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Safety Evaluation Report

U. S. Nuclear
Regulatory Commission

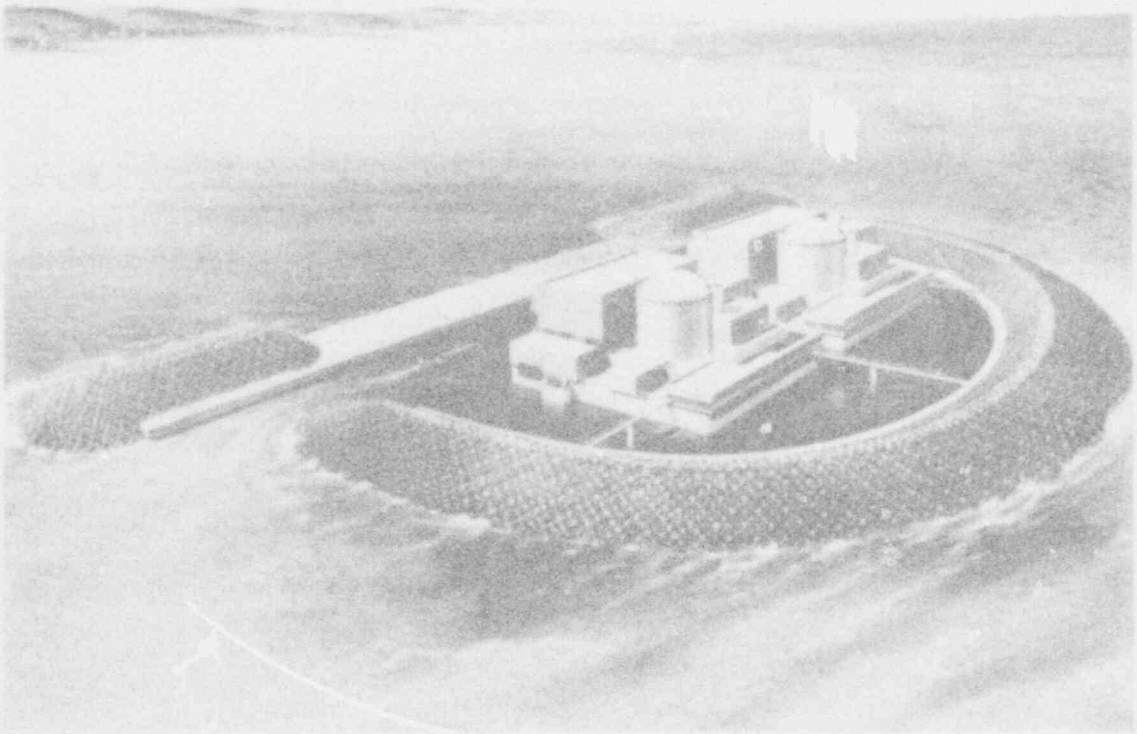
related to

Offshore Power Systems Floating Nuclear Plants (1 - 8)

Office of Nuclear
Reactor Regulation

Docket No. STN 50-437

February 1980



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NUREG-0054
Supplement No. 3
to NUREG-75/100

SUPPLEMENT NO. 3

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

OFFSHORE POWER SYSTEMS

FLOATING NUCLEAR PLANTS (1-8)

DOCKET NO. STN 50-437

February 1980

ABSTRACT

The staff has performed an evaluation of a refractory sacrificial bed, called a core ladle, proposed by Offshore Power Systems (OPS) to be installed in Floating Nuclear Plants (FNPs) to delay the melt-through penetration of molten core debris in the unlikely event of a core meltdown accident. The core ladle design as proposed by OPS is in response to the staff's manufacturing license Condition 4 given in the Final Environmental Statement, Part III, NUREG-0502, dated December 1978 (FES-III). This condition requires that a pad constructed of magnesium oxide, or other equivalent refractory material, be installed in the FNP structure within the lower reactor cavity. The material must provide increased resistance to melt-through by the molten reactor core, must not react with core debris to form a large volume of gases, and must not have any deleterious effects on safety. A heat transfer computer code (MELSAC) was specifically developed to analyze the coupled upward and downward heat transfer from the core debris, including incorporation of the important feedback effect on molten pool heat transfer resulting from the heating of reactor cavity structures above the molten core debris and ladle. Predictions from the MELSAC computer code indicate that a core ladle with walls and floor thickness of 5.25 feet would provide retention of molten core debris within the FNP for a period of time from 2 days to 1 week before melt-through of the FNP barge hull would occur. We conclude that (1) the applicant has satisfactorily met the staff's proposed manufacturing license Condition 4 of FES-III, and (2) the applicant's core ladle design concept is feasible and can be engineered to delay significantly the melt-through penetration of molten core debris through the FNP barge hull. We also conclude that, although recognizing the uncertainties and complexities of fission product behavior during core meltdown events, fission product evaporation, sparging, and subsequent release to both the air and liquid pathways are expected to be less with an MgO core ladle than with concrete.

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

APPENDICES

TABLE OF CONTENTS (Continued)

LIST OF FIGURES

LIST OF FIGURES

LIST OF TABLES

I. INTRODUCTION

The review of Offshore Power Systems (OPS) application to manufacture eight (8) Floating Nuclear Plants (FNP) was initiated in 1973. At the request of the Advisory Committee on Reactor Safeguards (ACRS) and in accordance with the Nuclear Regulatory Commission's (NRC) responsibilities under the National Environmental Protection Act (NEPA) of 1969, the staff conducted a detailed study to assess the comparability of impacts of accidental radioactive releases to the hydrosphere from floating and land-based nuclear power plants. The staff concluded in Reference 1, the Liquid Pathway Generic Study (LPGS), NUREG-0440, dated February 1978, that the risks associated with radioactivity releases to the liquid pathway at an FNP are greater than those at a typical land-based plant (LBP) for core melt accidents. The ACRS provided their comments to the NRC on the LPGS report in May 1978 (see Appendix A).

The staff also found as part of the LPGS that effective interdiction at the site in the event of a core melt accident is not likely for an FNP as designed at that time. The FNP design is somewhat unique in that no basemat is provided below the reactor vessel. For the original FNP design, a thin (4 feet) nonstructural concrete pad was located in the reactor cavity with structural support provided by the steel barge. Any core melt event was likely to result in fairly rapid melt-through of the concrete pad and steel barge with release of activity to both the air and liquid pathways. However, for a LBP, containment of core debris by the soil-rock foundation materials would act to slow the transport of radioactivity. Timely interdiction to confine the release to the immediate site area could reasonably be expected at most LBP sites.

The staff issued Part III to the FNP Final Environmental Statement, NUREG-0502, in December 1978 (Reference 2). Part III of the FES provided a comparison of overall risks from accidental releases of radioactivity for floating and land-based nuclear power plants. The staff concluded in Reference 2 that the FNP manufacturing license be conditioned so as to provide adequate protection of the environment in the event of a core-melt accident. Specifically, with respect to a postulated core-melt accident, Condition 4 (see page XV of NUREG-0502) as reproduced below requires that certain modifications be made to the present FNP design:

> "4. The applicant shall replace the concrete pad beneath the reactor vessel with a pad constructed of magnesium oxide (see Appendix E) or other equivalent refractory material, that will provide increased resistance to melt-through by the molten reactor core in the event of a highly unlikely core-melt accident and which will not react with core-debris to form a large volume of gases. The pad should be as thick as practical, taking into account space availability and applicable design and operating considerations, but not thinner than the concrete pad currently proposed. The proposed refractory material and pad design should not compromise safety requirements and the applicant shall obtain NRC approval of the selected material and pad design prior to the start of construction of major elements of the FNP hull structure."

As part of the Final Environmental Statement, Part III, the staff determined that a delay in the melt-through penetration of the molten core debris would provide time so that steps might be taken to mitigate the radiological consequences of such an event. Also, as part of the FES-III, the staff provided an assessment, which is provided in Appendix B, which establishes that it is feasible to achieve significant delay times through the replacement of the 4-foot thick concrete radiation shield with a refractory sacrificial material, such as magnesium oxide.

In order to satisfy the above manufacturing license condition, OPS has replaced the concrete pad below the reactor vessel with a pad constructed of magnesium oxide (MgO). Their preliminary design and evaluation are provided in Reference 3, OPS Topical Report No. 36A59, "FNP Core Ladle Design and Safety Evaluation," dated April 1979, including Revisions 1 and 2.

The staff and its consultants met with OPS on May 7-8, 1979 to discuss materials and design considerations related to the FNP core ladle. (The MgO replacement pad is referred to by OPS as a "ladle" because it is similar to refractory ladles employed in metal refining operations for handling high temperature molten materials.) The staff's meeting summary is provided in Appendix C.

On June 27, 1979, the ACRS Subcommittee responsible for the review of Offshore Power Systems application to manufacture eight floating nuclear plants, met with the applicant and the staff to review the FNP core ladle design as submitted by the applicant in Topical Report No. 36A59, "FNP Core Ladle Design and Safety Evaluation." Subsequent to the ACRS Subcommittee meeting, the ACRS at its July 1979 meeting issued a letter (see Appendix D) requesting additional information from the applicant and an evaluation of the response by the staff. By letter dated September 14, 1979 (see Appendix E) the applicant provided additional information which the staff has reviewed and evaluated. The staff's review and evaluation of OPS's response to the ACRS letter of July 25, 1979 is presented in Appendix F. On November 17, 1979, another ACRS subcommittee meeting was convened in Los Angeles, California, to review both the applicant and the staff responses to the ACRS request for additional information.

This Safety Evaluation Report (SER) Supplement provides the staff's assessment of the FNP preliminary core ladle design as described in OPS Topical Report No. 36A59. It should be noted that the applicant will be required to obtain NRC approval of the final ladle design prior to construction of major elements of the FNP hull structure.

II. BACKGROUND

The initial core ladle design as described in Reference 3 was critically reviewed by the staff and its consultants during May - June 1979 with regard to satisfying the general requirements and functional criteria set forth in Reference 2. The evaluation concentrated on the heat transfer aspects of the ladle design. In particular, the following separate heat transfer calculations were reviewed:

(a) Molten core penetration into the MgO bed.

(b) Gas generation from concrete around the core ladle due to thermal front propagation.

(c) Effects of thermal radiation from the molten pool surface on the walls of the cavity.

The melt penetration calculations carried out by OPS were hand calculations and were based on the approach taken in Appendix E of Reference 2. The penetration rates presented in Reference 3 were checked and (given the asumptions made by OPS) found to be numerically correct. Differences between the assumptions made by OPS and those made in Reference 2 related primarily to the initial size of the molten pool and to the amount of core decay heat associated with noble gases and volatile fission products. OPS assumed a larger pool than that considered by the staff. Differences between the pool volumes relate primarily to the assumed density of the constitutents. The OPS calculation used a lower mean density resulting in a larger pool. In the context of the hand calculations made in References 2 and 3, the effect on penetration of assuming different pool volumes is minimal. There is a slight reduction in the volumetric heat source in the larger pool resulting in less penetration than would be expected with the same amount of decay heat distributed in a smaller pool. For example, the decay energy required for the larger pool to penetrate a distance of 8 feet would be sufficient for the smaller pool to

penetrate 8.4 feet assuming a similar upward/downward heat split. Also, OPS assumed a smaller fraction of the core decay heat to be associated with noble gases and volatile fission products (20 percent vs 30 percent) resulting in more conservative penetration rates. The models used were recognized by OPS to be only an approximation but were thought to bracket the possible heat flux distribution.

An estimation was made by OPS of heat conduction ahead of the melt front during core penetration into the MgO. It was determined that a computer code TAP-A, Reference 4, with modifications to simulate a moving melt front, could be used in the analysis. The purpose of the calculations was to demonstrate, using a one-dimensional model, that the thermal front propagation ahead of the melt front would be sufficient to outgas the concrete (situated outside the MgO bed) before the melt front reaches the concrete. This would allow the gases released from the concrete to be channeled along the interface between the concrete and the steel tank enclosing the MgO. Most of the gases released from the concrete would therefore be vented directly into the containment atmosphere rather than bubbling through the molten pool. The thermal profiles presented in Reference 3 were checked by integrating the area under the curves and adding the equivalent energy to the energy required to melt the MgO. The resulting energy was compared with the integrated decay power at the corresponding time and the calculations were found to be numerically correct. An interesting result from the calculations was that the TAP-A code predicts that only 5-12 percent of the energy directed into the MgO would be conducted ahead of the melt front. It was concluded that if outgassing of the concrete can be achieved by such a small fraction of the decay energy, then the calculations performed by OPS would appear to represent a conservative estimate. However, the configuration of the core ladle during the May-June 1979 time frame, consisting of 4 feet thick walls and an 8 foot thick base, was such that outgassing of the concrete probably could not have been achieved before the vertical vent path was breached by the melt front. If we assume equal horizontal and vertical penetration, the molten pool would reach the vertical vent space between the core ladle and the concrete with 4 feet of MgO remaining intact under the pool. From an inspection of the thermal profiles presented in Reference 3, it is clear that the thermal front would not have reached the concrete beneath the MgO when the vertical vent path was breached. If the intention is to outgas a large fraction of the concrete beneath the core ladle before arrival of the melt front, and if a similar vent path is to be used, then a core ladle configuration with the same thickness of MgO in the walls as in the base is recommended by the staff.

