oxygen reactions. In steam, no pressure dependence was observed.

Fission product release rate data was developed for 17 elements which allows calculations of mass and activity releases for any temperature-time history of a reactor core. Experiments with steam have been initiated which have thus far shown that I and Cs releases in steam are greater than in air. Release of Te, Ag, Sb, and Mo were found to be considerably higher than assumed in WASH-1400. Dr. Albrecht also plans to study fission product release during the 4th phase of core melt-concrete interactions.

H. Status of NAUA Code Analysis and Development - Dr. Schock, KfK

NAUA is an aerosol behavior code that computes fission product removal in an LWR containment during a core meltdown accident. The model will be experimentally validated. There are preliminary indications that the airborne aerosol concentrations after one day will be very high. Consequently, if the containment fails at about one day, there will be a large release. If it fails after 3 days, there will be no increase in the release. This indicates that any mitigation features have to be activated within one day if they are to be effective.

III. Meeting with Dr. M. Peehs at Kraftwerk Union (KWU), Erlangen, Germany

On November 28, 1980, we met with Dr. Peehs, Dr. Hassman, Mr. Czech, and Mr. Beyer at the laboratories of Kraftwerk Union (KWU), in Erlangen, Germany. The meeting was divided into 3 parts: (1) discussion of core melt phenomena, (2) severe accident mitigation features, and (3) decay heat removal systems.

A. Core Melt Phenomena

A large number of very interesting experiments were discussed during the meeting. Two concrete experiments were described using a plasma flux and differential thermal analysis on basaltic concrete. The gravel and sand used in the concrete in Germany are nearly pure SiO_2. The concrete constituents are essentially free of limestone. Water losses were measured and CO_2 did not evolve from the concrete, as expected. No spallation was observed when the plasma torch heated the concrete. Limestone concrete was also examined. The Germans felt that the melting point of limestone concrete would be no higher than basalt concrete due to the multi-component nature of the system.

Experiments are planned to examine the effect of molten core debris on prestressed concrete crucibles. Devices will load the crucibles during the experiments in order to study the erosion of stressed areas for comparison with unstressed regions. Manufacturing of the experimental devices is now nearly finished.

With regard to effects of scale in corium-concrete experiments, several parameters are important. The crucible must be several times the maximum gravel size (35 mm). Since the thermal conductivity of concrete is poor, any specimen that satisfies the gravel criterion will also be adequate from a thermal conductivity viewpoint. Experiments have been conducted with small scale specimens and the results have been compared with results from large scale specimens. It has been found that experiments with 100 kg of molten corium give the same results as 1 kg tests.

Experiments have also been conducted to examine the differences in the interactions between corium with dry and water saturated soil and clay. It has been found that a leak resistant, protective crust, similar to a glass, forms ahead of the melt. At present, it appears that leaching of radionuclides is not a problem due to the crust formation. A report will be released during 1981 on this work.

In one series of experiments, the effects of water on the melt attack on concrete were examined. It was found that no energetic events occurred. Varying amounts of a nickel melt (1-5 kg) at 1500-1600°C were poured on 10-20 cm diameter concrete crucibles. The crucibles were then flooded with from 50 cc - 10 liters of water. When water entered the crucible, a crust formed immediately. The crust was porous allowing gas from the concrete decomposition to escape. At the end of some of the experiments, the crust was broken with a tube of Al_2O_3 without producing any fuel-coolant interaction. With a very low water pouring rate, a slight fragmentation of the crust occurred. At higher water pouring rates, no energetic reactions or significant fragmentation were observed. A void formed between the crust and melt, and the melt continued to penetrate the concrete. The German researchers believe that the crust/void formation essentially insulates the melt from the overlying water pool, thereby allowing the melt to continue to penetrate concrete.

Other work at the Technical University of Hannover tends to confirm the absence of fragmentation. In these experiments, non-condensible gases were blown through molten tin and antimony heated in a crucible containing holes. With molten material at 300-500°C, little fragmentation was observed when either gas or water were blown through the metals.

Work is beginning on an examination of the long term behavior of solid core debris penetrating concrete. Some results may be available by July 1981.

Dr. Riemann and Peehs disagree with the disintegration enthalpies found by Sandia Laboratories for limestone concrete. They believe that the Sandia values are too high. There is no disagreement with the Sandia results for basalt concrete.

B. Severe Accident Mitigation Features

Consideration is being given to a mitigation feature for core melt accidents that would involve spraying water on the outside of the containment steel shell. The sprays will be located in the annulus between the outer concrete confinement building and the inner steel shell. This spray system is still in a preliminary stage. It is felt that the water spray system would provide enough heat transfer to condense the steam in the containment and limit the pressure so that overpressure failure can be avoided. However, the hydrogen concentration will increase correspondingly when the steam concentration is reduced by condensation. The increased hydrogen concentration must also be considered.

The Germans discussed the design criteria and requirements that they envision for the spray system. There was considerable agreement between their criteria and what we are considering for severe accident mitigation features for the ZIP plants. The only significant difference was in the area of seismic design criteria. The Germans feel that severe accident mitigation features should not be designed for any seismic loads, as contrasted with our approach to design for operating basis earthquakes.

Drawings of the KWU spherical containment and safety equipment locations are shown in Attachment III. The failure pressure for this containment is estimated to be 8.5 bars gauge, and the containment was designed for 6.3 bars gauge. Note that the concrete basemat thickness, as shown in Attachment III is about 6 m.

C. Decay Heat Removal Systems

We discussed with KWU the extent to which their systems (for their standard four-loop PWR design) for removing decay heat and for maintaining reactor coolant inventory differ from U.S. reactor designs. Generally, the Germans have specified more stringent design requirements for decay heat removal systems in an effort to cope with a greater range of emergency conditions, such as sabotage, airplane crash, and explosive pressure waves. Compared to the U.S. designs for decay heat removal systems, the Germans provide more redundancy, increased structural protection, and special emergency decay heat removal systems.

Deviations from common U.S. practice for decay heat removal exist and are discussed below for the auxiliary feedwater system, the residual heat removal system, and the chemical and volume control system. The reader is referred to Attachment III which provides schematics for these systems.

With respect to the auxiliary feedwater system, the system is modified to include four separate trains which are protected by a bunkered structure. Within thirty-five seconds following a reactor shutdown, the steam generator relief valves close and the bunkered system forms a closed loop. Each train consists of a pump, a demineralized water pool, and an emergency condenser. The emergency condensers are cooled by an essential service water system.

In addition to the four train auxiliary feedwater system, there is a special emergency feedwater system, consisting of two trains which are also bunkered. This system has one pump per train; however, the trains share a common condensate storage vessel. The special emergency feedwater system can operate automatically for ten hours.

A bunkered, normal residual heat removal system is provided consisting of four trains, each with a pump and heat exchanger. Each train of the four train component cooling water system has two pumps in parallel and one heat exchanger. The essential service water system has four trains, with one pump each and an ultimate heat sink.

A special emergency residual heat removal system is also provided, but it uses some components from the normal residual heat removal system. Two residual heat removal system heat exchangers are used along with two special emergency residual heat removal system pumps to form two trains. Two trains, with one pump and heat exchanger per train, of the component cooling water system are used. The special emergency service water system consists of two trains, each with two pumps in parallel and one ultimate heat sink. These special bunkered systems operate automatically for ten hours.

In addition to the normal chemical and volume control system, a bunkered four train special emergency borating system, consisting of a pump and a tank per train, is provided. This system operates automatically for ten hours.

It is worth noting that the German criteria require maintaining decay heat removal despite a component failure in combination with a maintenance outage, whereas U.S. criteria require sustaining only a single active failure.

We also discussed briefly the German plans for installing high point vents and level detectors. They have made a decision to install high point vents on the reactor vessel. They are examing various means for detecting the level of water in the reactor vessel and have not decided which method to employ. The level detector will probably be coupled to the high point vents.

IV. Meeting at the Ministry for Research and Technology (BMFT), Bonn, Germany

On December 1, 1980, we met with Dr. D. Lummerzheim and Dr. K.H. Krewer of the German Ministry for Research and Technology (BMFT), Bonn, Germany. This ministry is responsible for funding research and development in nuclear technology. It also responds to requests from the German Ministry of the Interior (BMI), which is responsible for reactor licensing, by funding reactor safety research in support of BMI's concern. Consequently, although BMFT has no responsibility for reactor licensing, it sponsors research to make both current and future plants safer.

One of BMFT's activities is to conduct reactor safety research and to provide knowledge for judgements that must be made in the future. They are concerned with the reduction of the residual risk from nuclear power plants. Risk studies are conducted for certain plants for BMFT, although BMFT is not involved in the application of these studies. BMFT also is concerned with the protection of personnel from radiation and employs actual plant inspection for this purpose.

Risk studies are conducted based on a probabalistic analysis approach, along the lines of the Rassmussen WASH-1400 study. It is felt that WASH-1400 is not directly applicable to nuclear power in Germany due to the greater population density there and the different types of plants. Their risk studies do not rely entirely on probabalistic analysis but also include deterministic analysis of the plants and engineering optimization of safety systems. Experiments are conducted in support of these analyses. Judgements are made on improving component performance where necessary.

It is not BMFT's task to consider mitigation of core melt accidents. However, some of the results obtained in work that they are sponsoring can be applied to mitigation. A decision on the need for retrofitting reactors with core melt mitigation devices is the responsibility of the Ministry of the Interior.

The Federal Ministry of the Interior (BMI) has issued a report on TMI which concludes that core melt accidents remain hypothetical. Consequently, from a legal viewpoint, no countermeasures are required. This situation may change when practical methods for mitigation are developed. If BMI decides that core melt mitigation studies are needed, then BMFT will respond.

If there is a big improvement in knowledge and technology, the German regulations require that this must be taken into account during the licensing process.

V. Meeting at the German Ministry of the Interior (BMI), Bonn Germany

Later, on the afternoon of December 1, 1980, we met with Dr. H.E. Dreisvogt, Dr. H.H. Buechler, Dr. L. Weil, and Mr. K.P. Bachus at the offices of the German Ministry of the Interior (BMI) in Bonn, Germany. The responsibility for reactor licensing in Germany is divided between BMI and the individual Lander (or States). Although the Minister of the Interior has the authority to license nuclear plants, this authority has been delegated to the states.

Currently, no measures are required to mitigate Class 9 accidents in Germany. Special features are not required for core melt accidents although studies have been funded to examine the possibilities. At present, they believe that it is best to minimize the probability of a core melt accident, rather than requiring mitigation features. It should be noted that there is some disagreement in this area. The RSK (the German equivalent of the ACRS) feels that there is no reason to install features to mitigate a core melt accident. On the other hand, the BMI staff concerned with licensing believe that precautions are needed. They believe that research and development is needed in this area, but that it will be at least 5 years before any requirements are imposed.

BMI is looking with great interest on the American ZIP studies. They indicated that whatever mitigation features were incorporated in the U.S.A. would probably also have to be required in Germany due to the pressure of public opinion.

One reason for public concern is the high population density in Germany which is as great on the average as the population density near the ZIP plants. Remote siting of nuclear plants is not an available option in Germany due to a lack of space.

As previously indicated, American actions will also be closely watched because German regulations require that they consider advances in technology in their licensing procedures. The last German license for a nuclear plant was issued three years ago. Rioting has occurred at plant sites and licensing has consequently become much more difficult.