The effect of thermal radiation from the surface of the molten pool on the reactor cavity side walls was also calculated using the TAP-A computer code. The calculations were performed by assuming that a fraction of the decay heat is directed upwards and an equivalent black body temperature from the molten pool to the reactor cavity wall is calculated. Transient conduction into the reactor cavity wall was then computed using TAP-A given a constant fraction of the decay heat as the upward heat flux. The objective of the OPS calculation was to show that a layer of MgO can be used to protect the concrete side walls from the effect of thermal radiation from the pool. The thermal profiles presented in Reference 3 were checked and given the assumptions made by OPS, found to be numerically correct. It was not, however, clear that the analysis performed by OPS bounded the thermal transient that could occur in this region.

A number of the staff's concerns, namely: core ladle configuation, protection of the upper reactor cavity structure, energy necessary to outgas concrete and the TAP-A computer code regarding the thermal analysis carried out by OPS were resolved at a subsequent meeting held on May 7-8, 1979 (see Appendix C). The configuration of the core ladle had not been finalized and the only constraint on the shape was outside dimensions. The recommendation to make the core ladle walls as thick as the floor was accepted and protection for the upper reactor cavity would be provided if calculations indicated the need for protection in this area. The calculations performed by OPS were recognized to be overly conservative. The directional heat split from the molten pool was calculated independently and parametrically, and the calculations provide an estimate of the worst possible thermal conditions.

It became clear at the meeting that future analytical efforts should attempt to simultaneously solve the upward and downward heat transfer from the pool, including the feedback effect of heating up the structures in the reactor cavity.

At a presentation of the core ladle design to a subcommittee of the ACRS on June 27, 1979, the OPS thermal evaluation was again considered to be too conservative. As a result of the subcommittee meeting and a subsequent full committee meeting in July 1979, it was decided that further information was needed by the ACRS before the review of the FNP could proceed. The information required by the ACRS was transmitted to the NRC (see Appendix D). Three of the ACRS concerns relate to the thermal analysis of the core ladle and are repeated below:

a.1 Calculate the fraction of decay heat radiated from the pool for the proposed design.

a.2 Calculate the effects of heat radiation in Item a.1 on the rate of:

(a) Disintegration and collapse of exposed concrete;

(b) Disintegration and collapse or melting of concrete behind the 6 inch magnesite brick wall;

(c) Collapse of steel from the reactor cavity.

a.7 Discuss the possibility of the heat flux being higher on the sides of the molten mass than the botton (FRG conclusion for concrete melt) with melt going horizontally faster than vertically.

A meeting was arranged between OPS and the NRC and their consultant on August 15, 1979 to discuss the ACRS request. Subsequent to the meeting, OPS provided responses (see Appendix E) to the ACRS matters. The specific OPS responses to Questions a.1, a.2 and a.7 were reviewed by the staff and its consultants, and our evaluation was included in the transmittal to the ACRS on November 2, 1979 (see Appendix F). The response provided to the ACRS were based on the analysis described in this safety evaluation report and the conclusion arrived at remain unchanged.

As previously mentioned, the need for an integrated mathematical model became obvious during May 1979. However, existing melt front models, References 5 and 6, are not directly applicable to the FNP configuration. In its original form, the GROWS code, Reference 5, assumes an overlying pool of sodium. A new version (GROWS 2) of GROWS was presented at a recent meeting held at Argonne National Laboratory (ANL) on July 10, 1979. The new code differs substantially from the original GROWS code and heat transfer by thermal radiation from the molten pool surface to a constant sink temperature is modeled. Unfortunately, the important feedback effect resulting from the heating of the structures around the pool is not modeled and GROWS would require modification to allow for this effect. The INTER code, Reference 6, was developed at Sandia primarily to formulate a mathematical model of the molten core/concrete interactions observed in the Sandia experimental program. The code was intended to model experiments in which melts were poured into concrete crucibles and large quantities of gases were observed to be released from the concrete. Particular attention is, therefore, given in INTER to the effect of gas release on molten pool heat transfer. The effect of heating the structures above the pool is, again, not modeled in the code. No gas generation has ever been reported during the thermal erosion of MgO (except for minor amounts of impurities) and thus the INTER code would require extensive modifications before the code could be applied to the FNP core ladle and reactor cavity configuration. The effect of heating the structures above the pool would also have to be introduced into INTER. It was therefore necessary to develop a model more applicable to the FNP configuration and the result is the MELSAC computer code.

A full description of the thermal analysis of the core ladle carried out by staff consultants at the Brookhaven National Laboratory using the MELSAC computer code is provided in a separate report, Reference 7. In this safety evaluation report, we will include our latest thermal evaluation of the core ladle. As the design of the core ladle and surrounding structures has been under development, significant changes have been made since the OPS topical report was published in April 1979. We have included a description of the latest proposed core ladle design, as presented to the ACRS subcommittee on November 17, 1979 in Los Angeles, California.

III. CORE LADLE FUNCTIONAL CRITERIA

Considering Condition 4 of FES-III, the NRC established and the applicant accepted the following general functional design criteria for the FNP core ladle:

1. It shall provide increased resistance to core debris melt-through.

2. It shall not react with core debris to form a large volume of gases.

3. It shall be at least as thick as the current concrete pad (4 feet) and as thick as practicable.

4. It shall not compromise safety.

The staff concludes that these general functional design criteria provide an acceptable basis upon which to proceed to the development of a final design for the FNP core ladle.

IV. CORE LADLE DESIGN DESCRIPTION

The original core ladle proposed in Reference 3 was provided with internal dimensions of 27 feet long and 17.5 feet wide and about 2 feet deep which was sufficient to hold an initial core volume of 980 cubic feet. There was approximately 4 feet thickness of MgO in the walls and 8 feet in the base. Subsequent to discussions with the staff and concerns that additional steel from the reactor vessel may melt into the core ladle (after the initial melt through), the ladle dimensions were altered to those shown in Figures 1 and 2. The new design reduces the base thickness from 8 feet to 5.25 feet but increases the wall thickness from 4 feet to 5.25 feet. The internal depth of the ladle has been increased to about 10 feet making its initial volumetric capacity equal to 4000 cubic feet. This volume should contain most of the materials from the entire reactor vessel in a molten state.

Further modifications relate to upper reactor cavity wall protection. Initially only a single layer of MgO bricks was proposed; however, subsequent calculations indicated the need for greater protection. In the latest design, the walls consist (up to the 91.5-foot elevation of 2 feet of MgO, 0.25 feet air gap, and 1.75 feet of concrete on the inside of bulkheads "G" and "H" and the pressure bulkheads. Additional concrete is located outside this boundary for shielding purposes only, and preliminary calculations indicate that the insulation provided by such concrete will result in higher steel bulkhead temperatures. The intention is, therefore, to move the shielding concrete away from the outside surface of the bulkheads. Thinner protection in the upper regions of the reactor cavity is necessary due to space limitations. A number of possible combinations of insulating materials have been suggested for these regions using ZrO_2 or refractory concrete, but a final configuration has not yet been established.

The core ladle design will take advantage of many years of experience that exists with the design and operation of crucibles and hearths used in large furnaces of the steel industry for containing molten materials