VI. Meeting at the Company for Reactor Safety (GRS), Cologne Germany

On the morning of December 2, 1980, we met with Dr. Banaschik of Gesellschaft fur Reaktorsicherheit mbH (Company for Reactor Safety) or GRS. Although GRS is a private company, 46 percent of its shares are held by the Federal Government and 8 percent are owned by the states. The remainder is held by the "technical supervisory committees" (TUV), which are technical bodies set up by each local state. GRS advises and acts as a technical consultant to the reactor licensing segment of BMI and to the German states. Its purpose is to advise and harmonize licensing activities. GRS works for both BMI and BMFT with 15 percent of its resources devoted to BMI research, 40 percent to BMI calculations, and 40 percent to BMFT research.

There are seven separate licenses required in the construction of a nuclear plant in Germany, related to the installation of various major components. An overall safety evaluation report is prepared and the plant is required by law to meet the "state of the art". In the licensing process, the applicant initially applies to BMI, which transmits the information provided to the local state involved. The state then seeks assistance in evaluation of the request from the local TUV which may or may not be capable of providing assistance. If the local TUV cannot provide help, it seeks assistance from GRS. The state then sends a report to BMI, which in turn, asks GRS to evaluate the report. Once BMI has approved a project, the local state government cannot veto it as far as is known (the matter has not yet been tested in the courts).

The technical problem which most concerns the Germans in the event of a core melt is hydrogen stratification and burning. If steam is present in the containment, the hydrogen detonation point will not be reached even if the hydrogen concentration is quite high. However, if the steam is condensed, as by cooling the outer containment steel shell with water as proposed by KWU, conditions will be reached where the hydrogen may burn. In two compartment calculations, hydrogen eventually concentrates or stratifies in the upper compartment, even when it is introduced at the bottom. Consequently, some type of mixing device may be needed to maintain a uniform concentration and prevent the development of regions of higher than average hydrogen in concentration.

Our most interesting moment came when Dr. Banaschik advised us that the KWU plants have only one foot of space between the bottom of the reactor vessel and the floor of the reactor cavity (see Attachment III - Reactor Building). Consequently, there is no feasible way to install a core retention device in the existing standard German large dry PWR spherical containment plants if it should be decided that such a device is desirable. This fact has undoubtedly been influential in their consideration of the desirability of mitigating core melt accident consequences, especially those eminating from core debris-concrete interactions.

VII. Meeting at the Reactor Safety Commission (RSK), Cologne, Germany

Later, on December 2, we met with Dr. Armin Johns, Dr. Wolfgang Viefers, and Mr. U.I. Erven of the Reactor Safety Commission (RSK), a group analogous to the American ACRS.

After the TMI-accident, the RSK started to discuss possibilities to further reduce the risk of nuclear power plants. In the course of these discussions, Class 9 accidents played an important role. The possible responses were divided into two areas: A) measures, which prevent the core from meltdown or other degraded core damage, that is measures for accident prevention; and B) measures, which limit the consequences of a core meltdown and prevent a severe release of radioactivity to the environment, that is measures for consequence mitigation.

Measures of type A are characterized by the fact that they present an interference into a system which is in a physically well defined state. Therefore, these measures can be performed in a pinpointed and effective way.

Measures of type B, on the other hand, operate on a system which is, at least partially, in a poorly defined physical and chemical state (interaction between the molten core and its environment). The actual knowledge on this state is relatively poor. Furthermore, possibilities to acquire experimental data on such a system are limited.

Although measures of type B may appear attractive to some observers, possibly due to their apparent simplicity, RSK gave priority to measures of accident prevention. RSK believes that the concept of restoration or maintenance of decay heat removal by the emergency core cooling system, the RHR-system and emergency auxiliary feedwater system should be favored and that the necessity for hasty human actions should be precluded by automation.

With these premises, RSK proposed the investigation of the following measures for accident prevention:

1) Measurement of pressure and temperature of the primary system to generate a signal to indicate whether the primary system is in a subcooled or saturated state.

2) Remotely controlled degasification of the reactor pressure vessel or of the primary system.

3) Faster secondary shut-down operations to improve control of small leaks.

4) Improved temperature monitoring of the reactor core.

5) Provision of instrumentation to indicate the water level in the pressure vessel.

6) Further improvement of station service supply or use of diverse drive units for safety systems (pumps, armatures) respectively for better control of a failure of the emergency power supply.

7) Further improvement of the containment function, particularly of closing devices in the containment and the reactor building.

(8) Increase of the admissible pressure in the secondary coolant circuit for better control of breaks in steam generator tubes (requiring stronger pipes up to the turbine bypass station).

(9) High pressure injection from the sump.

(10) Further development of programs for simulator-training (different developments for disturbances and accidents).

(11) Improved display of the state of the primary system in the control room (electronic data processing).

(12) Fast access to the operation manual by use of electronic data processing.

(13) A computer for accident and accident consequence analysis.

(14) Information display outside of the control room for accident information.

(15) Further development of the concept for automated protection.

(16) Investigation of natural convection within the primary system in the case of two-phase conditions (including the dynamics of the pressurizer and the presence of non-condensible gas).

Mechanisms for a severe release of radioactivity to the environment are:

(a) Failure to isolate the containment at start of the accident or at the initiation of a core meltdown.

(b) Steam explosion in the reactor pressure vessel.

(c) Failure of the containment as the result of a slow pressure increase.

Mechanism (a) can only be mastered by preventive measures. This proves clearly that risk reduction to zero is impossible. On the other hand, the importance of further improvement of the containment becomes evident.

Available experimental information indicates that the energy release from a steam explosion would not suffice to destroy the reactor pressure vessel or the containment. Nevertheless, despite the low probability for the occurrence of a steam explosion which would fail containment, the RSK has recommended enhanced activities in this field.

Measures for core melt consequence mitigation may mainly be aimed at the later subsequent containment failure due to overpressure.

The following measures for consequence mitigation are under consideration:

(1) Heat removal from the containment by an outer containment water spray system - This would prolong the time for emergency response activities. Its technical feasibility, however, has to be demonstrated.

(2) Venting of the containment through filters or into a pressure suppression system - Further investigations and an evaluation of the merits are required.

(3) Development of a dry reactor building basemat to act against the melt through of the core and to diminish the production of gas and steam - A necessary relocation of the containment sump should be investigated.

Supported by basic research in this field, initiated much earlier than the TMI accident, detailed investigations are now under way. The RSK clearly favors the above mentioned activities over studies related to underground siting.

After evaluation of proposed measures for accident prevention and consequence mitigation, the RSK expects that safety engineering improvements might be possible. Due to the already achieved high safety level, any such improvements, however, may only be marginal. Nevertheless, the RSK strongly supports these studies and will watch closely for the realization of noticeable improvements.

On the other hand, with current knowledge, the RSK does not plan to recommend Class 9 accidents as design basis accidents, meaning incorporation of systems that are fully designed to cope with accidents into the licensing procedure.

VIII. Meeting at Interatom, Bergisch-Gladbach Germany

On December 3, 1980, we met with Dr. H. Friedrich, Dr. Vassebrecker, and Dr. Gilles at Interatom in Bergisch-Gladbach Germany. Interatom is a private concern and is wholly owned by KWU, which in turn is wholly owned by Siemens. Interatom is responsible for the design, development, and construction of the SNR-300 Liquid Metal Fast Breeder Reactor (LMFBR) under construction at Kalkar, Germany. Our interest in their activities focused on the design and construction details of the core retention system that is being installed in the SNR-300, for possible application to LWR plants.

The SNR-300 core catcher is constructed of 22 tons of urania bricks in a single layer encapsulated in steel. The bricks are 8 cm high and consist of 100 thin urania disks encapsulated in sheet metal. A fuel distributor is used to reduce the probability of recriticality. It is 60 cm high with a 2m diameter and is constructed from the same urania bricks. The core catcher is cooled by a NaK system rated for 5 MW. The total cost of the core catcher system is expected to be 30 million DM ($16 million). Figure 1 shows a vertical cut through the lower parts of the core-catcher. NaK enters a lower annular distribution pipe. From there, 16 parallel cooling coils curve to the center and wind back the same way at an upper level to an annular collecting pipe. A top view of the piping arrangement is shown in Figure 2. Note the NaK-inlet pipe, from which 16 curved cooling pipes wind to the center. The time-dependent penetration and resolidification of the sacrificial barrer is shown in Figure 3 for a 10 cm thick decay-heat producing UO_2-$Pu0_2$ layer at 900°C. The layer lies on a 8 cm thick barrier of cold UO_2 which is cooled from beneath. The maximum penetration is about 3 cm after 3.5 hr at a maximum downward specific heat flux of 7.7 w/cm^2. The resolidifications ends after 280 hr. Figure 4 shows the overall heat transfer characteristics of the core catcher and Figure 5 shows the transient temperature behavior of the sodium pool overlying the core-catcher.

In the SNR-300, the core retention system selected employed urania because it is one of the molten core debris constituents. Consequently, the core retention material does not introduce additional adverse interactions, such as additional chemical reactions, chemical incompatibilities, depression of the eutectic temperature or density differences.

Interatom may have some information available on thoria and will provide it to us. It was felt that thoria did not