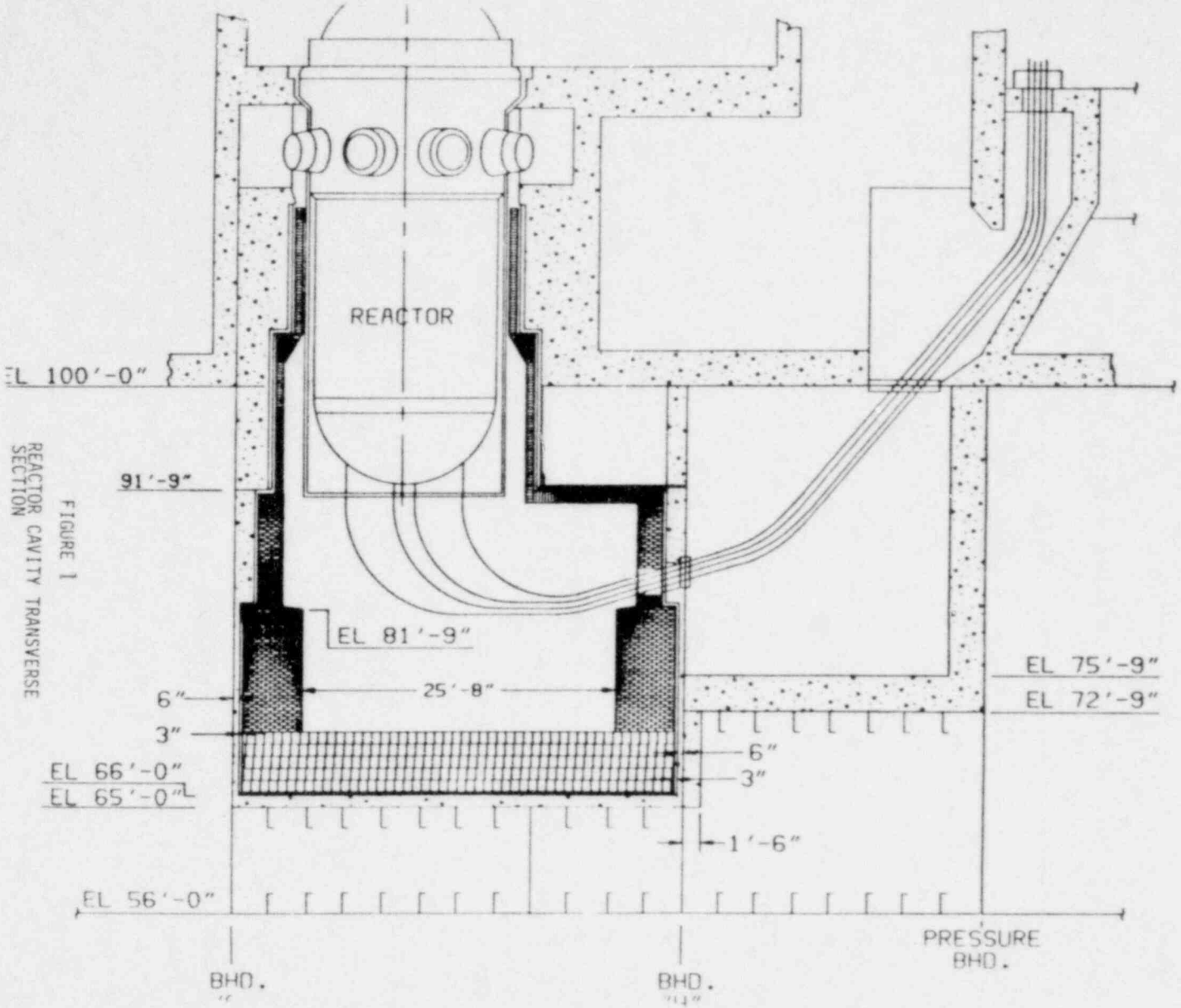


FIGURE 1
REACTOR CAVITY TRANSVERSE SECTION

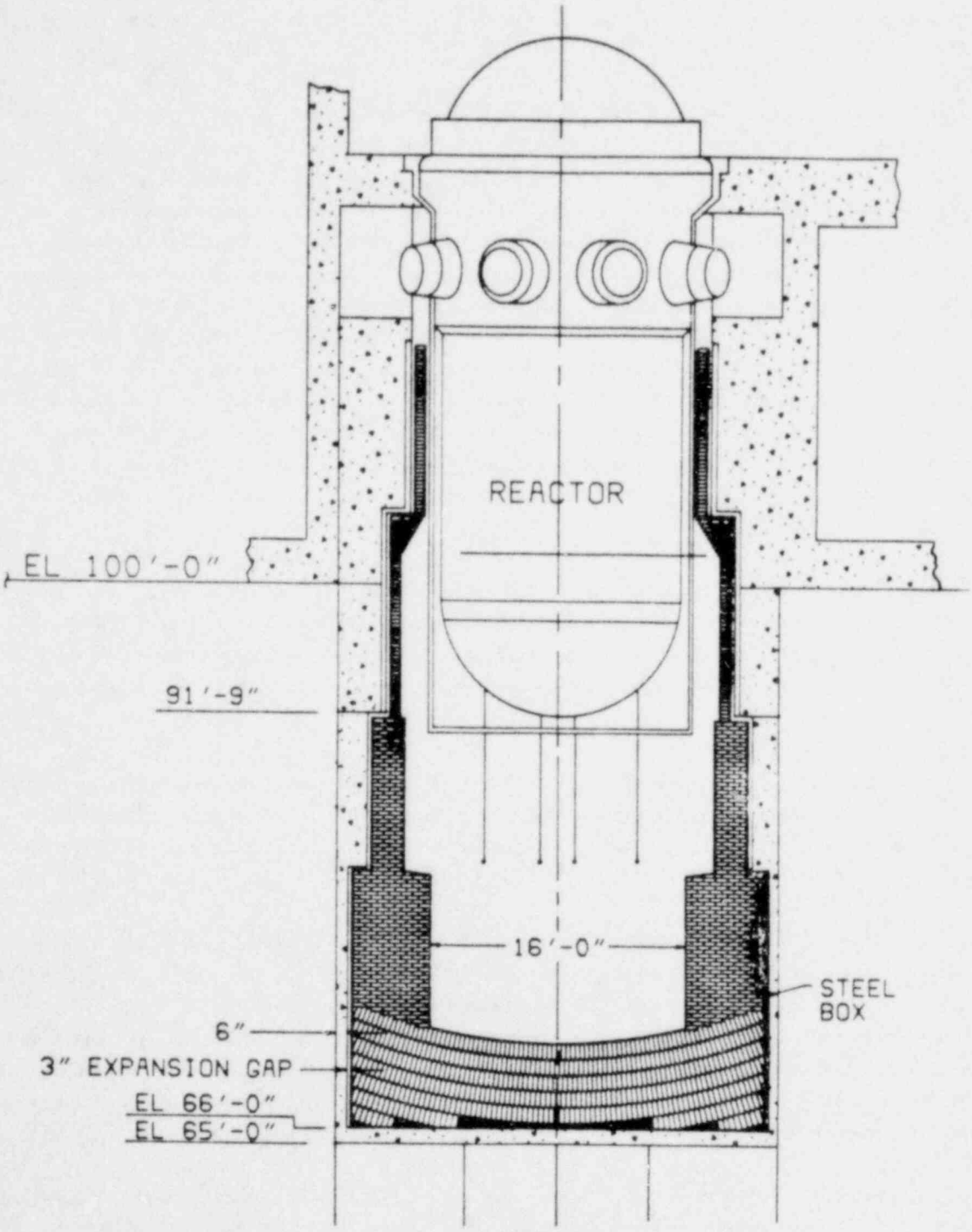


FIGURE 2
REACTOR CAVITY LONGITUDINAL SECTION

(iron, steel, nickel) under severe service conditions for long periods of time during refining processes. The ladle will utilize burned high purity magnesite bricks, such as HARKLASE, which will be arranged in a cylindrical arch configuration with a staggered bricking pattern in both the transverse and longitudinal directions. As shown in Figures 3, 4 and 5, the bricks form rows of inverted arches with double interlocking tongue and groove joints. Each course of bricks will be offset and staggered from joints of the courses immediately above and below it to provide a tortuous flow path for molten core debris penetration. No mortar is used between the bricks. A dry magnesite ramming mix may be used as a leveling material at the ladle shell-MgO brick interface to fill any voids. In addition, an inert ceramic material can be used as filler material to accommodate thermal expansion at the interface between the refractory material and the steel shell. MgO mortar may be used on the sidewalls above the ladle.

The bricks in the interior regions of the ladle are laid in place as close together as possible (approximately 1/32 inch). When the bricks heat up, they will expand, soften and fuse together without crushing forming a monolythic structure. Most of the thermal expansion is taken up at the outer extremities of the core ladle and not between bricks. The thermal expansion of the cylindrical arch configuration results in forces and movement in the downward and lateral directions, but not upward; therefore, this configuration along with the double tongue and grooves should be very effective in preventing potential brick float-up caused by higher density core melt debris. In addition, the cylindrical arch configuration results in the bricks always being in compression on heatup, so that cracking or shearing will not result in bricks coming loose. The creep of MgO is significant at higher temperatures, and this tends to fix thermal expansion considerations to a temperature of about 2600 degrees Fahrenheit. Therefore, the bricks tend to relieve themselves at higher temperatures without crushing. The top-most layer of the ladle will probably be a row of chemically-bonded magnesite bricks (see Table C-1, OPS Topical Report No. 36A59, April 1979, for property list) which has high resistance to thermal shock and physical impact.

It should be emphasized that additional safety factors result from (1) the formation of a lower melting point eutectic between the core melt debris and MgO, and (2) dilution of core melt with MgO will lower the melt temperature. Both of these effects will result in making the underlying layers of MgO more refractory compared to the melt material, since as time passes the MgO melt temperature becomes greater compared to the core melt/MgO system temperature. The applicant has committed to provide additional design information to the staff as it is developed.

Completely enclosing the MgO pad is a watertight steel liner plate. This steel enclosure acts as a sump liner during normal operation of the plant. The primary function of the steel enclosure is to prevent water that may enter the lower reactor cavity during operation, refueling or as a result of leakages, from coming into contact with the MgO bricks.

The region directly below and on the sides of the core ladle steel shell is provided with 12 inches and 6 inches, respectively, of concrete. The concrete provides radiation shielding, thermal insulation, and corrosion protection of the steel bulkheads and floor plate of the containment. For those areas of the reactor cavity which are exposed to high temperatures, either basaltic type concrete or some other equivalent will be used to minimize gas generation. Limestone type concrete will not be used in the reactor cavity. A 3-inch gap is provided between the core ladle steel shell and the surrounding concrete. This gap is used to accommodate thermal expansion of the core ladle and provides a passage for venting gas and vapor from heated concrete.

The purpose of any protection that may be used in the reactor cavity is to prevent the non-structural concrete from decomposition and thermal damage of any structural steel. A scoping study using the MELSAC code is presented later in subsection V.B.2.b in which various input parameters are varied including the configuration of the reactor