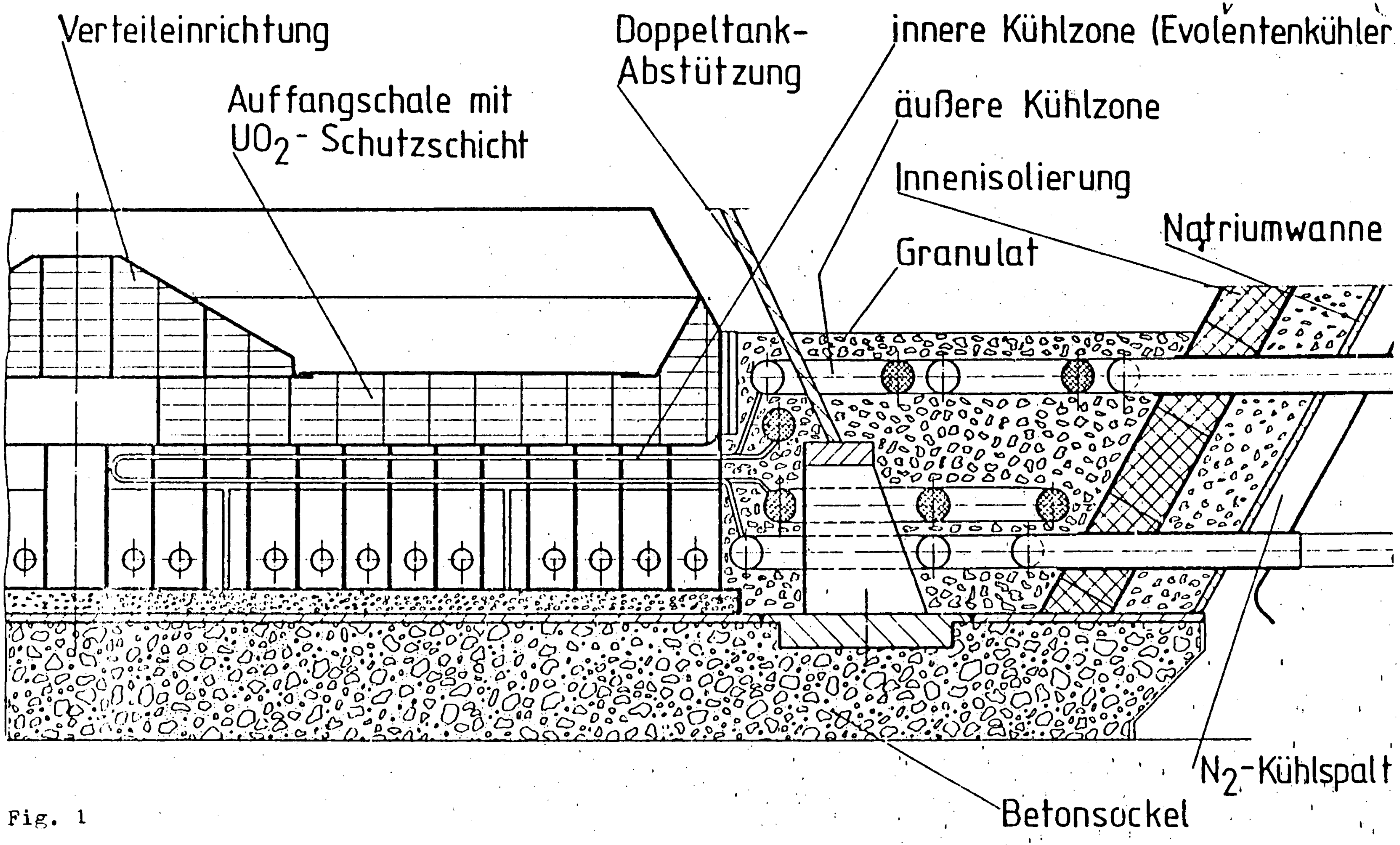


Fig. 1

Detail Bodenkühleinrichtung

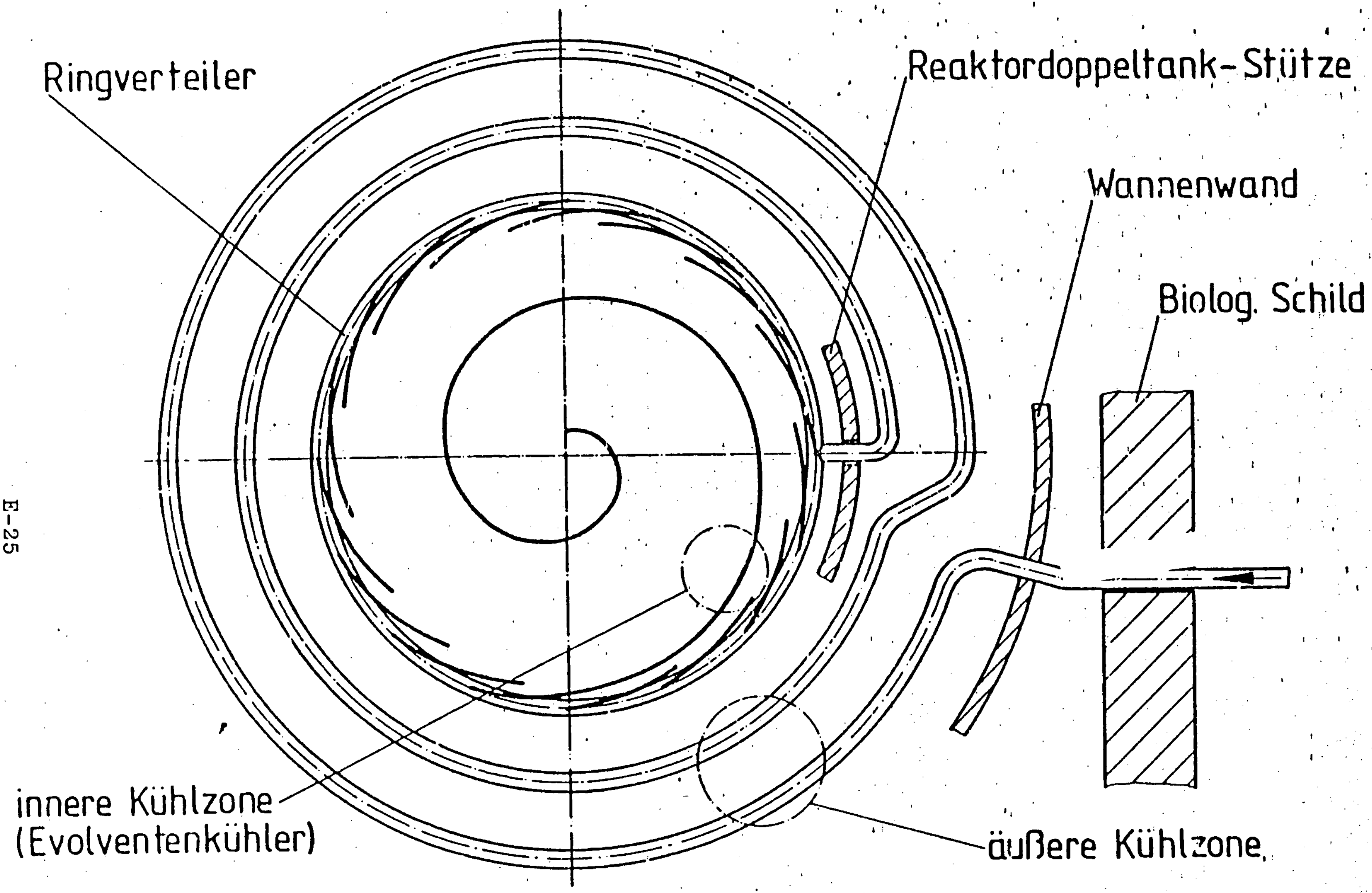


Fig. 2

Draufsicht Untere Zuflußebene Bodenkühler

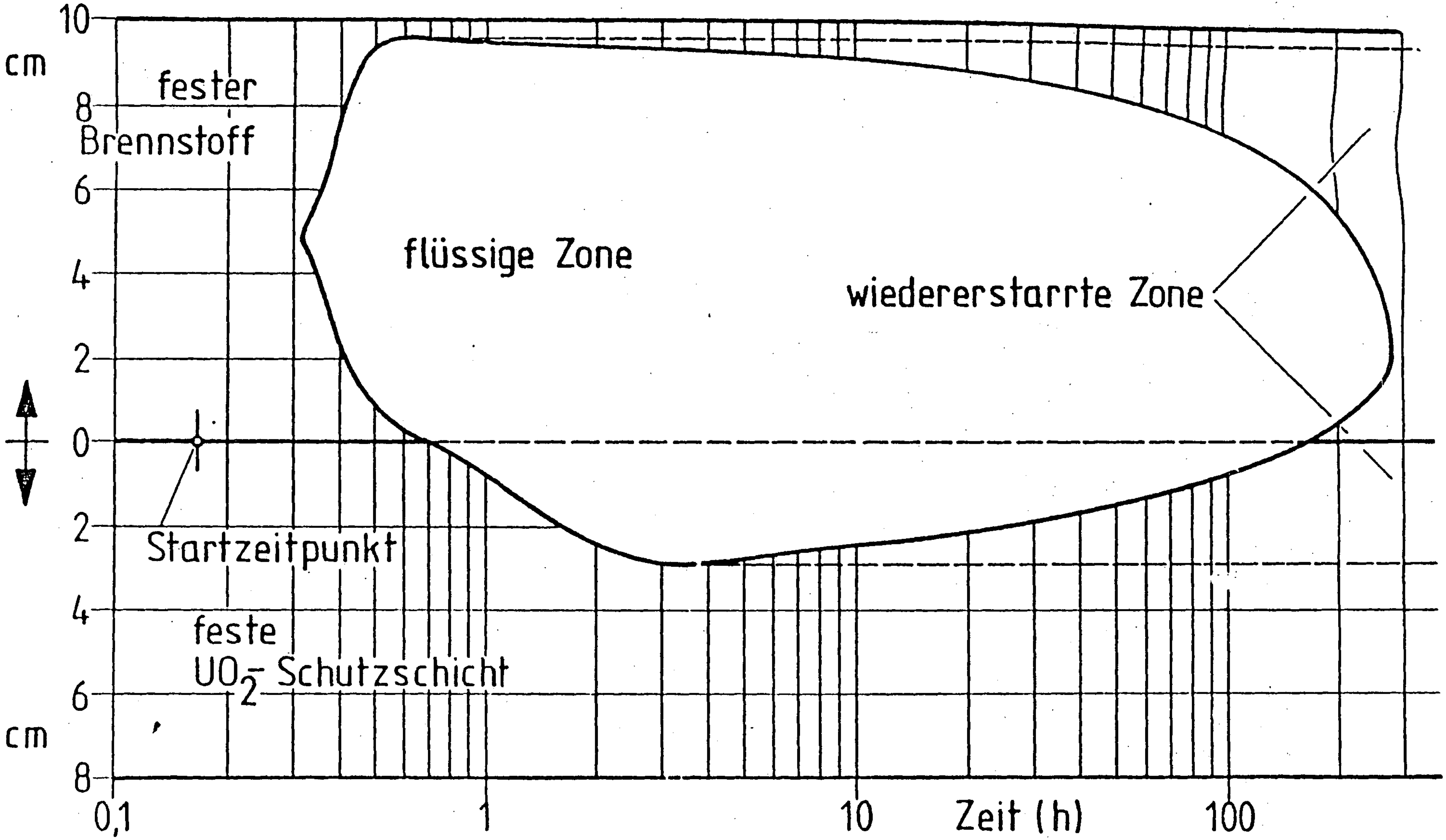


Fig. 3 Eindringen von Brennstoff in eine UO_2-Schutzschicht

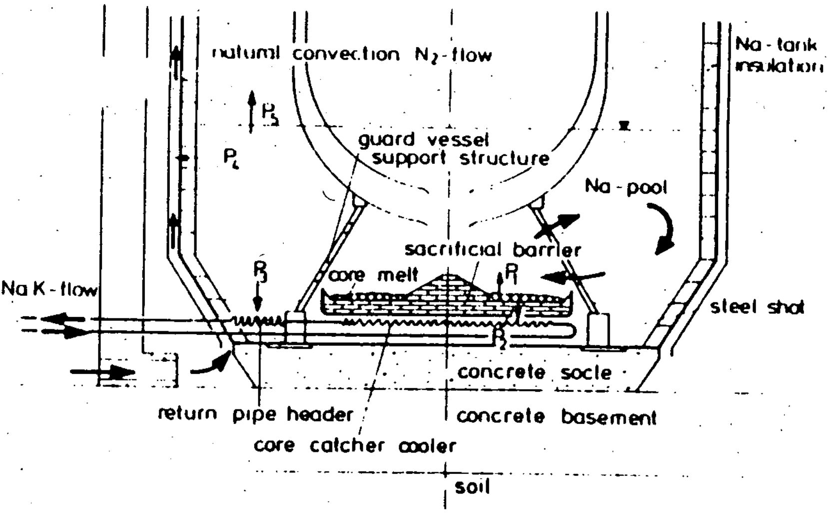


Fig. 4 Heat Transfer Characteristics of Core Catcher Na-Pool

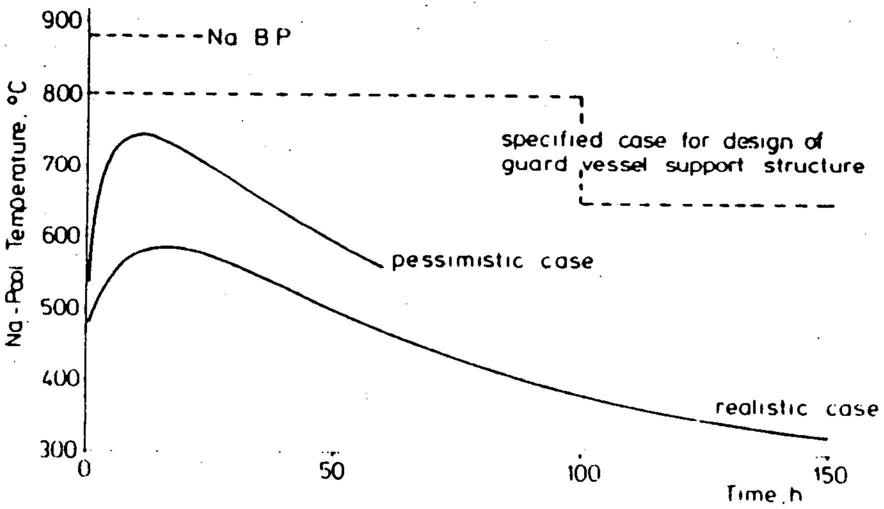


Fig 5 Dynamic Temperature Behavior of Core Catcher

offer any advantage over urania in the SNR-300 LMFBR environment (which is different from LWR's). They felt that thoria would be more difficult and expensive to fabricate. The waste from processing would also be more difficult to dispose of than urania. (In LWR applications, thoria is attractive because it is more resistant to iron oxide slag attack than other materials nad is compatible with water. Iron oxide is formed in LWR's by the reaction between water and molten iron.)

Even though a decision to install a core retention device in the Kalkar plant was made in 1973, there are no plans to install similar equipment in subsequent LMFBR plants because the Germans believe that they can contain a molten core in-vessel. Their in-vessel system would consist of a steel dish arrangement cooled by sodium flowing by natural convection. This sodium would be able to cool the steel even with a molten core sitting in the steel dish. We feel that they are very optimistic concerning their ability to calculate these phenomena, especially in the absence of any experimental data. It is not apparent that they have done the coupled heat transfer, structural analysis calculations needed to verify that the molten core debris can be contained in-vessel without the steel structures failing.

We had the opportunity to tour the Kalkar station including the areas where the core catcher will be installed. The station is most impressive and is now nearing completion. The containment was particularly interesting compared to American LMFBR designs. It consisted of an inner steel lined, reinforced concrete containment building and an outer reinforced concrete confinement building. The thickness of the inner and outer concrete walls is 2m and 1.5m, respectively.