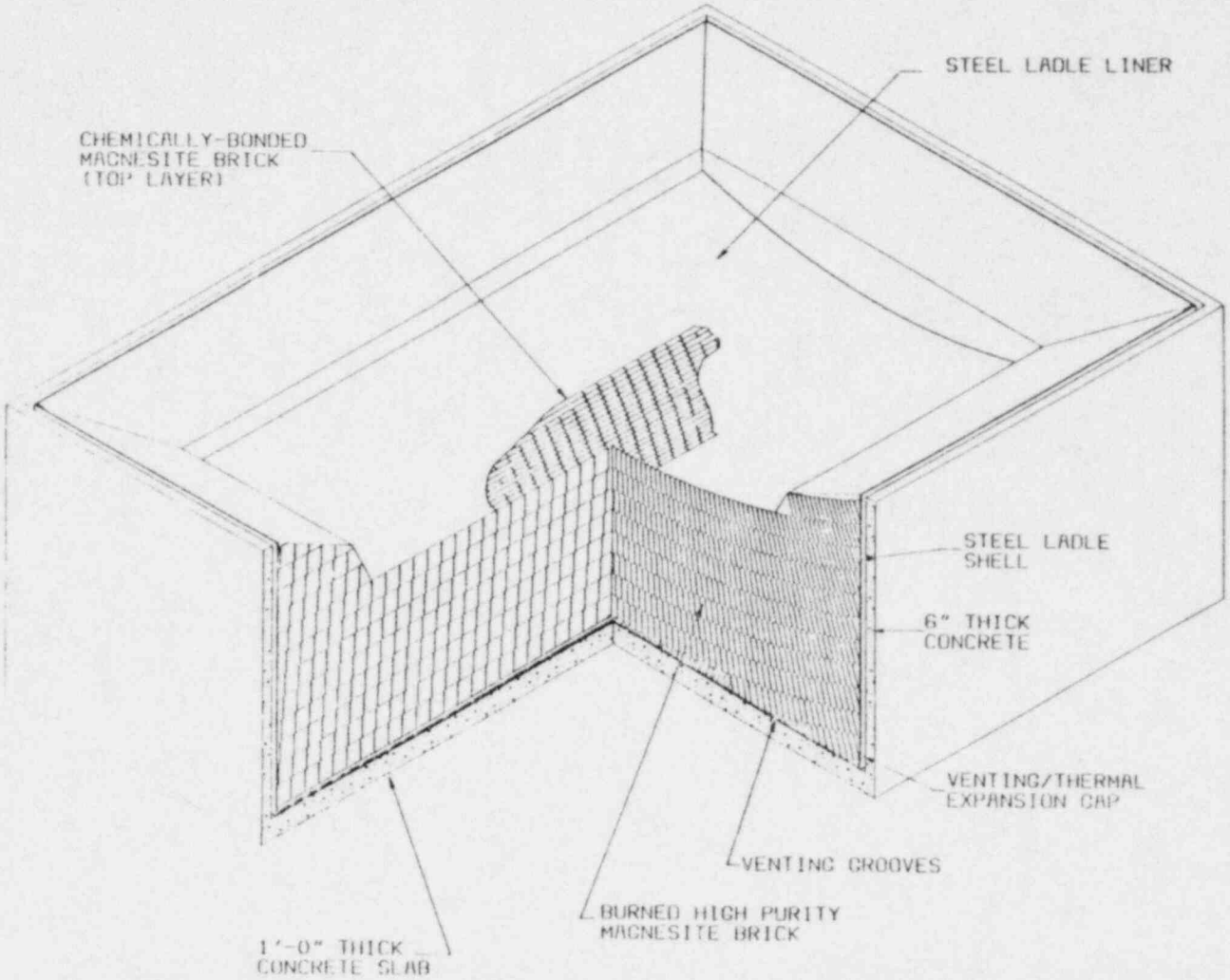


FIGURE 3
ISOMETRIC VIEW OF CORE LADLE

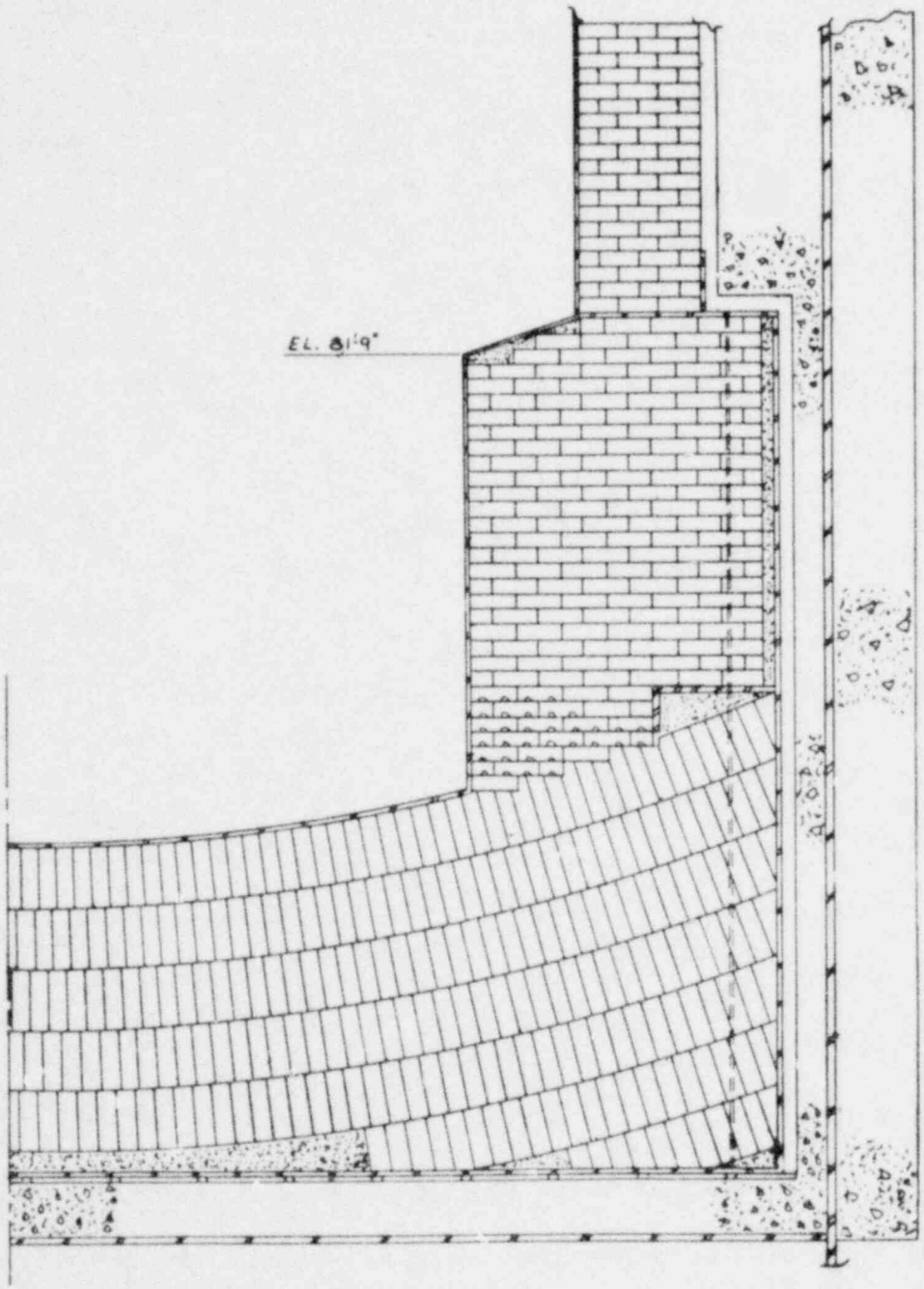


FIGURE 4
CROSS SECTION OF CORE LADLE

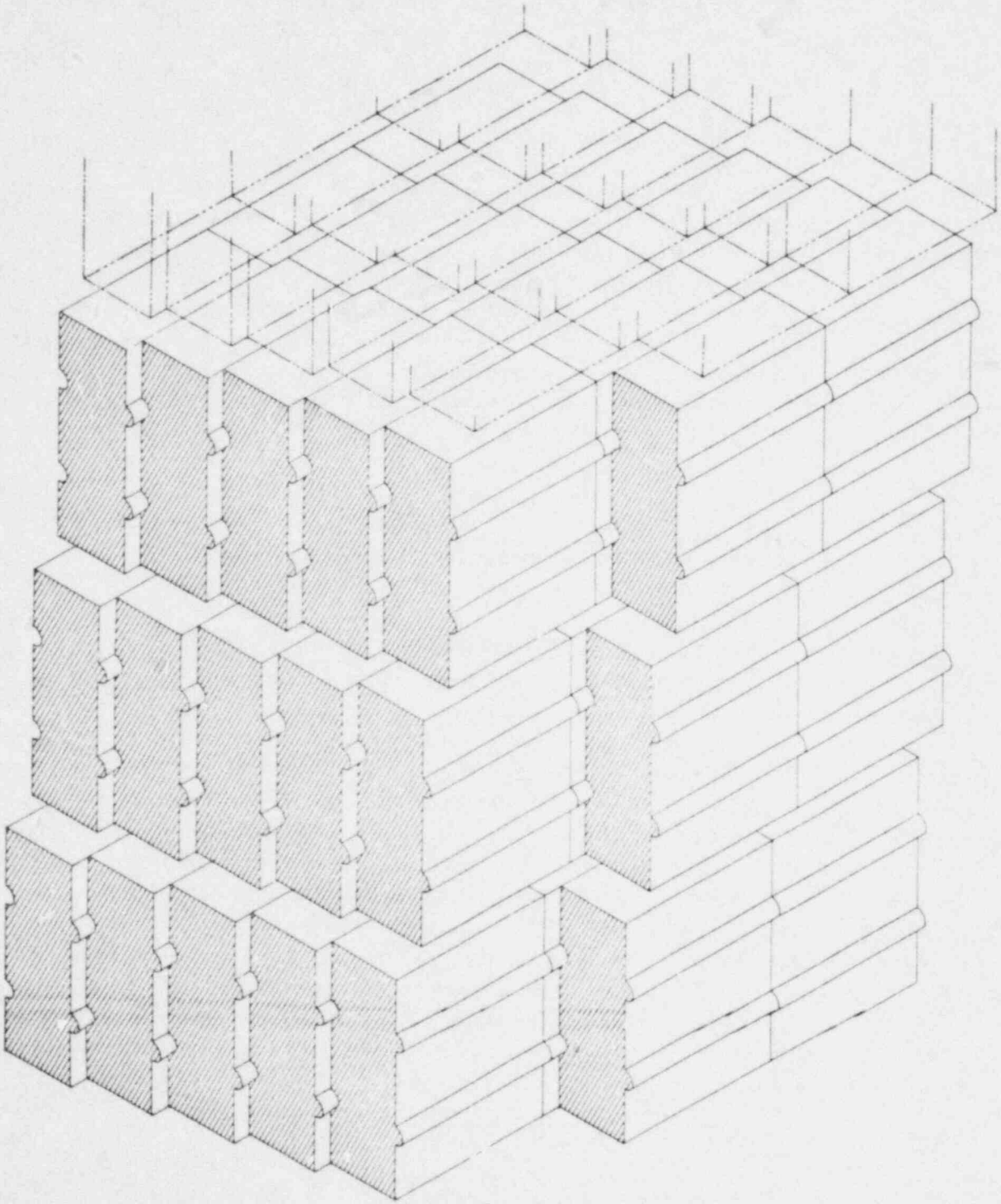


FIGURE 5
CORE LADLE DOUBLE INTERLOCKING TONGUE AND GROOVE JOINTS AND STAGGERED BRICKING ARRANGEMENT

cavity. Based on this scoping study, a run was then made (Case A) using the latest proposed core ladle and reactor cavity configuration and the results of the calculation are presented in Subsection V.B.2.c. The cavity wall is modeled as a slab of MgO, 2 feet thick, against a slab of 2 feet thick concrete with natural convection from the steel bulkheads behind the concrete.

A number of runs were carried out in which the thickness of the MgO and concrete were varied in order to determine a configuration that would maintain the concrete below its decomposition temperature. The results of the analysis are also presented in Subsection V.B.2.c. The way in which the core ladle and reactor cavity are modeled in MELSAC is discussed in detail in the following section.

V. EVALUATION OF CORE LADLE

This section provides the staff's assessment of the FNP core ladle preliminary design, including consideration of materials, thermal performance, structural adequacy, shielding, effect on containment pressure response, radiological aspects, and related steel-industry experience. Although there was a concentrated effort in reviewing and evaluating OPS Topical Report No. 36A59 which provides the applicant's analyses in the above areas, the staff also utilized its own independent sources of information, calculational tools and expert consultants in the field. Since the current manufacturing license review is analogous to a Construction Permit (CP) review, the staff's evaluation reported herein focused on whether it is feasible to incorporate a core ladle into the Floating Nuclear Plant that will substantially increase the melt-through delay time in the event of a core melt accident. Besides evaluating the core debris hold-up time afforded by the ladle, the staff also performed analyses in the above areas to ensure that the core ladle has no deleterious effects on safety. As part of the amended manufacturing license review which is analogous to an Operating License (OL) review, the applicant will be required to provide a final core ladle design report and obtain NRC approval of the selected refractory sacrificial material and pad design prior to the start of construction of major elements of the FNP hull structure.

V.A Materials Considerations

In Appendix E of Reference 2 (FES-III, NUREG-0502, dated December 1978) which is attached as Appendix B to this report, the staff evaluated various refractory sacrificial materials that appeared promising in terms of delaying melt-through of the FNP barge structure by a molten core. The staff concluded that magnesium oxide (MgO) appears to be the best of the candidate materials. This stems primarily from considerations related to high melting point (2850 degrees Centigrade), high specific heat (0.31 calories per gram per degree Centigrade), miscibility with UO_2, stability with respect to molten UO_2 and steel, essentially no gas generation, easy to fabricate into bricks, readily available, relatively low cost, existence of applicable experimental data, and considerable experience with its use in steel-making operations. We also indicated in FES-III that it is possible that some lesser known materials might provide more resistance to melt-through, but their identification would probably require an extensive research and development program, and it is not obvious at this time that any other material would offer any significant advantages. The staff did point out, however, that although there is reasonable assurance that a material can be selected that will substantially increase the melt-through time, there are uncertainties related to the physical, chemical and mechanical behavior of all of these materials at the extreme temperatures existing during a core meltdown event, and additional research is required before the staff could determine the suitability of a particular material. A discussion of the research and development needs is provided in Section VI.

The applicant has examined a number of promising candidate materials for delaying core melt-through and has selected MgO as the refractory sacrificial material for the FNP core ladle.

Until further research and development is performed, the staff concludes that the best choice of a refractory material for the FNP core ladle is MgO.

V.B Thermal Performance

The staff provided a preliminary thermal analysis in Appendix E of Reference 2, FES-III, NUREG-0502, dated December 1978, which is attached as Appendix B. This analysis was performed to estimate the time for a molten pool of core debris to penetrate a refractory sacrificial bed composed of MgO. The calculations were performed to indicate the holdup time that could be afforded by installing a sacrificial MgO bed to cope with a core meltdown accident in an FNP.

Since the publication of FES-III, NUREG-0502, in December 1978, both the staff and the applicant have performed more detailed thermal analyses of the FNP core ladle. The staff has reviewed the applicant's heat transfer analyses and has performed its own separate independent calculations. In this section, we will present our thermal performance evaluation, including a description of the manner in which the core ladle and reactor cavity are modeled in the MELSAC computer code, the scope of MELSAC, a discussion on molten pool heat transfer (which is pertinent to Question a.7 of the ACRS letter of July 25, 1979), the results obtained from MELSAC, including a comparison with the applicant's results, and finally our conclusions on the thermal performance of the FNP core ladle.

V.B.1 Computational Methods and Assumptions

In Section IV, the configuration of the reactor cavity and core ladle were described. Following a postulated melt-through of the reactor vessel, a number of core materials would enter the core ladle. The composition of the molten pool can vary over a substantial range. The composition in the range used for the thermal calculations is shown in Table 1. The masses presented in Table 1 were obtained by assuming 25 percent of the steel in the lower reactor vessel head and internals to accompany all of the steel and fuel in the core region into the ladle. All of the cladding (Zircalloy-4) and silver in the core regions were also added to the pool.

After the molten core enters the ladle, heat and mass transfer processes, as illustrated in Figure 6, can occur. Heat transfer from the pool would primarily occur via radiation from the top surface to the upper cavity structures and by melting and conduction from the lower surface into the core ladle. Mass addition to the pool would result from the melting of the core ladle and possibly from the reactor vessel and associated structures in the upper cavity. It was noted in the introduction that an analytical model is not at present available to analyze this particular heat and mass transfer problem. The MELSAC code, which is described in Subsection V.B.1.a is an attempt to incorporate the most important features of the problem within a reasonably simple framework. Heat transfer within the molten pool is a complex phenomenon and the subject of a great deal of current research. Therefore, a discussion on molten pool heat transfer (and in particular, on the correlations used in MELSAC) is included in Subsection V.B.1.b. Finally, the physical properties used in the MELSAC code are listed in Subsection V.B.1.c.

V.B.1.a MELSAC Computer Code

The heat and mass transfer problem illustrated in Figure 6 is modeled in MELSAC as shown in Figure 7. The molten pool is assumed to be initially pure UO_2. As the melt front moves into the core ladle, the code computes the dilution of the UO_2 with molten MgO. Conduction ahead of the melt front into the core ladle is modeled. The possibility of a crust forming on the exposed upper surface of the molten pool is also considered. Heat transfer from the pool surface is by thermal radiation to the reactor vessel and the cavity wall. The cavity wall is represented by a slab and one-dimensional heat conduction is modeled. The reactor vessel is modeled as

TABLE 1

PROPOSED REACTOR CORE AND MOLTEN POOL CONSTITUENTS

REACTOR CORE CONSTITUENTS	MASS(LB)
UO_2	222739
Zr	50913
Steel	94000
Ag	5000
TOTAL	372652

MOLTEN POOL CONSTITUENTS	MASS (LB)
UO_2	200059
Steel	84600
UO_2 } Eutectic*	22680
Fe_3O_4 } Eutectic*	12992
ZrO_2	68773
Ag	5000
TOTAL	394104

* Assume 10% of Iron is Oxidized.

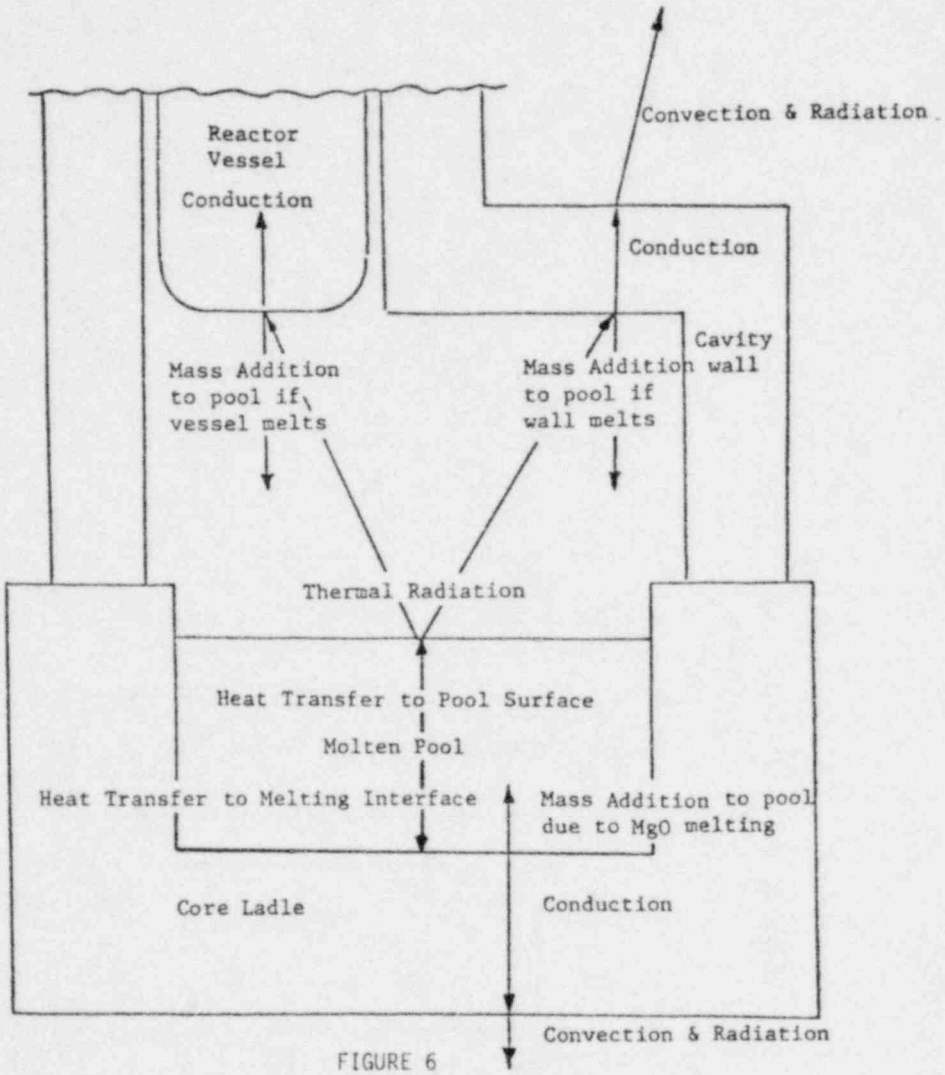
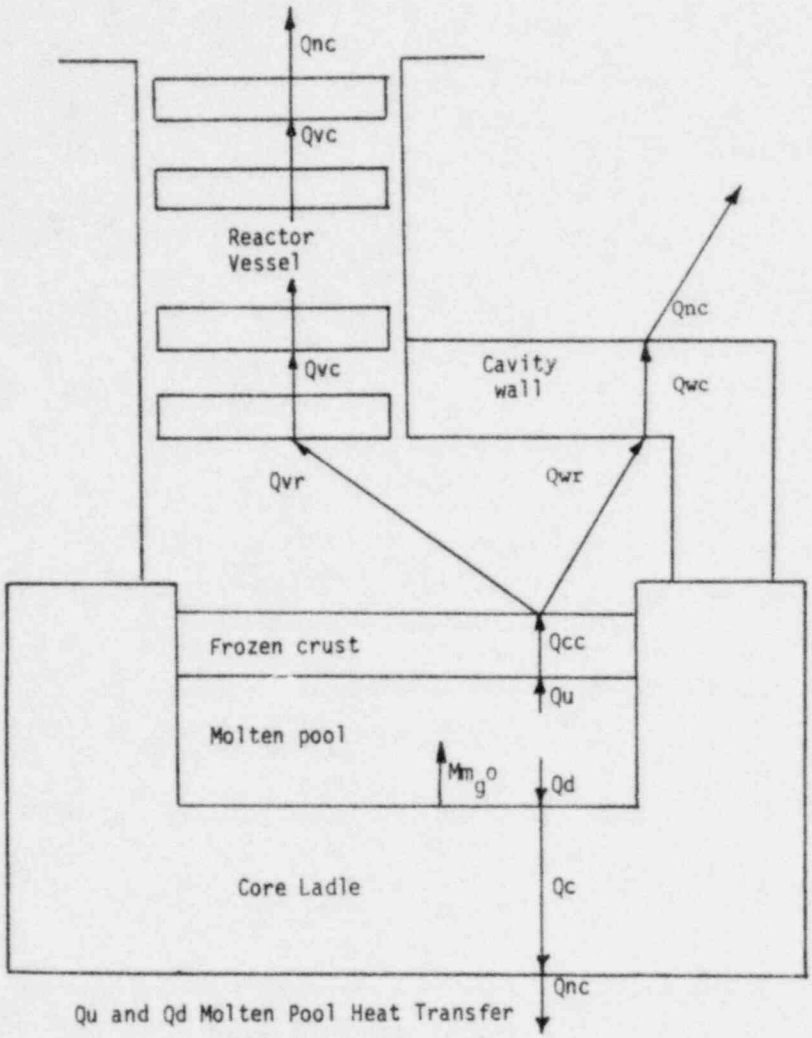


FIGURE 6

Heat and Mass Transfer in Reactor Cavity



Qu and Qd Molten Pool Heat Transfer

Qc Conduction in core ladle

Qcc Conduction through frozen Crust

Qvr and Qwr Thermal Radiation to vessel and cavity wall

Qwc Conduction through cavity wall

Qvc Conduction in reactor vessel

Qnc Natural convection from behind structures.

$Mm_g o$ Mass addition of molten $m_g o$ from meltfront.

FIGURE 7
Model used in MELSAC

a series of connected masses. Heat transfer between each of the vessel masses is by conduction. Heat loss from the back of the reactor vessel, outside of the cavity wall and core ladle, is modeled as natural convection heat transfer.

At present, the only mass addition to the pool is from the melt front in the core ladle. The energy necessary to melt the reactor vessel is computed, but the addition of any molten steel that may fall from the vessel into the molten pool is not modeled at present. The effect of steel and Zircalloy addition to the pool is discussed in more detail in Section V.B.2.

The computational procedure adopted in the MELSAC code is described in detail in a separate report, Reference 7. Very briefly, at the beginning of a time step, MELSAC computes the directional heat transfer from the bulk pool temperature to the boundaries of the pool using established molten pool heat transfer correlations. Upward pool heat transfer is compared with the heat lost from the pool surface by radiation to determine the possible growth pattern of a crust. Heat transfer from the crust or pool surface determines the temperature rise and possible melting of structures in the reactor cavity. Downward pool heat transfer results in melting of MgO and conduction heat transfer ahead of the melt front. Given these heat losses and mass addition to the pool, an energy balance is carried out to determine a new bulk pool temperature at the end of the time step. The procedure is repeated for subsequent time steps.

V.B.1.b Molten Pool Heat Transfer

The ACRS suggested (Question a.7, Appendix D) that the heat flux on the sides of the molten mass may possibly be higher than that on the bottom, with melting going horizontally faster than vertically. This would impact on estimations of molten pool penetration. The ACRS supported the above possibility by reference to an experimental program in the Federal Republic of Germany (FRG) of molten core/concrete interactions. However, the irregular erosion pattern observed during penetration of concrete is the result of gas generation from the decomposing concrete. It is noted in the introduction that no gas evolution has been observed during the erosion of MgO so that the concrete data is not applicable to molten core/MgO interactions.

There is a lack of experimental data on molten core/MgO interactions. However, a number of simulant experiments have been carried out to determine the heat transfer processes that are operating within melting systems that could possibly be applied to the molten core/MgO configuration. The earlier correlations, Reference 8, developed for pool heat transfer were based on a Rayleigh number formulation, which uses the difference between the bulk pool temperature and the melting interface temperature as the driving force. However, more recent experiments performed under the direction of L. Baker at the Argonne National Laboratory (ANL) and I. Catton at the University of California, Los Angeles (UCLA), have indicated that the earlier correlations are not applicable to the specific case of molten core penetrating MgO. In this case the pool is more dense than the molten MgO at the melt interface and simulant experiments indicate that there is considerable buoyancy-driven motion induced under these circumstances. The experiments at UCLA and ANL indicate that the Rayleigh number formulation should be based on the density difference between the pool and the melting material. The density-driven correlations, Reference 9, result in much higher heat transfer coefficients than those of Reference 8.

Applying these correlations to prototypic conditions was done through the GROWS code. The original version of the GROWS code used the temperature-driven formulations and the heat transfer correlations which tended to favor lateral penetration in preference to downward penetration. A later version of GROWS (GROWS 2) uses the density-driven formulations. When there are large differences in density between the pool and the molten MgO, the code predicts downward penetration rates much faster than the lateral penetration rates. However, as the pool is diluted with molten MgO, the density difference between the pool and the molten MgO is obviously reduced

and the density-driven heat transfer correlations are reduced to the original temperature-driven formulations. The net result is that for motlen core penetration of MgO over a period of several days, the lateral and downward penetrations are approximately the same.

For the above reasons we have made the lateral heat transfer coefficient equal to the downward heat transfer coefficient in MELSAC. We only distinguish between the effects of density-driven heat transfer and the earlier convective formulations using a temperature difference driven Rayleigh number. There is clearly not any experimental evidence yet available that would suggest changing the heat transfer formulations in MELSAC. If such evidence becomes available, it could easily be incorporated into MELSAC for the final design evaluations.

V.B.1.c Physical Properties

In the present version of MELSAC, the physical properties (density, specific heat, thermal conductivity, expansion coefficient and viscosity) of the constituents are not temperature dependent unless there is a change of state. The property values used in MELSAC are therefore mean values, which best represent the physical properties over the temperature range under consideration. The values selected for the physical properties of MgO and UO_2 in both the liquid and solid phases are included in Tables 2 and 3, respectively. The properties of concrete in Table 4 are given only for the solid phase because melting of concrete is not considered in the analysis.

As the molten UO_2 melts the MgO, the pool becomes diluted with molten MgO. It is assumed for simplicity that when a quantity of MgO is melted in a given time step that it is immediately and uniformly distributed within the molten pool at the end of the time step. The bulk physical properties of the homogeneous mixture of UO_2 and MgO are required to determine the pool heat transfer correlations. The properties of the pure substances (Tables 2 and 3) therefore have to be combined in some logical manner to provide an estimate of the mixture properties. The specific heat, density and viscosity of the pure substances are combined using the pool mass fraction. The thermal conductivity is combined through the pool volume fraction. The melting point of the $MgO-UO_2$ mixture is assumed to follow the idealized system shown in Figure 8.

V.B.2 Heat Transfer Results

V.B.2.a Introduction

A particular advantage of using MELSAC is that it allows scoping studies to be carried out in which the sensitivity of the response of the system to a number of input parameters can easily be determined. The first part of this section is devoted to such a scoping study. The sensivity of the penetration of the core ladle and the heating up of the structures in the reactor cavity to six input parameters (namely: pool heat transfer correlations, MgO melting temperature, heat transfer from back of structures, reactor vessel heating, heat transfer from pool surface to structures, and cavity wall configuration) are examined. Based on judgments with regard to the physical configuration of the reactor cavity and core ladle and related phenomena, it is believed that the sixteen cases described in Table 5 were sufficient to determine the sensitivity of the system. The input parameters to MELSAC, as well as the important results obtained from the code, are included in Table 5. The scoping study is discussed in detail in subsection V.B.2.b.

Based on the scoping study, a set of input parameters were established, which best represent the heat and mass transfer under consideration, given the present limitations of the MELSAC code. These parameters, together with the latest core ladle and reactor cavity configuration, were used to produce a series of best estimate cases (Cases A through C). Cases A through C are discussed in detail in subsection V.B.2.c.

TABLE 2

PHYSICAL PROPERTIES OF MgO

PROPERTY	VALUE
Melting Point	3073 K (5072°F)
Boiling Point	3873 K (6512°F)
Heat of Fusion	1.9×10^6 J/Kg (817 Btu/lb)
LIQUID PHASE:	
Density	2700.0 Kg/m^3 (169 lb/ft^3)
Specific Heat	1507.0 J/Kg K (0.36 Btu/lb °F)
Thermal Conductivity	17.2 W/m K (9.9 Btu/ft hr °F)
Expansion Coefficient	10^{-4} K^{-1} (1.8×10^{-4} °F^{-1})
Viscosity	0.001 N s/m^2 (0.0007 lb/ft s)
SOLID PHASE:	
Density	3027.5 Kg/m^3 (189. lb/ft^3)
Specific Heat	1300.0 J/Kg K (0.31 Btu/lb °F)
Thermal Conductivity	11.8 W/m K (6.8 Btu/ft hr °F)
Thermal Diffusivity	3×10^{-6} m^2/s (32.3×10^{-6} ft^2/s)

TABLE 3

PHYSICAL PROPERTIES OF UO_2

PROPERTY	VALUE
Melting Point	3123 K (5160°F)
Boiling Point	3700 K (6200°F)
Heat of Fusion	2.8×10^5 J/Kg (120 Btu/lb)
LIQUID PHASE:	
Density	8700.0 Kg/m^3 (543 lb/ft^3)
Specific Heat	502.0 J/Kg K (0.12 Btu/lb °F)
Thermal Conductivity	3.5 W/m K (2 Btu/ft hr °F)
Expansion Coefficient	10^{-4} K^{-1} (1.8×10^{-4} °F^{-1})
Viscosity	0.005 N s/m^2 (0.0003 lb/ft s)
SOLID PHASE:	
Density	10000.0 Kg/m^3 (624 lb/ft^3)
Thermal Conductivity	2.93 W/m K (1.7 Btu/ft hr °F)

TABLE 4

PHYSICAL PROPERTIES OF CONCRETE

PROPERTY	VALUE
Melting Point	1473 K (2190°F)
Boiling Point	2573 K (3990°F)
Heat of Fusion	2.9×10^5 J/Kg (125 Btu/lb)
SOLID PHASE:	
Density	2600.0 Kg/m^3 (162 lb/ft^3)
Specific Heat	1047.0 J/Kg K (0.25 Btu/lb °F)
Thermal Conductivity	2.1 W/m K (1.2 Btu/ft hr °F)
Thermal Diffusivity	7.7×10^{-7} m^2/s (8.3×10^{-6} ft^2/s)

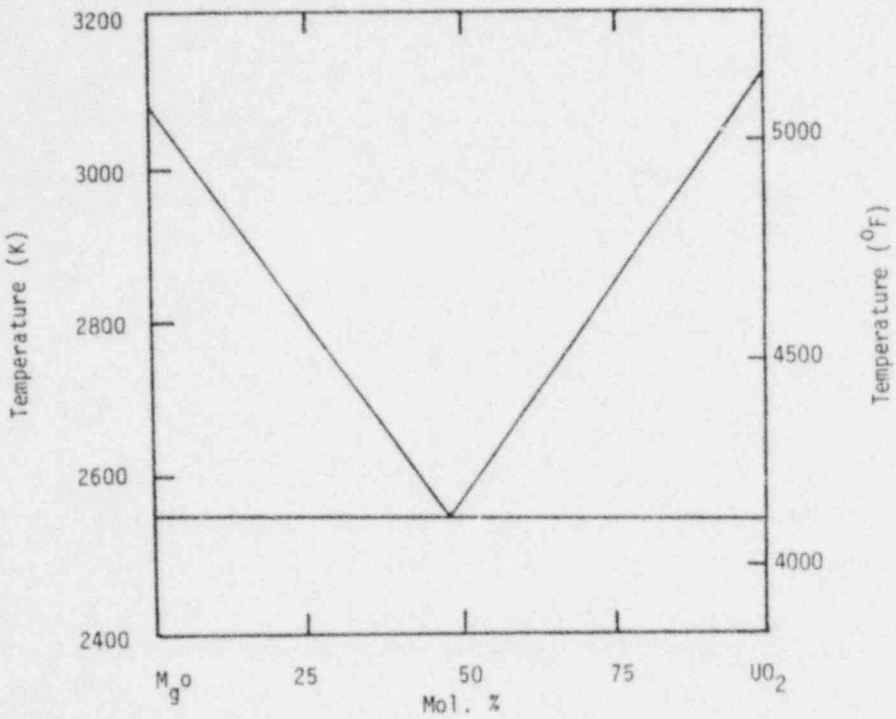


FIGURE 8

Idealized $MgO-UO_2$ System

(Reproduced from Ref. 10)