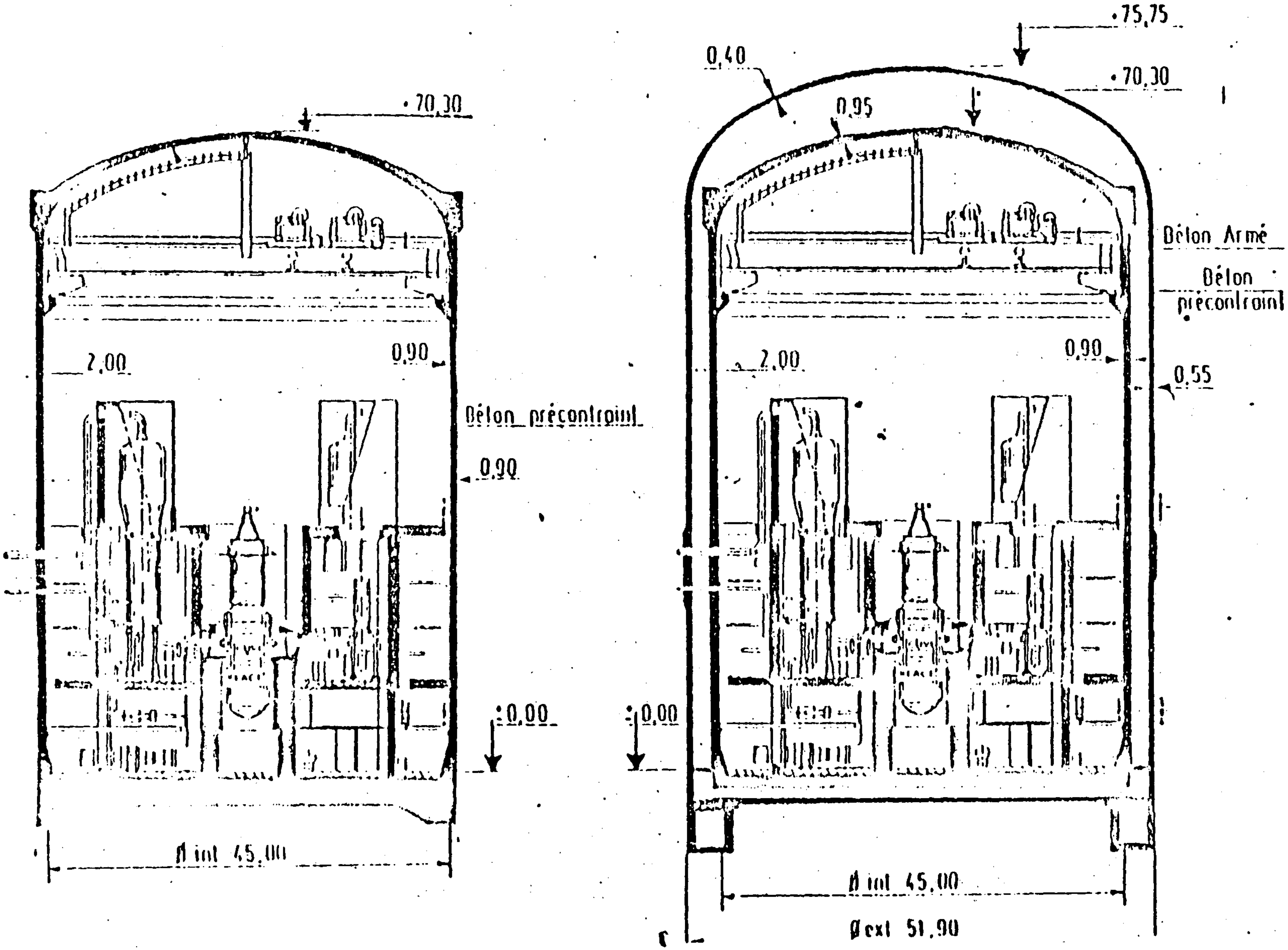
IX. Meeting at the Commissariat A L'Energie Atomique, Fontenay-aux-Roses, France

We met with Drs. J. Dufresne, Dr. Y. Cornille, Dr. R. Avet-Flancard, Dr. J. Duco, Dr. M. L'Homme, Dr. R. Zammite, and Dr. D. Costes of the Commissariat A L'Energie Atomique (Atomic Energy Commission) or CEA on December 5, 1980. The above individuals are involved in nuclear safety analyses for LWR's. A copy of the agenda can be found in Attachment IV. The meeting was well organized and detailed presentations were given for most items on the agenda. The French were very forthright in their presentations. We were surprised that the French are doing a considerable amount of work related to degraded core/core melt accidents in LWIR's.

A. Core Melt Sequence Analyses

Core melt sequence analyses are being used to obtain a better understanding of containment response, to compare different containment concepts and to improve the safety of LWR plants. The French have both single and double containment plants (see Figure 6). The single containment plant is a steel lined, reinforced concrete structure. In the double containment plants, both walls are constructed from reinforced concrete and the air between the two containments is filtered continuously. The design pressure of the inner containment is 4 bars at 140°C, and the outer containment is designed for 1 bar. There is less safety component redundancy in French plants (Figures 7 and 8) as compared to American plants (Figure 9).

As a consequence of the different designs, the accident sequences are somewhat different from those expected in American plants. In the case of the double containment, the French concluded that two modes of containment failure were possible. These were penetration of the concrete basemat by core debris and overpressurization of the containment.


SIMPLE ENCEINTE

DOUBLE ENCEINTE

Fig. 6 - ENCEINTES DE REACTEURS PWR 1300 MWe

AD ~ A[illegible]
AF ~ AC
AG ~ AC
AH ~ AD ~ AB.

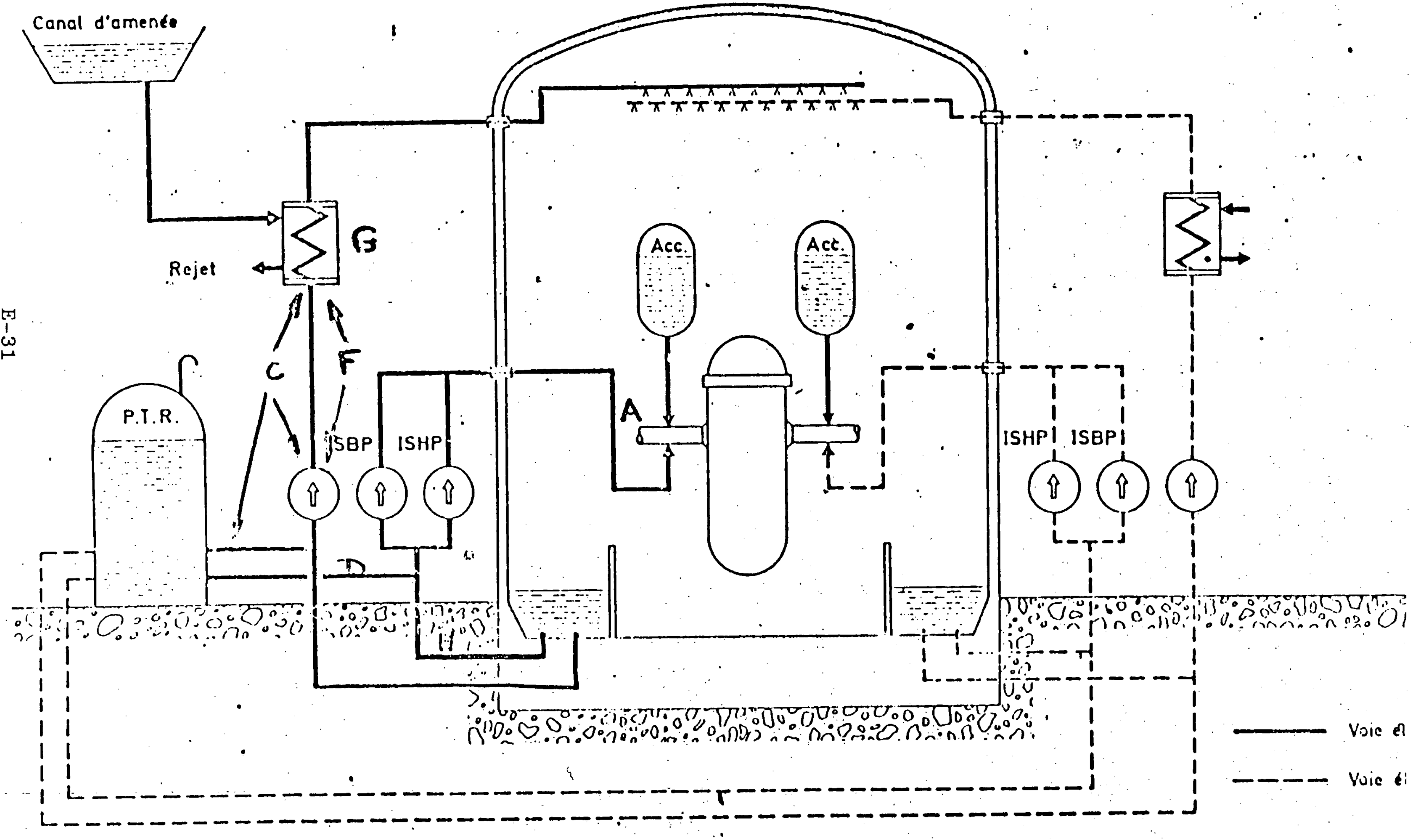


Fig. 7 _ Systèmes de sauvegarde des PWR 900.

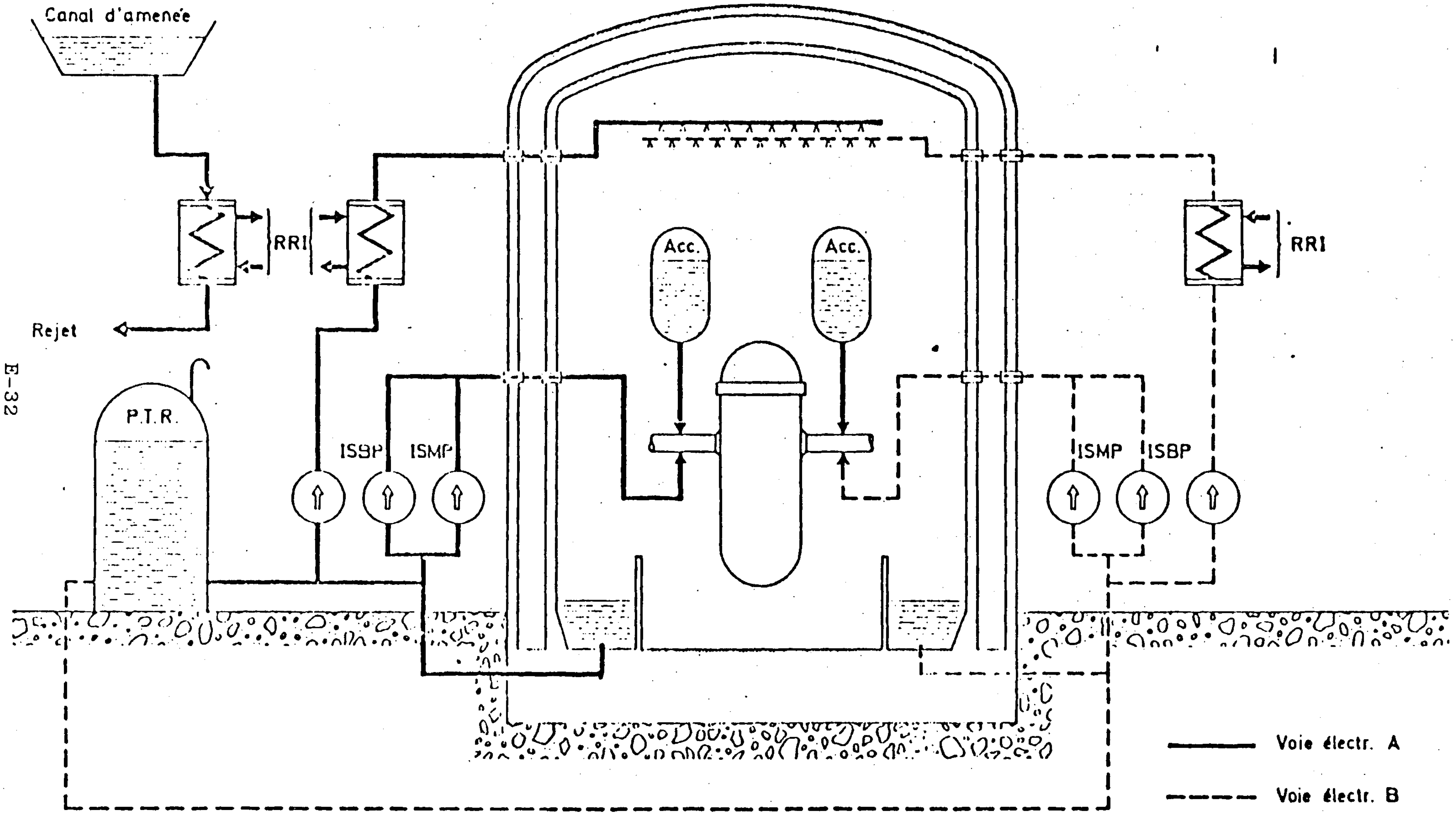


Fig. 8 _ Systèmes d[illegible]auvegarde des PWR 1300

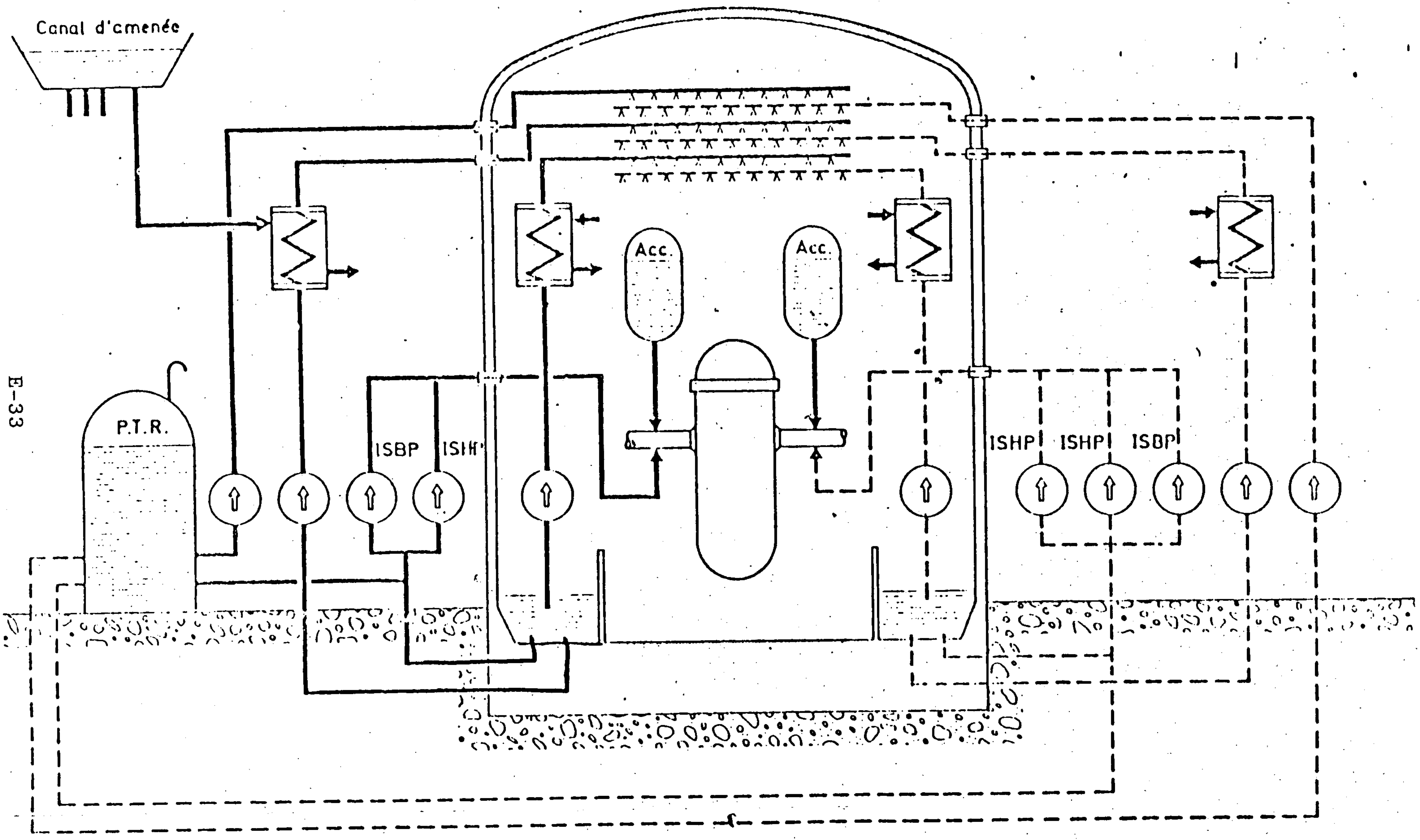


Fig. 9 _ Systèmes de sauvegarde du réacteur SURRY

The containment transient pressure response to TMLB' core melt sequence looked similar to what we are calculating for the ZIP plants. For their single containment, which is designed for 4 bars, they are predicing that it would fail at about 8 bars which is reached in about 24 hours for a TMLB' accident sequence.

B. Failure Mode of the Containment

A very detailed failure mode and location study for their containment building is being conducted; this study is more extensive than what we are doing in the U.S. Realistic failure modes are being analyzed for the liners, the reinforcement, and the concrete. This effort should be closely followed by NRC.

The containment failures modeling study has as its objectives the determination of the thresholds and modes of failure when the containment is subjected to internal pressure and temperature greater than the design value. It is also planned to study the probable failure locations, the nature and size of the failures, and the leakage rate versus increasing internal pressure.

Both the single and double walled containments will be studied. The accident sequence selected is a LOCA associated failure of the ECCS and the containment heat removal spray system (an ADG sequence in WASH-1400 terminology). A variety of models and computer codes are being employed. It is assumed that major penetrations are not weak points.

Preliminary results indicate the structure is damaged initially by relaxation of the prestressing, without concrete damage, followed by crack initiation at the outer face of the wall in the range of 8-12 bars internal pressure. Cracks then propagate through the wall and the steel reinforcement yields. They believe they have a safety factor in the range of 2-3 for the pressure at which containment would fail, following a core-melt accident.

C. Hydrogen Generation and Control

There is concern about possible stratification of hydrogen in some regions following degraded core/core melt accidents. An effort is being made to evaluate the usefulness of recombiners, particularly in the areas of capacity and cost. Various mitigation schemes are being examined, including water droplet sprays, and the use of halons. Inerting the containment with either nitrogen or carbon dioxide is also under study.

D. Corium-Concrete Interactions

No presentation was given in this area. We were told that no significant resources have been committed to this area; however, they are following with great interest the German and American work in this area.

They will look at better types of concrete to mitigate core concrete interactions. They did not indicate what they planned to use. We pointed out the possibility of using refractory concretes but also mentioned that their high water content could lead to overpressurization by hydrogen generation and/or burning.

CEA requested a copy of the Sandia CORCON code for predicting the penetration of core debris into concrete and the recent Sandia reports on ZIP. Mr. Marchese agreed to pursue this with NRC's Office of Research.

E. Fission Product Filtration During an Accident

The objective of this program is the definition and qualification of a system capable of containment 99 percent of the activity in a PWR containment in the event of an accident during an intentional depressurization.

Filtration systems under consideration include existing filters available commercially, sand filters, and the use of an alkaline pool. It is felt that the existing filters cannot cope with the large flows anticipated concurrently with steam. With the sand filter, there is a high by-pass risk due to steam condensation. They have considered using the intermediate space between the two containment walls as a storage tank.