TABLE 5

SCOPING STUDY USING MELSAC

CASE	INPUT PARAMETERS							OUTPUT DATA FROM MELSAC				
									REACTOR VESSEL		CAVITY WALL	
	CAVITY WALL MgO THICKNESS (FT)	CAVITY WALL CONCRETE THICKNESS (FT)	REACTOR VESSEL MODEL	POOL SURFACE TO STRUCTURES HEAT TRANSFER	MOLTEN POOL DOWNWARD HEAT TRANSFER	STRUCTURES BACK SIDE HEAT TRANSFER	MgO MELTING TEMP. (°F)	TIME TO PENETRATE CORE LADLE (DAYS)	TIME TO START VESSEL MELT. (HR)	TIME TO COMPLETELY MELT VESSEL (DAYS)	TIME TO START OF REFRACTORY MELT. (DAYS)	TIME TO START OF CONCRETE MELT. (DAYS)
1	1.0	1.0	YES	N.B.B.	D.D.	N.C.	5072	4.75	2.60	1.67	-	0.39
2	0.5	1.5	YES	N.B.B.	D.D.	N.C.	5072	4.46	2.60	1.54	-	0.15
3	1.5	0.5	YES	N.B.B.	D.D.	N.C.	5072	5.08	2.60	1.75	-	0.69
4	2.0	0.0	YES	N.B.B.	D.D.	N.C.	5072	5.50	2.60	1.75	-	0.86
5	1.0	1.0	YES	B.B.	D.D.	N.C.	5072	4.83	2.24	1.29	-	0.42
6	1.0	1.0	NO	N.B.B.	D.D.	A	5072	3.87	-	-	-	0.37
7	0.5	1.5	NO	N.B.B.	D.D.	A	5072	3.71	-	-	-	0.15
8	1.5	0.5	NO	N.B.B.	D.D.	A	5072	3.92	-	-	-	0.65
9	2.0	0.0	NO	N.B.B.	D.D.	A	5072	3.88	-	-	-	0.72
10	1.0	1.0	NO	N.B.B.	T.D.	A	5072	4.08	-	-	0.59	0.35
11	1.0	1.0	NO	N.B.B.	T.D.	A	4136	POOL FROZEN IN 9 HRS.		-	-	-
12	1.0	1.0	NO	N.B.B.	D.D.	A	4136	POOL FROZEN IMMEDIATELY		-	-	-
13	1.0	1.0	NO	B.B.	D.D.	A	5072	3.96	-	-	-	0.38
14	1.0	1.0	NO	N.B.B.	D.D.	N.C.	5072	4.17	-	-	-	0.37
15	0.5 ZrO_2	1.5	YES	N.B.B.	D.D.	N.C.	5072	3.50	2.08	1.42	1.98	0.41
16	1.0 ZrO_2	1.0	YES	N.B.B.	D.D.	N.C.	5072	3.50	2.08	1.42	2.04	1.39

YES: Model includes 1.48×10^6 lb reactor vessel steel (modeled as 20 nodes).

NO: Model does not include reactor vessel.

N.B.B.: Pool surface to structures effective emissivity 0.5.

B.B.: Pool surface to structures effective emissivity 1.0.

D.D. = Density driven heat transfer correlations (Ref. 21).

T.D. = Temperature driven heat transfer correlations (Ref. 20).

N.C. = Heat transfer from back structures by natural convection.

A = Back structures adiabatic.

V.B.2.b Scoping Study

The effect of a number of molten pool heat transfer correlations is examined in subsection V.B.2.b.1. In this section, the effects of various possible MgO melting temperatures and heat transfer coefficients from the back of structures in the reactor cavity are also assessed. Then the effect of heating up and eventually melting the reactor vessel is discussed in subsection B.V.2.b.2. MELSAC cannot yet model the addition to the pool of molten steel from the reactor vessel. The possible effects of dilution of the molten pool by a large quantity of molten steel are discussed in subsection V.B.2.b.2. The effect of varying the effective emissivity from the surface of the pool to the structures is examined in subsection V.B.2.b.3. Various wall configurations are considered in Subsection V.B.2.b.4, including the possibility of protecting the concrete with ZrO_2 rather than MgO.

V.B.2.b.1 Effects of Pool Heat Transfer Correlations

Two basic downward heat transfer mechanisms have been considered. The first, Case 10, uses the downward heat transfer coefficient given by the Kulacki-Goldstein, Reference 8, correlation for an internally heated molten pool. The second, Case 6, models the density-driven heat transfer coefficient, Reference 9, which arises when a material that is less dense than the pool material is melted, so inducing buoyancy-driven motion in the pool.

Figure 9 shows the heat transfer coefficients and downward heat fluxes for these two cases. Although the heat transfer coefficient for Case 10 is two orders of magnitude less than that for the density-driven case (Case 6), the downward heat fluxes are not radically different. The reason for this can be seen in Figure 10. Apparently, the decrease in downward heat transfer (by a factor of 10^{-2}) causes the system to respond by raising the pool temperature 200 degrees Kelvin (360 degrees Fahrenheit). This is an effective increase in the pool-to-bottom-surface temperature differential of, again, about two orders of magnitude, so that the net difference in heat flux is small.

With the downward heat fluxes comparable for Cases 6 and 10, it would be expected that the penetration rate would be similar. Figure 11, indeed, shows this to be the case. The core ladle is penetrated in 4.1 days for Case 10 and 3.9 days for Case 6. The obvious conclusion here, is that barring extreme changes in the pool temperature, which might shift the upward radiative heat transfer, the penetration rate does not seem to be sensitive to the downward heat transfer coefficient.

Another parameter that strongly affects downward heat transfer is the melting temperature of the MgO. Some experimental evidence, Reference 10, indicates that the MgO should melt at the lower MgO-UO_2 eutectic temperature (see Figure 8). This certainly may be true for the initial melting phase, however, as MgO melts and accumulates at the surface, the melting temperature may not be the eutectic temperature. This is because the UO_2 must diffuse through the molten MgO at the surface to form a eutectic composition. The melting rate then becomes a function of the diffusion process rather than the heat transfer process. However, when the pool itself becomes sufficiently diluted with molten MgO (certainly at the point beyond the eutectic composition), the melting temperature must be more nearly the pure MgO melting temperature (3073 degrees Kelvin, 5072 degrees Fahrenheit).

Two cases were run using the eutectic melting temperature. The first case, Case 11, employed the Kulacki-Goldstein downward heat transfer coefficient and the second case, Case 12, used the density-driven heat transfer coefficient. In Case 11, the bulk pool temperature reached the MgO-UO_2 binary solidification temperature of the pool in about 9 hours after the start of MgO melting. With the enhanced heat transfer coefficient in Case 12, the pool again freezes, but, in this case, almost immediately after the MgO begins to melt.