Their preliminary design consists of a large alkaline pool with a possible iodine filter and an aerosol filter. A hydrogen burning device might also be employed.

F. Probabilistic Risk Analysis

No systematic overall probabilistic assessment of accident sequences is being conducted.

An effort is being made to conduct a systematic assessment of the probability of complete failure of safety related systems. This effort is focused on the period from initiation to the beginning of unacceptable situations and on mitigating procedures and/or devices. A regulatory assessment is also being performed of studies and very simple mitigation features to prevent overpressure failure which may be proposed by EDF, the French electric utility. In these studies, the emphasis is on the prevention of such situations as overpressure failure of containment. No systematic investigation is being made of such physical phenomena involved during core melting.

An assessment is being made of the probability of single event failures, such as reactor pressure vessel failure, blockage of pressurizer safety valves, and off-design basis earthquakes.

There is no external incentive to conduct an overall risk assessment of PWR's in France due to the general public acceptance of nuclear power. Furthermore, since an assessment was conducted in America (WASH-1400), there seems to be no further need in France for a similar effort. However, the French appear to be interested in improving the ability of their plants to cope with core melt accidents and are devoting some effort in this area.

There is also no internal interest to a similar overall risk assessment from the French licensing authorities. However, there is interest in using probabilistic assessment as a tool to help in making licensing decisions.

G. Rupture Probability of the Pressure Vessel

Detailed analysis are being conducted to determine the rupture probability for reactor vessel failure. Preliminary results indicate that the failure probability for pressure vessel rupture is much less than for a LOCA or a pipe failure. The French will present their results on this subject at the Fracture Mechanics Meeting in Cannes, France in April, 1981.

H. Advanced Decay Heat Removal and Containment Design Concepts

This presentation dealt with a number of advanced decay heat removal and containment design concepts, many of which are patented in France. A copy of the viewgraphs is attached (Attachment IV) and includes sketches of the concepts. The advanced decay heat removal proposals included a reserve of secondary steam to provide post-accident heat removal, an internal reserve of ice or cold water and improvements to the ECCS.

In the area of molten core retention, the use of "reception beds" was discussed. These included refractory concrete, which would slow down the penetration but would still generate large amounts of H_2O, CO_2, and CO on contact with molten core debris. Refractory beds of MgO, Al_2O_3 or UO_2, which would be penetrated slowly, were also suggested. Melting beds of glass or lead, which would require large volumes were mentioned. In addition, the use of graphite was discussed although it can engage in an exothermic, gas generating reaction with UO_2. Graphite is also highly soluble in molten iron.

Next, molten core cooling was discussed. Cooling coils might be used to cool the core melt debris. It was suggested that water could be used in two ways. A large thermal explosion could be avoided by the progressive introduction of water at a controlled rate. The French do not believe that flooding the melt with water is a good idea. On the other hand, fast cooling by water could avoid generation of large amounts of hydrogen. Finally, it was suggested that transpiration through a cracked material was a feasible means of containing molten core debris. This suggestion is similar to proposals under study in this country.

X. Meeting at Euratom, Ispra, Italy

During our meeting at Euratom in Ispra, Italy on December 8, 1980, we met with Mrs. Fasoli-Stella, Dr. Kottowski, Dr. Randles, Dr. Donea, and Dr. Hohmann. Our agenda is provided in Attachment V.

A. LOBI Facility Tour

We began with a tour of the $20 million LOBI facility which is essentially a German program intended to simulate loss of coolant accidents in PWR reactors. There are large and small break test matrices planned for this facility. The facility features a 5.4 megawatt electrical heating supply and simulates two loops including steam generators of a standard German PWR.

B. Vapor Explosions - Dr. Kottowski

Dr. Kottowski is examining solid particulates in his vapor explosion studies. Several patterns of coolant-melt contact and mechanisms which might lead to violent vapor explosions have been discussed in the past. Various physical models have been developed, which under particular boundary conditions, demonstrate a high probability of a vapor explosion. There is also experimental evidence that vapor explosions exist. Detailed knowledge on the various factors determining the occurrence of vapor explosions is still lacking.

Experiments have been performed in a shock tube using molten steel and UO_2-granulates into which water columns with different impact momentum have been injected within 3-6 millisec. The driving force for the impact momentum was produced by the pressure difference between the gas blanket above the water column and the melting crucible. In the UO_2, 500 μ particles heated to 1900-2100°C were used.

At melting and before the injection of the water column, the pressure above the melt was held at about 10^{-3} Torr for all tests. The blanket gas pressure was varied between 1 bar and 26 bars and the water temperature between 20°C and 220°C. The variables measured during the experiments were (a) the pressure generation as a function of time and (b) the vapor generation as a function of time. With the help of these variables, the mechanical work as a function of time was calculated. The experiments demonstrated that violent vapor explosions only occur when melt and coolant are mixed rapidly and intimately and a geometrical constraint is present.

Violent vapor explosions were observed and pressure peaks of up to 450 bars were measured. In case of partial channel blockage due to melt particles, carried with the coolant-vapor mixture, the pressure peak reached 750 bars. The increase of system pressure up to 26 bars did not prevent vapor explosion when fast fragmentation and mixing occurred.

These efforts are more of a scientific nature (rather than engineering) and are primarily being made to determine the maximum conversion efficiency under very ideal circumstances. When more material participates in a vapor explosion, less heat is converted to mechanical work due to poorer mixing, and thus the conversion efficiency is less. No decision has been made on whether to do fuel-coolant interactions with corium. Previous experiments have employed solid particles rather than molten liquids. The greatest conversion efficiency obtained was 3.6 percent with steel; with UO_2, the efficiency is 0.8 percent. With liquid-liquid interactions, it is felt that the efficiency should be less than 1-3 percent.

There are two important factors in vapor explosions, namely triggering and premixing. Triggering can be inhibited by increasing the pressure applied to the liquid. If premixing can be inhibited, no vapor explosion will occur.

Dr. Kottowski's calculations indicate that approximately ten tons of melt must be mixed homogenously with water to generate an explosion capable of failing a reactor pressure vessel. He believes that the probability of such a large vapor explosion is very small. However, it is not possible to prove that the melt and water cannot be mixed in the amounts required to generate a large vapor explosion.

One question of interest concerns whether a reactor cavity initially should be wet or dry. If the cavity is initially wet, the water provides a heat sink but a vapor explosion is possible. In a dry cavity, there is no risk of a vapor explosion but there is the possibility of substantial concrete attack by the core debris. If water is then added to core debris in a dry cavity, Dr. Peehs of KWU, as indicated earlier, has experimental evidence to show that a thick crust would form over the molten core debris and concrete penetration would still continue. On the other hand, Dr. Kottowski indicated that he has experimental evidence which shows different results. A crust forms immediately on the molten core debris and when water contacts this crust, it breaks open and never closes. When water contacts the melt initially, cracks open and form steam geysers. The debris is cooled as more water enters the crack. The interface is very agitated and particles are emitted and freeze, generating domes, craters, and cracks. This refragmentation can generate debris particles. Consequently, it is suggested that water should be introduced at a controlled rate on the surface of a melt to preclude an explosion. The cavity should not be preflooded as this will simply break open the melt crust increasing the probability of a vapor explosion. For these reasons, Dr. Kottowski feels that it is better to have jets of water fall on the melt than to have the melt fall into a pool of water.

C. SUPERSARA - Dr. Randles

This program was initiated in 1975 under Italian sponsorship. It currently enjoys wider support from the European Community and is also receiving sponsorship from NRC. Testing is expected to commence after 1983.

Small, medium, and large pipe break effects can be examined in the SUPERSARA loop. A reactor provides a very flexible high flux source of neutrons that are highly thermalized. Severe fuel damage tests in pile will be conducted in this facility provided that it can be licensed for these tests. It is recommended that NRC follow this program closely as it is very relevant to the degraded core accidents that we are studying. There is a need for better coordination between NRR and RES on SUPERSARA. Perhaps RES should even consider having a resident engineer at Ispra to follow this program as well as others covered in this report.

The severe damage program will cover TMI plus other sequences where core uncovering is likely. It will measure fission product release from the control rods but not their transport and deposition.

There are five accident steps that can be examined in these tests. These include clad ballooning and rupture, clad oxidation, and clad melting and dissolution of the urania pellets. The formation of a frozen Zr/UO_2 slag and the formation of a rubble bed, due to fragmentation after the core is reflooded and quenched, can also be examined.

They will determine whether or not they have a rubble bed after the core is uncovered in their tests. As there may be a limited core melt, it is necessary to design a way of containing the molten core. Consequently, a core-catcher design study has been initiated and should be closely followed by NRC. Materials under examination for this purpose include molybdenum and a "zirocolay" cloth. The latter is a new material under development at Pacific Northwest Laboratories

(PNL) and can be made into a monolithic assembly. The cloth is made of zirconia and reputedly provides protection to 3000°K. RES should make contact with PNL (Ed Courtwright) and obtain information on this material. Great interest was expressed in the core retention work that NRC is sponsoring.

D. Development of Models and Codes - Dr. Donea

Dr. Donea is doing detailed analytical studies on fluid-structural interactions for the LMFBR. He is developing computer models to calculate the structural behavior of the reactor vessel during core disruptive accidents. The models are being applied to the French Super-Phenix reactor.

E. PAHR Out of Pile Studies; FARO Project - Dr. Hohmann

Dr. Hohmann has conducted fuel-coolant interaction work in which high temperature UO_2 and steel were dropped into water. In a typical experiment, 2 kg of UO_2 were released into 200 l of water. The ratio of melt to water varied from 1/1000 to 1/2. With steel at 1800°C or UO_2 at 2900°K, and water in the range between 20-80°C, no vapor explosion was observed. Fragmentation occurred and the fragment size varied with the melt-water ratio. With relatively large quantities of water, the fragment size, in the case of stainless steel, is small (100-500 μ). With small quantities of water, the fragments were very large chunks.

The FARO-THERMOS facility is a huge new facility under construction for melting fuel and studying interactions with water, concrete, or core retention materials on a relatively large scale. The equipment will use 1.5 megawatts to melt 100 kg of urania for this purpose. This is another Ispra program which NRC should follow very closely.

ATTACHMENT I

Agenda for the PNS Colloquium

Wir erlauben uns, Sie zum

Jahreskolloquium 1980

des Projekts Nukleare Sicherheit
Kernforschungszentrums Karlsruhe
einzuladen.