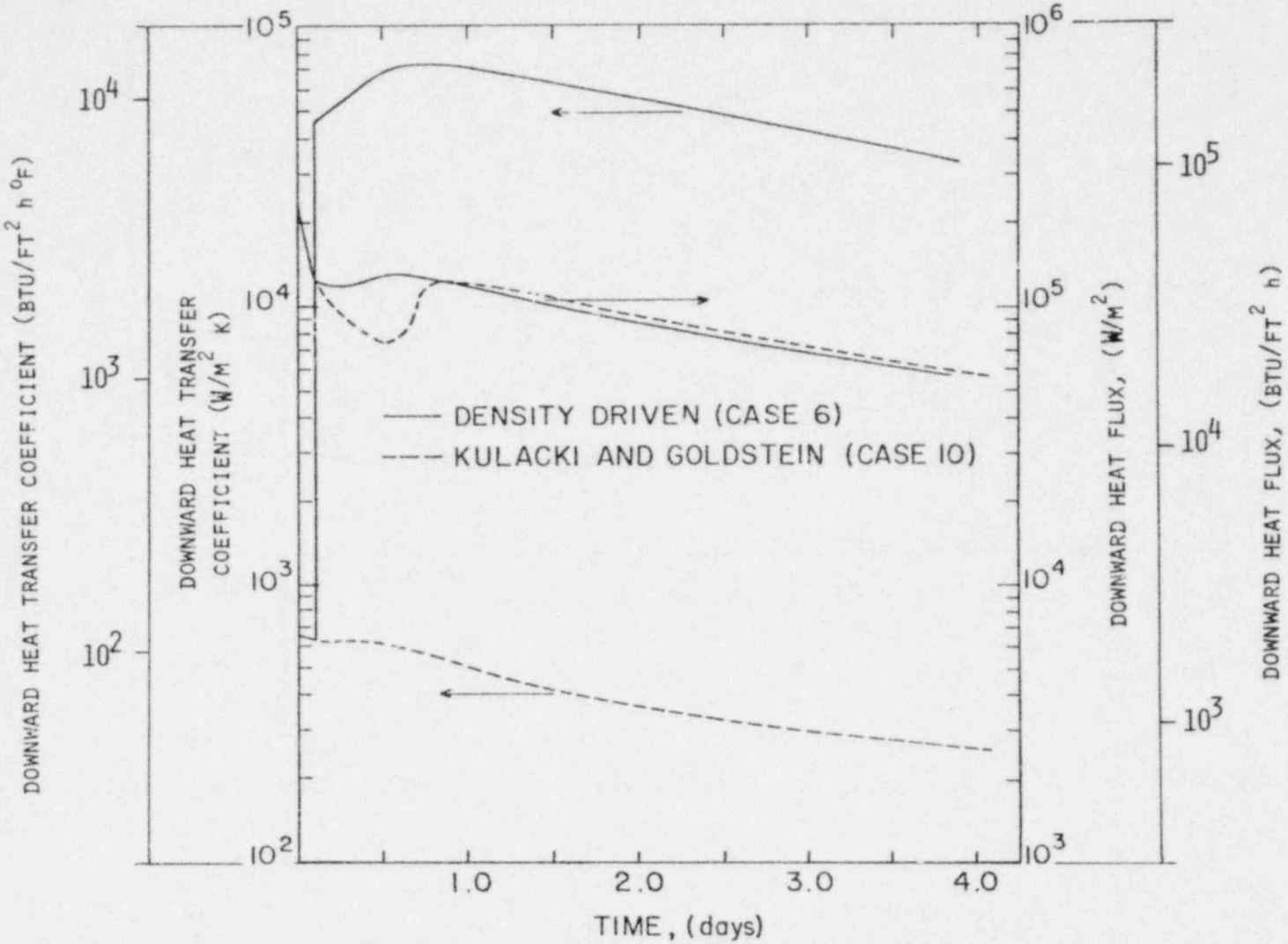


FIGURE 9

Comparison of Molten Pool Heat Transfer Coefficients and Heat Fluxes

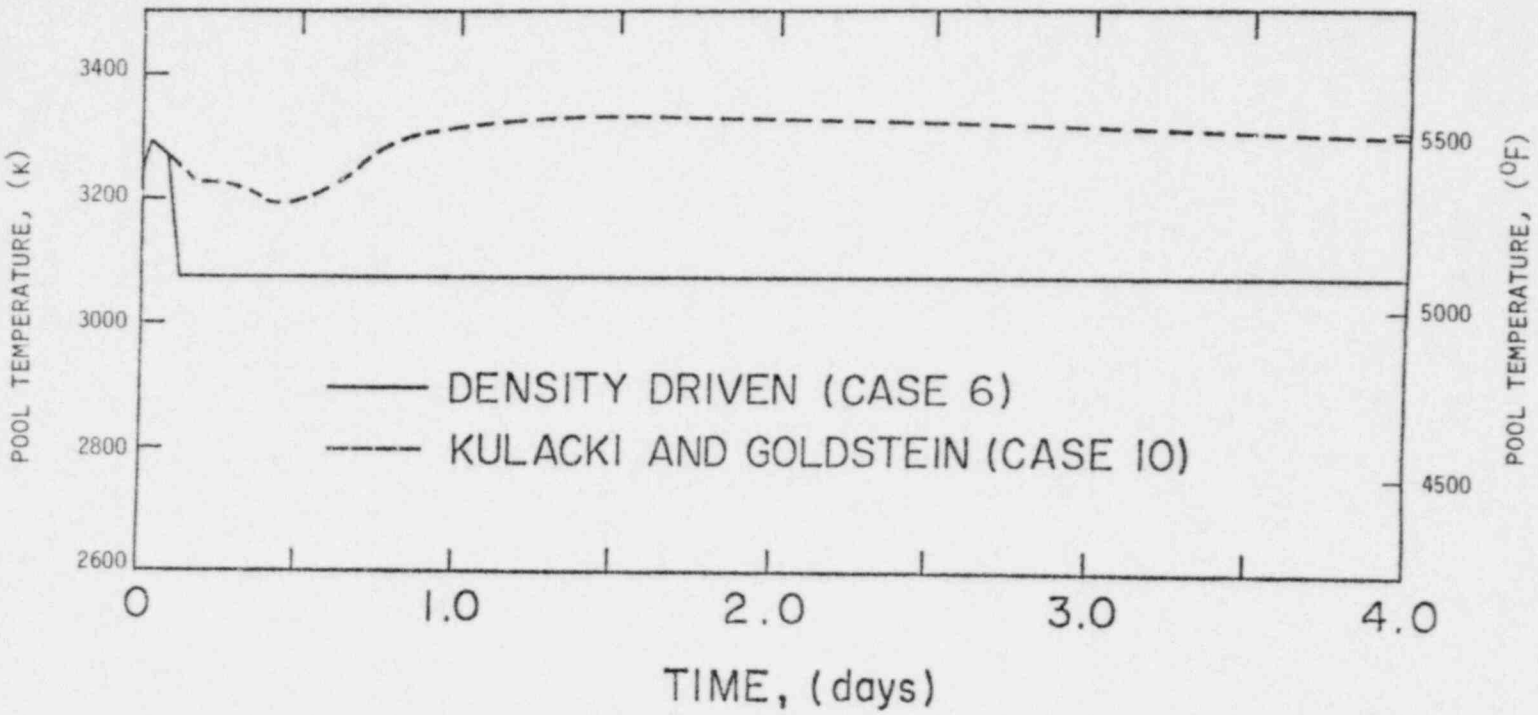


FIGURE 10

Effect of Heat Transfer Correlations on Molten Pool Temperature

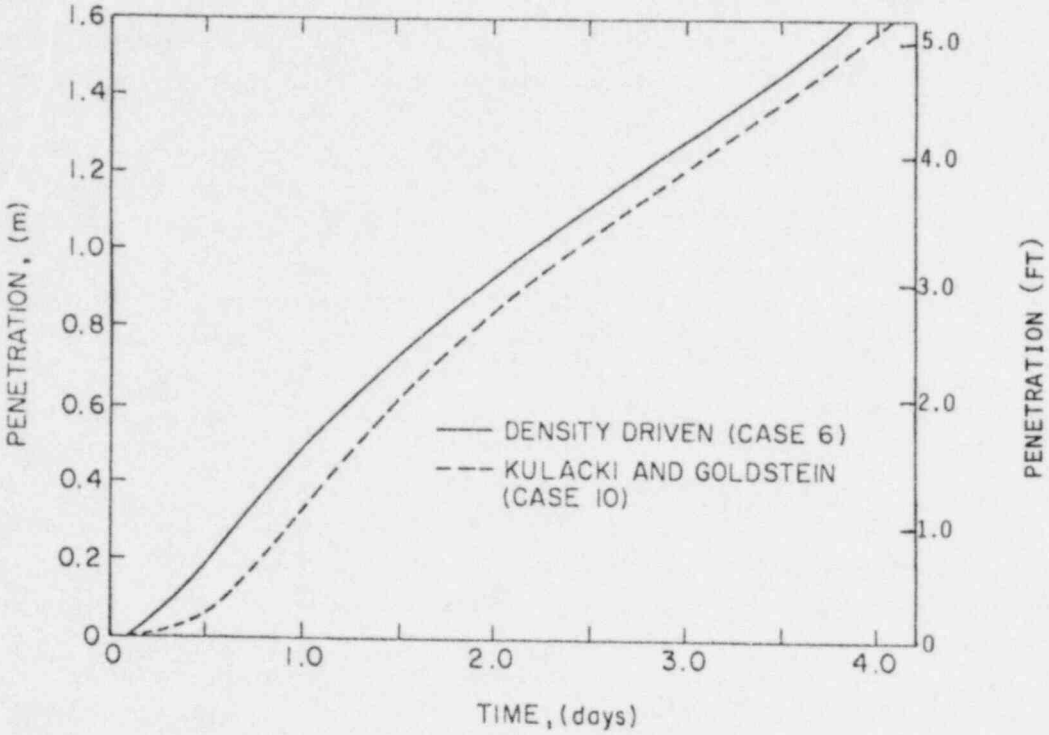


FIGURE 11

Effect of Heat Transfer Correlations on Core Ladle Penetration

The problem is that the eutectic temperature is sufficiently below the melting temperature for the $MgO-UO_2$ binary mixture during the early part of the transient that the pool temperature falls below its melt temperature. The MELSAC code does not presently have a model for calculating a freezing process at the bottom of the pool and, therefore, the calculation is stopped when the pool temperature reaches its freezing point.

An additional case was run in connection with the downward heat transfer process to determine the effect of assuming a heat loss term due to natural convection from the bottom of the core ladle and the back of the cavity walls. Case 14, which employs a natural convection heat transfer coefficient, can be compared to Case 6 which assumes an adiabatic wall. The penetration time is not strongly affected and increases from about 3.9 days to about 4.2 days. In addition, taking credit for heat transfer from the back of the wall does not effectively change the time at which the concrete begins to decompose (1473 degrees Kelvin, 2190 degrees Fahrenheit).

V.B.2.b.2 Effect of Reactor Vessel Melting

A model has been developed to calculate the rate of heat-up and melting of the reactor vessel by radiative heat transfer from the pool surface. However, the current pool model has not been updated to allow for the addition of molten steel as the reactor vessel melts and falls into the pool. The present results were obtained by assuming that the molten steel does not accumulate in the molten pool. The code simply keeps track of how much steel has melted. Furthermore, the quenching effect of molten steel on the pool temperature is not therefore included.

Cases 1 and 6 can be compared to determine the effect of the vessel on the system response. Inclusion of the vessel with its relatively low melting temperature (1683 degrees Kelvin, 2570 degrees Fahrenheit), has the effect of increasing the radiative heat flux and, thus, decreasing the penetration rate. Table 5 shows that the time to penetrate the floor mat is 3.87 days without the vessel and 4.75 days with the vessel. This represents an increase in penetration time of nearly 23 percent.

It is not clear at the present time what the total effect of introducing molten steel will have on the pool conditions. One effect may be to decrease the pool temperature. Whether the resulting mixture will remain molten and in what configuration (layered, or mixed) is not currently known. A simple calculation, which brings the pool into thermal equilibrium with the molten steel just at the time that the entire vessel has completely melted (for Case 1), yields an equilibrium temperature of about 2000 degrees Kelvin (3140 degrees Fahrenheit). This is considerably above the steel melting temperature and below the steel boiling temperature. Introducing molten steel into the pool at the rate it is being melted may have the effect (if it is assumed to be miscible) of lowering the pool temperature, thus, decreasing the radiative heat transfer to the vessel and its resulting melting rate. On the other hand, quenching the pool with molten steel may drop the pool temperature below the MgO melting temperature. If penetration of the core ladle stops, then dilution of the pool with molten MgO would also stop. The question then arises as to whether or not the quenching and diluting effects of the vessel steel would compensate for the loss of dilution previously provided by the molten MgO. The 2000 degrees Kelvin (3140 degrees Fahrenheit) equilibrium temperature is clearly a lower limit. The pool temperature history, allowing for the effect of steel dilution, cannot yet be predicted but would be between the present MELSAC predictions and 2000 degrees Kelvin (3140 degrees Fahrenheit).

V.B.2.b.3 Effects of Heat Transfer from Pool Surface to Structures

Four cases, two with the reactor vessel heat sink (Cases 1 and 5) and two without the reactor vessel heat sink (Cases 6 and 13) can be compared to show the effect of upward heat transfer on the system response. Cases 1 and 6 incorporated non-black body radiation (ε=0.5) while Cases 5 and 13 used black body radiation (ε=1.0).

Although the enhanced upward heat transfer associated with the black body assumption increases the time required to penetrate the sacrificial material, the difference is not significant. There are two effects which tend to produce the above result. The first effect is that with higher pool surface heat fluxes, the wall surface temperature builds up more rapidly with the result that the radiative temperature differential between the pool surface and the wall is decreased. The second effect is that, again due to enhanced upward heat flux from the surface, the crust solidification rate is increased, resulting in a greater crust thickness. The thicker crust more effectively insulates the pool. This, in turn, has the effect of both decreasing the pool crust surface temperature, thereby reducing radiation heat transfer, and keeping the pool temperature higher than might have been expected. The result is that downward penetration is not strongly affected.

The crust thickness for Cases 1 and 5 can be seen in Figure 12. The difference in the crust thicknesses is even more apparent during the initial part of the transient when the solidification rate is significantly greater for Case 5.

V.B.2.b.4 Effects of Cavity Wall Configuration

In view of the necessity to protect the cavity walls from thermal degradation, cases were run using various thicknesses of MgO brick to determine the amount of refractory material necessary to insulate the non-structural concrete and load bearing steel bulkheads. Cases 1 through 4 represent varying MgO thicknesses for the configuration which models the reactor vessel, and Cases 6 and 9 are the cases which do not model the reactor vessel. The general configuration includes a total wall thickness (MgO + Concrete) of 2 feet. Figure 13 shows the concrete surface temperature for Cases 1 through 4. Case 4, consists of 2 feet of MgO and no concrete so that the temperature shown in Figure 13 for Case 4 is the back side of the MgO wall, which in this case is in contact with the steel bulkhead and not concrete. It is evident that ordinary concrete cannot be maintained below its melting temperature in the present configuration for more than about 20 hours.

Two additional cases (Cases 15 and 16) were run using the same general configuration (2 feet of wall) and substituting ZrO_2 in place of MgO. The ZrO_2 has a significantly lower thermal conductivity and, as is shown in Figure 13, with one foot of ZrO_2 protecting the concrete, it can be maintained below the melting temperature out to about 1.4 days. With 1-1/2 feet of ZrO_2 and 1/2 foot of concrete, the concrete could probably be protected out to 2 days. However, an additional problem is encountered with the use of ZrO_2. Its melting temperature is about 123 degrees Kelvin (221 degrees Fahrenheit) lower than that of MgO with the result (as shown in Figure 14 (broken line - Case 15)) that the ZrO_2 begins to melt at about 2 days.

Figure 14 also shows the wall surface temperature for a selection of cases which have been previously discussed. The MgO wall insulation was threatened only in Case 10. In Case 10, downward heat transfer was limited to the Kulacki-Goldstein correlation. The extremely high pool temperature and upward radiation heat fluxes brought the wall to the melting temperature in about 0.6 days. The sudden changes in the slope of the temperature curves in Figure 14 correspond to the times at which the reactor vessel has been completely melted. When the heat sink available in the vessel is gone, the heat flux from the pool decreases which results in an increase in the pool surface temperature and ultimately in an increased heat flux to the cavity wall.

V.B.2.c Results Using Latest Reactor Cavity Configuration

The scoping study described in the previous section provided a basis for the final choice of the five input parameters listed in Table 6. A computer run was made (Case A) with the above parameters and the latest proposed reactor cavity configuration. This represents our current best estimate of the meltdown progression. This case can be used as a comparison with the latest calculations made by OPS and as a means of addressing some of the additional information required by the ACRS (see Appendix D).