E-44

You are invited to the

PNS Annual Meeting 1980

at the
Karlsruhe Nuc[illegible]esearch Center

Programm — Program

Montag, 24. November 1980

19.00 h	**Empfang** **Reception** ~~Gastdozentenhaus "Heinrich Hertz", Karlsruhe, Engesserstr. 3~~ SEE ATTACHED CHANGE NOTICE

Dienstag, 25. November 1980

9.00 h	**Eröffnung und Einführung** **Welcome and Opening Address** H. H. Hennies, KfK

Sitzung 1 — Session 1

Vorsitzender — Chairman: H. H. Hennies

9.15 h	**Menschliches Fehlverhalten: Ein Kernproblem der Kernkraftwerke?** **Human Failure: A Basic Problem of Nuclear Power Plants?** D. Smidt, KfK
10.00 h	**Fortschritte der Reaktorsicherheitsforschung im Projekt Nukleare Sicherheit** **Recent Advances of Reactor Safety Research in the Nuclear Safety Project** H. Rininsland, KfK
10.45-11.15 h	**Kaffeepause** — Coffeebreak
11.15 h	**NRC's Core Melt Research Program and its Relation to Current Regulatory Activities** T. E. Murley, U. S. Nuclear Regulatory Commission
12.00 h	**US Steam Explosion Research: Risk Perspective and Experimental Results** M. Berman, Sandia Laboratories; R. Sherry, U. S. Nuclear Regulatory Commission
12.45-13.45 h	**Mittagessen** — Lu[illegible]

Sitzung 2 — Session 2

Vorsitzender — Chairman: H. Rininsland

13.45 h	**Analyse des Ablaufs hypothetischer Kernschmelzenunfälle** **Analysis of the Course of Hypothetical Cor** **Meltdown Accidents** K. Hassmann, Kraftwerk Union Erlangen; M. Reimann, Institut für Reaktorbauelemente, KfK
14.15 h	**Zweiphasenmassenstrom-Messungen: Ein Vergleich verschiedener Meßverfahren** **Two-Phase Mass Flow Measurements: Comparison of Different Methods** J. Reimann, H. John, U. Müller, Institut für Reaktorbauelemente, KfK
14.45 h	**Methoden zur Fluid- und Strukturdynamik bei der Analyse von Störfällen in LWR** **Methods of Fluid and Structural Dynamics Applied to Postulated LWR Accidents** R. Krieg, U. Schumann, Institut für Reaktorentwicklung, KfK
15.15 h	**Messung der Brennstab-Hüllrohrtemperatur mit LOFT-typischen Thermoelementen unter Blowdown-Bedingungen in COSIMA** **Measurements of Clad Temperatures with LOFT-typical Thermocouples in the COSIMA Facility under Blowdown Conditi** G. Class, R. Meyder, Institut für Reaktorentwicklung, KfK; K. Hain, Hauptabteilung Ingenieurtechnik, Kf
15.45 h	**Schlußwort** — Closing Remarks

Diskussion nach jedem Vortrag
Discussion after each paper

ATTACHMENT 1

ATTACHMENT II

Agenda for US/FRG Core Melt Information Exchange Meeting

AGENDA

Information Exchange Meeting BMFT / NRC
Common Review Group Meeting Core Melt Research

November 26, 1980

TOP. 1	9.00 h	On going activities in FRG core melt research	BMFT
TOP. 2	9.20 h	Status of NRC-research program "Severe Fuel Damage and Core Melt Research"	NRC, R. Sherry
Coffee Break	9.40 h		
TOP. 3	9.55 h	Steam explosions theoretical and experimental investigations in steam explosions, overview	EURATOM, Dr.K[illegible]vs
TOP. 4	10.25 h	Summary of FITS-experiments	NRC D.Mitchell, SANDIA
TOP. 5	10.55 h	New investigations in code development:	
		- Thermal detonation model	Unger/Schwalbe, IK
		- New code development and calculations in USA	NRC M. Corradini,SANDI
Lunch	11.55 h		
TOP. 6	14.00 h	Analysis of MARCH-KESS comparison calculations	NRC, P.Cybulskis,B Dr. Hassmann, KWU
TOP. 7	14.40 h	New results of the ZIP study	NRC, M. Berman, SA[illegible]
		- H_2-deflagration and pressure build-up	
		- alternative containment concepts	
Coffee Break	15.20 h		

TOP. 8	15.30 h	Melt-concrete-interactions	
		- Status of the BETA-experiments	Dr.Hosemann, PNS
		- Model description of the melt-concrete-interaction with CORCON	NRC, J. Muir,SAND
		WECHSL	Dr. Reimann, KfK
		KAVERN	Dr.Hassmann, KWU
TOP. 9	17.00 h	EPRI-Core melt activities	Ritzmann,EPRI
	17.15 h	End Transfer by busses	
	19.30	Heinrich Hertz-Haus	

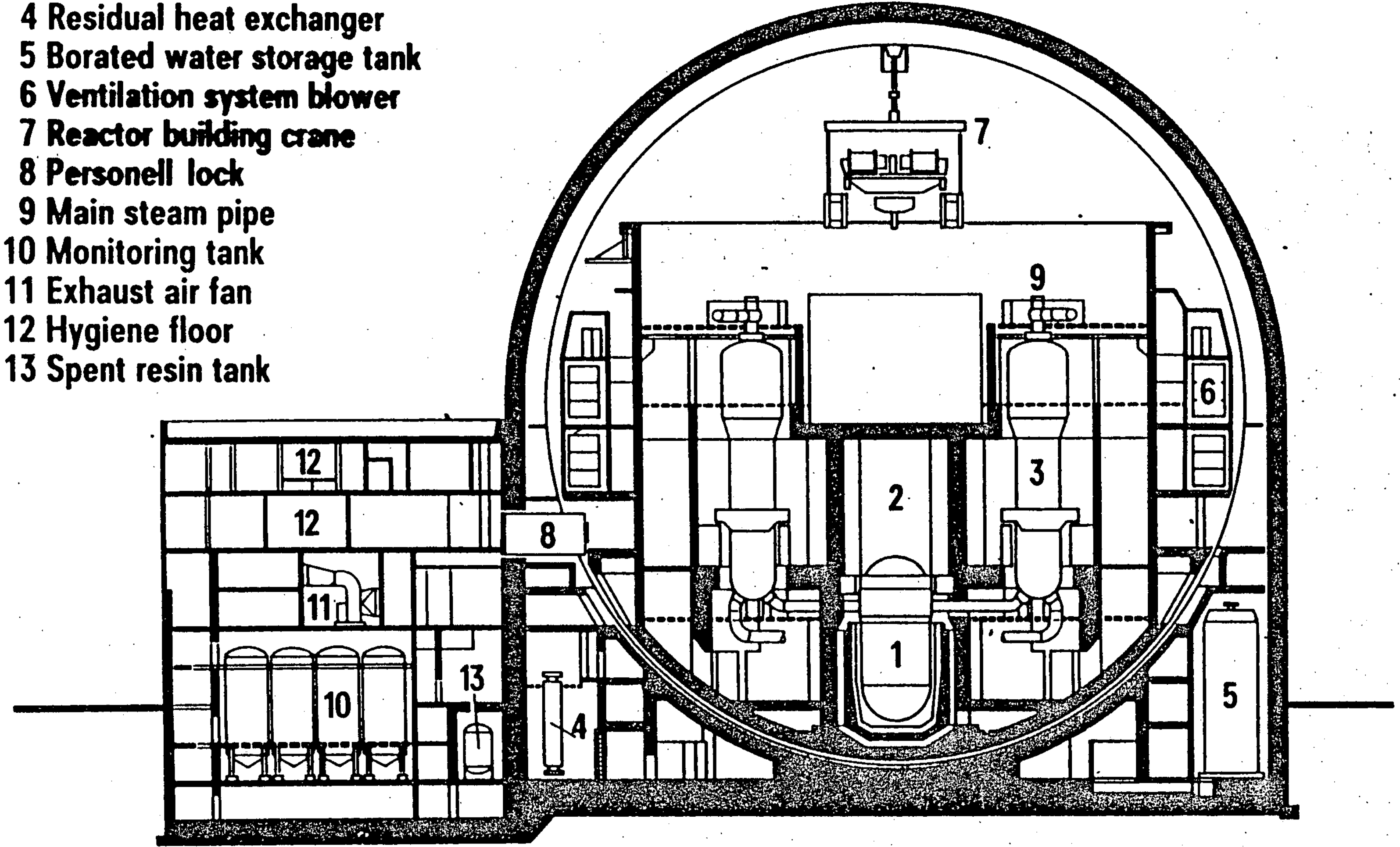
<u>November 27, 1980</u>

TOP.10	9.00 h	Status of SASCHA experiments	Dr.Albrecht,Kf
Coffee Break	9.40 h		
TOP.11	9.55 h	High temperature fission product release - planned ORNL-tests	A. Malinauskas,ORN
TOP.12	10.25 h	Status of NAUA code - analysis and development	Dr.Schöck, KfK
TOP.13	10.55 h	Status of TRAP-MELT code further code development plans	J.Gieseke, BCL
TOP.14	11.20 h	Chemical forms of Cs and I under severe accident conditions - discussions -	NRC, FRG
	11.50	Discussion	
	12.30	Lunch	

ATTACHMENT III

Meeting at Kraftwerk Union (KWU)

1 Reactor pressure vessel
2 Reactor compartment
3 Steam generator
4 Residual heat exchanger
5 Borated water storage tank
6 Ventilation system blower
7 Reactor building crane
8 Personell lock
9 Main steam pipe
10 Monitoring tank
11 Exhaust air fan
12 Hygiene floor
13 Spent resin tank



Section

PWR 1300 MW
Reactor Building and Reactor Auxiliary Building

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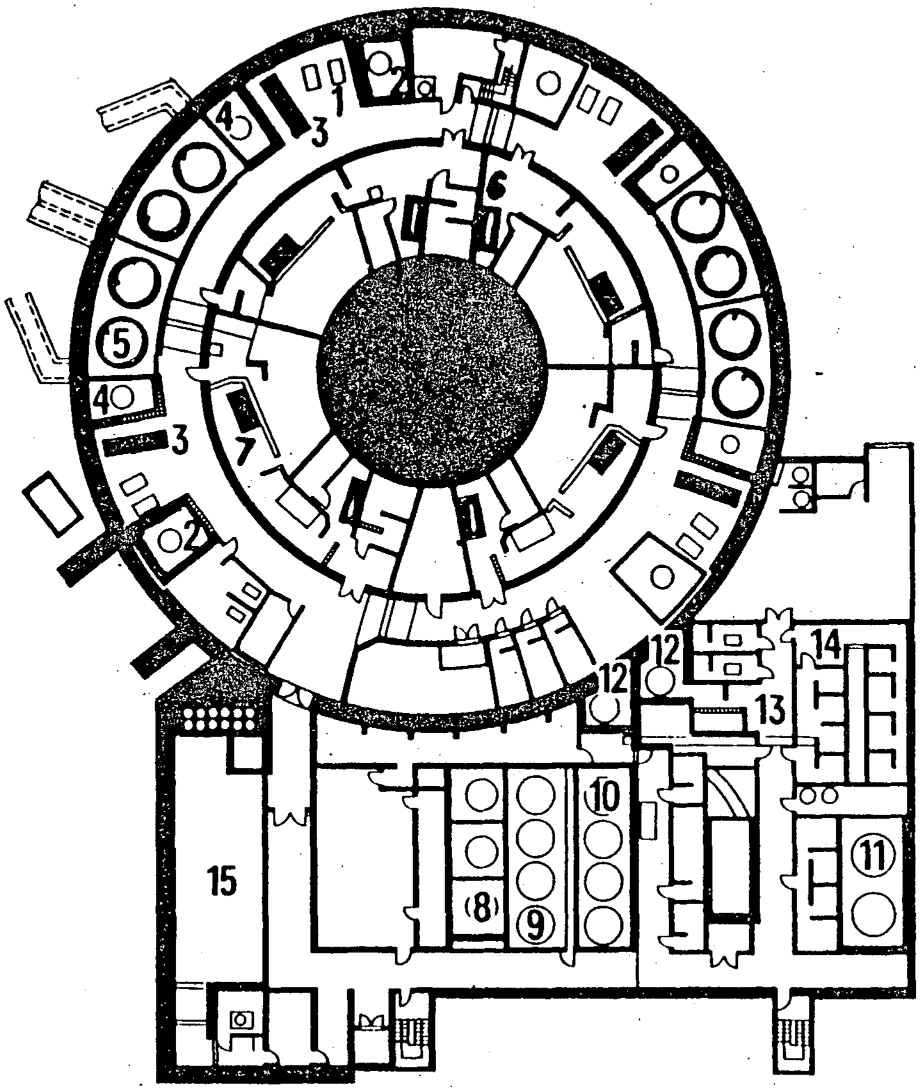
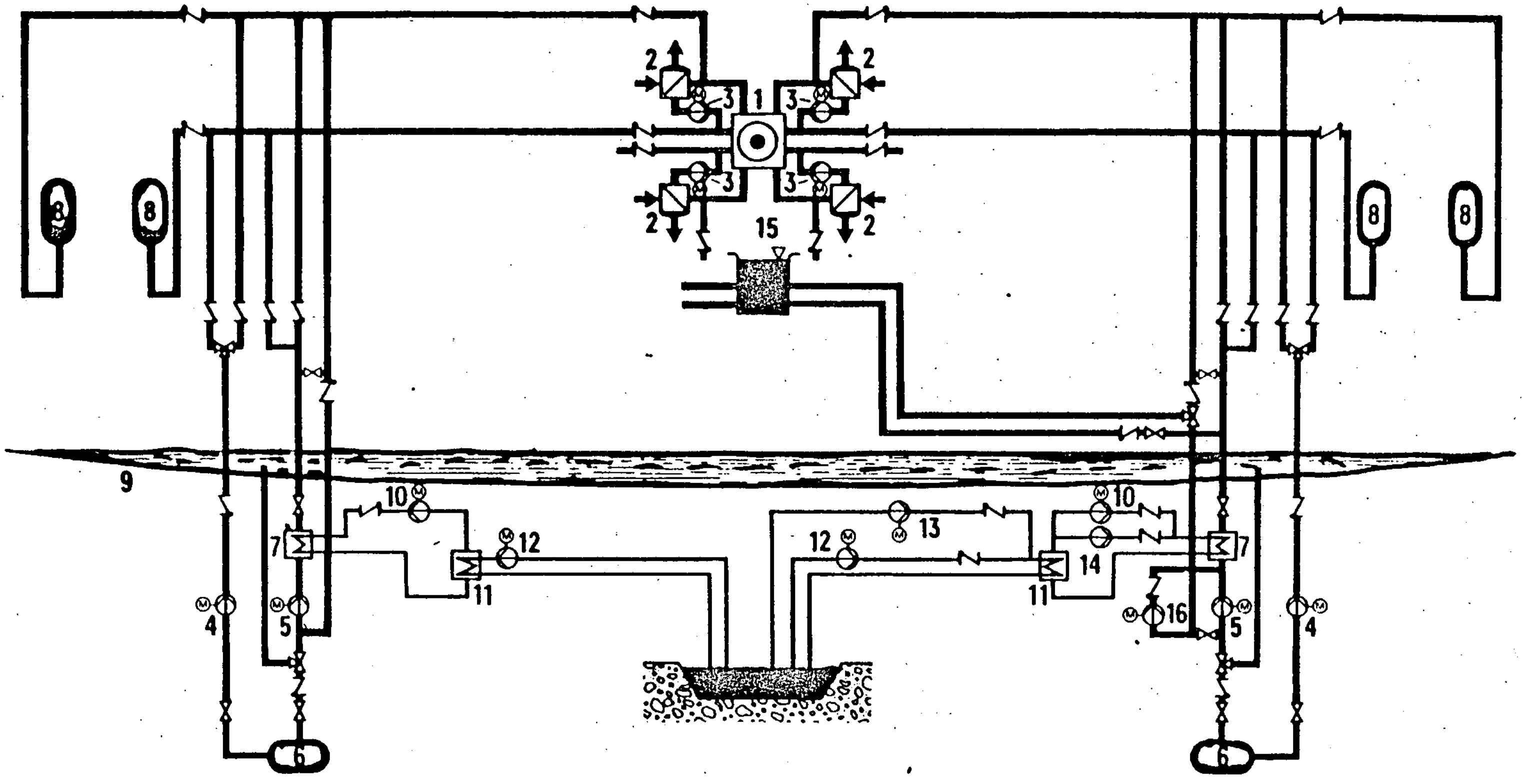


1 Emergency component cooling pump
2 Component cooling heat exchanger
3 Safety injection pump
4 Residual heat exchanger
5 Borated water storage tank
6 High pressure charging pump
▨ Residual heat removal pump
8 Concentrate tank
9 Liquid-waste collecting tank
10 Monitoring tank
11 Boric acid tank
12 Spent resin tank
13 Gaseos waste
14 Pumps and valves room
15 Drum store

Plan view — 6.00 m

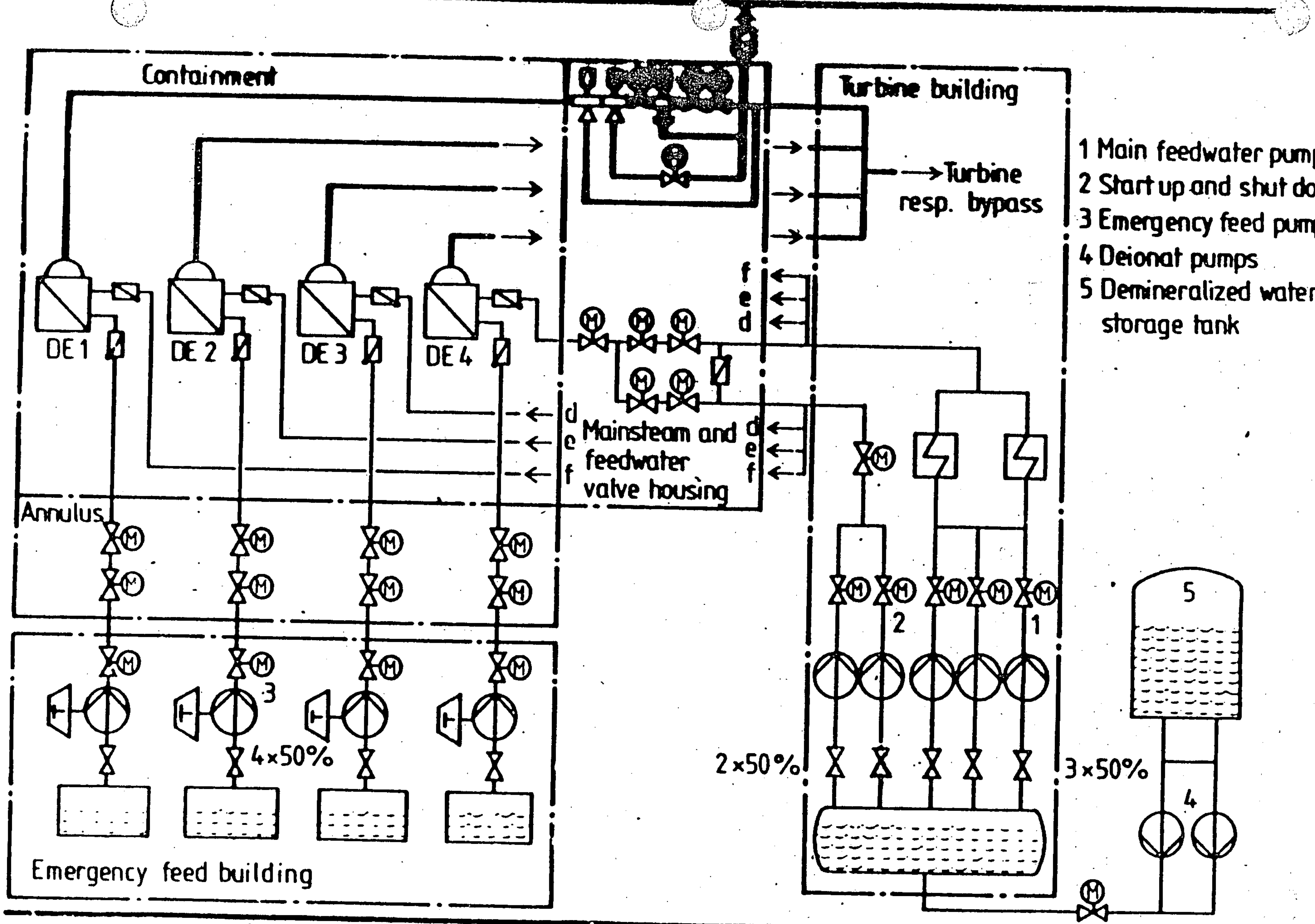
PWR 1300 MW
Reactor Building and Reactor Auxiliary Building

Kraftwerk Union



1 Reactor
2 Steam generator
3 Main coolant pump
4 Safety injection pump
5 Residual heat removal pump
6 Borated water storage tank
7 Residual heat exchanger
8 Accumulator
9 Containment
10 Component cooling pump
11 Component cooling heat exchanger
12 Secondary cooling pump
13 Emergency secondary cooling pump
14 Emergency component cooling pump
15 Spent fuel pit
16 Spent fuel pit cooling pump

Residual heat removal system and emergency core cooling system



1 Main feedwater pumps
2 Start up and shut down pumps
3 Emergency feed pumps
4 Deionat pumps
5 Demineralized water storage tank

Main Steam System and Steam Generator Feeding

V11/4.79

ATTACHMENT IV

Meeting at the Commissariat A L'Energie Atomique

MEETING WITH M. MARCHESE AND SWANSON

5 TH DECEMBER 1980

AGENDA

MORNING

- CORE MELT SEQUENCE ANALYSES	J. DUFRESNE
- CONTAINMENT LOADING : PRESSURE AND TEMPERATURE EVOLUTION, MELT TROUGH AND SENSIVITY ANALYSIS	Y. CORNILLE
- FAILURE MODE OF THE CONTAINMENT	R. AVET-FLANCARD
- HYDROGEN GENERATION AND CONTROL	J. DUCO
- CORIUM CONCRETE INTERACTION	

AFTERNOON

- FISSION PRODUCT FILTRATION DURING ACCIDENT	M. L'HOMME _ PE
- PROBABILISTIC RISK ANALYSIS	R. ZAMMITE
- RUPTURE PROBABILITY OF THE PRESSURE VESSEL	J. DUFRESNE
- ADVANCED DECAY HEAT REMOVAL AND CONTAINMENT DESIGN CONCEPT	D. COSTES

ATTACHMENT V

Meeting at Euratom

TENTATIVE AGENDA

Visit of Drs. Marchese, Swanson and Wright (USNRC)
JRC-Ispra, Monday 8 December 1980

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09.00: Reactor Safety (General) - Mrs. Fasoli-Stella

09.30: Sodium Boiling Studies and FCI - Mr. Kottowski

10.30: PAHR In-Pile Tests - Mr. Backs

11.30: Super-Sara - Mr. Randles

12.30: LUNCH

14.00: Development of Models and Codes - Mr. Donea

15.00: PAHR Out-of-Pile Studies, FARO Project - Mr. Hohmann

16.00: Subassembly Thermohydraulics Code Development and Validation - Mr. Nijsing

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