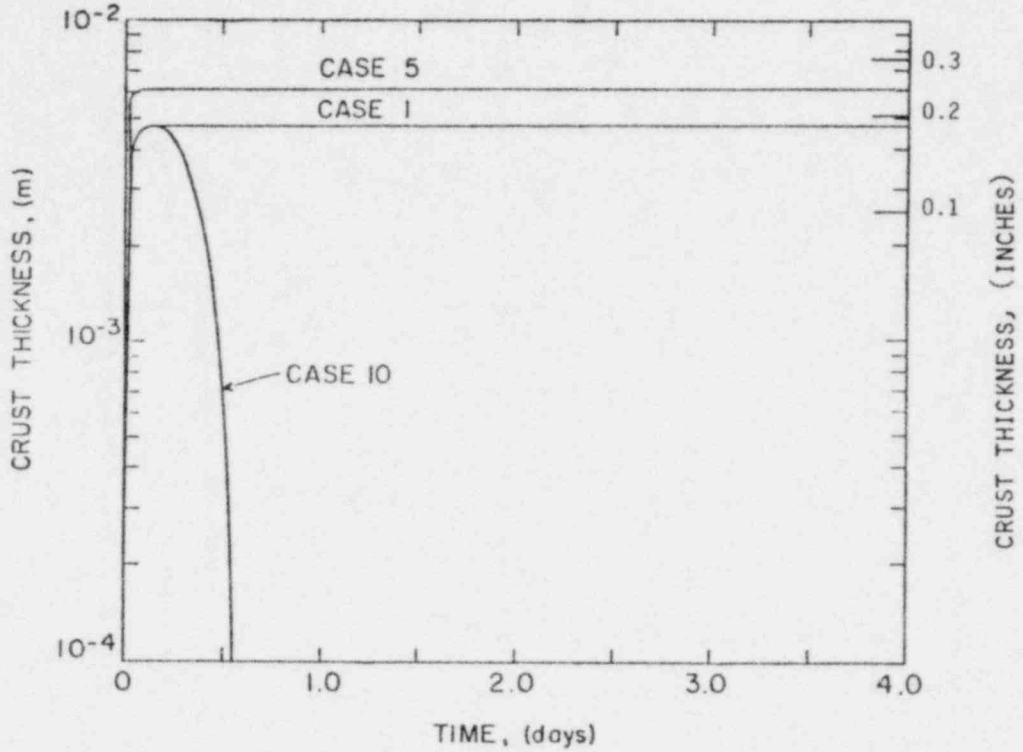


FIGURE 12

Effect of Upward Heat Transfer on Crust Thickness

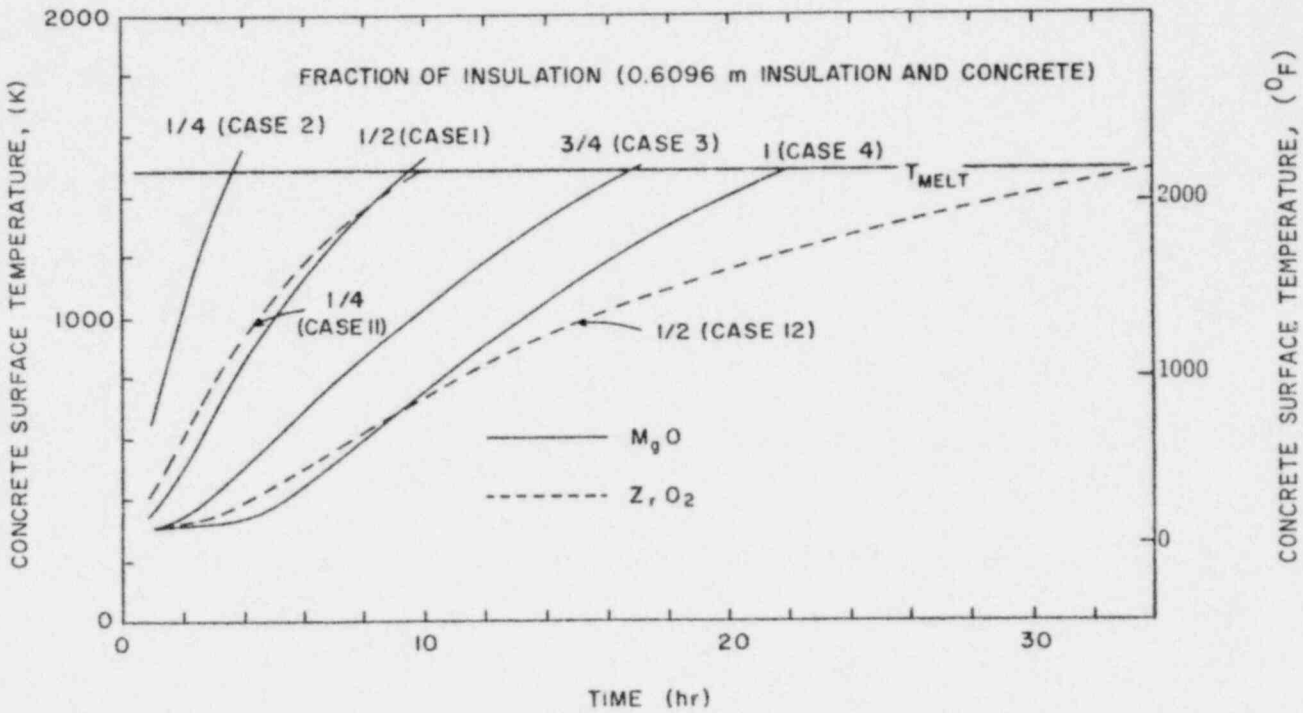


FIGURE 13
Effect of Cavity Wall Configuration on Concrete Temperature

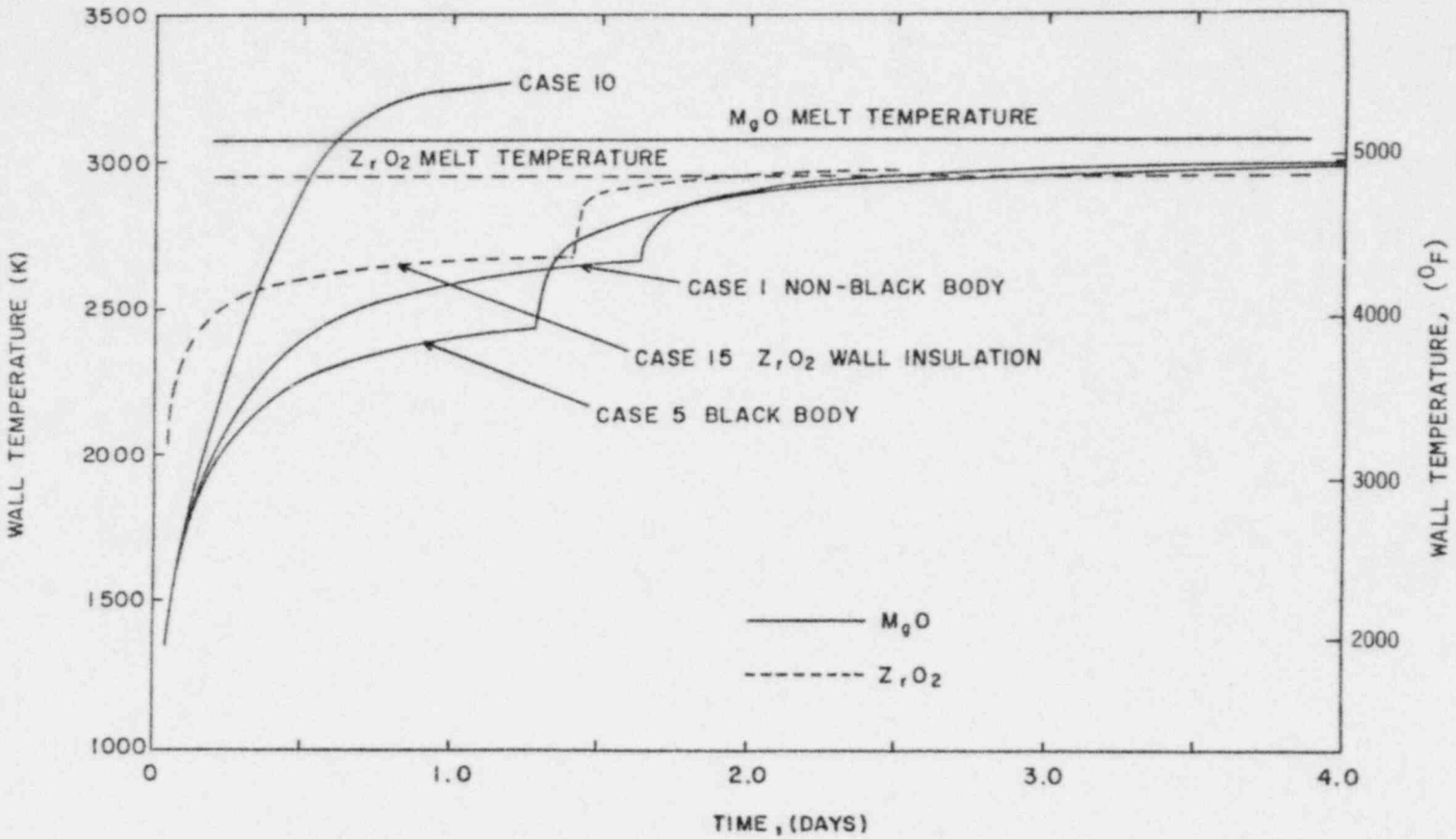


FIGURE 14

Effect of Upward Heat Transfer on Cavity Wall Surface Temperature

TABLE 6

PARAMETERS DERIVED FROM SCOPING STUDY

PARAMETER	VALUE
POOL HEAT TRANSFER CORRELATIONS	DENSITY DRIVEN[9]
MgO MELTING TEMPERATURE	3073 K (5072°F)
HEAT TRANSFER FROM BACK STRUCTURES	NATURAL CONVECTION
REACTOR VESSEL HEATING	670328 Kg 1.48×10^6 lb STEEL MODELED AS 20 NODES
HEAT TRANSFER FROM POOL SURFACE TO STRUCTURES	EFFECTIVE EMISSIVITY $\epsilon = 0.5$

The OPS calculations assume the radiation losses from the pool surface to be decoupled from the pool heat transfer processes. It is recognized by OPS that the problem is highly coupled and a computer program is being developed at OPS to solve the coupled problem. The MELSAC code, Reference 7, solves the coupled problem and consequently the results obtained from MELSAC differ from the calculations and assumptions made at OPS. A major difference between the OPS and the staff results relates to the pool surface temperature histories. OPS assumes the two surface temperature histories shown in Table 7. These temperature histories differ appreciably from the predictions of the MELSAC code (see Figure 10). The effect of the higher surface temperature history predicted by MELSAC is to transfer more heat to the upper structures. The result of this additional heat transfer to the walls is that MELSAC predicts (Case A) that the concrete in the cavity wall (2 feet MgO, 2 feet concrete) will reach its decomposition temperature (1473 degrees Kelvin, 2190 degrees Fahrenheit) after only 1 day, whereas OPS using the pool histories in Table 7, predicts that the concrete will be protected for at least 2 days. In order to determine a wall configuration that protects the concrete (less than 1473 degrees Kelvin, 2190 degrees Fahrenheit) and steel bulkheads (less than 810 degrees Kelvin, 1000 degrees Fahrenheit) beyond the two day period using MELSAC, a scoping study was carried out and the results are presented in Table 8. MELSAC, predicts (Case B) that a MgO wall 3 feet thick would be required to protect the concrete and steel for 2 days, given the input assumptions listed in Table 8. We therefore agree with the OPS response (see Appendix E) to Questions a.2(a) and a.2(b) of the ACRS letter of July 25, 1979, regarding the fact that the cavity walls can be protected for over two days using suitable high temperature insulating brick. However, we do not believe that the configuration of 2 feet MgO and 1.75 feet basaltic concrete as suggested by OPS is sufficient to protect the concrete or steel. We recommend at least 3 feet of MgO to protect the concrete and steel bulkheads.

Regarding Question a.1 in which the ACRS requested an estimation be made of the fraction of the decay heat radiated upwards from the pool, we have plotted the fraction predicted by MELSAC for Case B in Figure 15. The MELSAC result is also compared with the results provided by OPS in Figure 15. After one day, the fraction of heat stored in the walls and vessel ranges from 0.25 to 0.46 using OPS assumptions, whereas MELSAC predicts the heat stored to be 0.76. Similarly, after 5.79 days (melt-through for Case B) MELSAC predicts a fraction 0.5 in the walls and vessel compared with 0.11 and 0.17 at 6 days using OPS assumptions.

The times to melt the whole of the reactor vessel (requested in ACRS Question a.2(c)) by thermal radiation from the molten pool surface varied between 0.5 to 4 days under the assumptions made by OPS. In Table 8 we have included the MELSAC estimate of reactor vessel melting, which is about 1.8 days. The model used in MELSAC is different from that assumed by OPS and is also exposed to the higher pool surface temperature history. In MELSAC the reactor vessel is modeled as a series of connected masses. Heat transfer between each of the vessel masses is by conduction. However, the MELSAC prediction is not inconsistent with the range of reactor vessel melting suggested by OPS.

The erosion rates presented by OPS in Revision 2 of Reference 3 are decoupled from the upward heat transfer and were simply obtained by using a constant fraction of the volumetric heat capacity. The erosion rate predicted by MELSAC (for the wall configuration which protects the concrete and steel, namely, Case B) is compared with the OPS erosion rates in Figure 16. The MELSAC erosion rate clearly shows the coupled effect. At early times the heat transfer to the upper structures is high (because of the large temperature differences) allowing only a small fraction (F) of the decay heat to be directed into the MgO. At later times the upper structures are at higher temperatures and the upward heat transfer is reduced allowing half (F = 0.5) of the decay heat in the core melt to be directed into the MgO at the point of melt-through (5.79 days). This is consistent with Figure 15 in which the fraction of decay heat stored in the walls is predicted to be 0.5 at 5.79 days.

It should be noted from Table 8 that MELSAC Case B predicts that the 3 feet of MgO will protect the concrete and steel for 2 days, whereas the core ladle will hold-up the molten pool for 5.79 days. Under the assumptions of

TABLE 7

POOL SURFACE TEMPERATURE HISTORIES
ASSUMED IN REFERENCE 3*

TIME (DAYS)	TEMPERATURE	
Sandia Estimate:		
0	4712°F	2873 K
1	3632°F	2273 K
2	3524°F	2213 K
4	3308°F	2093 K
6	3092°F	1973 K
OPS (Black Body) Estimate:		
0	3641°F	2278 K
1	2394°F	1585 K
2	2232°F	1495 K
4	2092°F	1417 K
6	1952°F	1340 K

*Reproduced from Reference 3

TABLE 8

REACTOR CAVITY WALL SCOPING STUDY*

CASE	THICKNESS OF MgO IN WALL (FT)	THICKNESS OF CONCRETE IN WALL (FT)	TIME TO PENETRATE CORE LADLE (DAYS)	TIME TO START MELTING VESSEL (HRS)	TIME TO COMPLETE VESSEL MELT (DAYS)	TIME TO START OF CONCRETE MELT (DAYS)	STEEL TEMP. AT START OF CONC. MELT (°F)
A	2.0	2.0	5.25	2.46	1.78	1.04	190
B	3.0	1.0	5.79	2.53	1.83	2.01	1100
C	3.5	0.5	6.00	2.53	1.83	2.48	1600

*ASSUMPTIONS:

MASS OF UO_2	101032 Kg (0.22×10^6 lb)
MASS OF VESSEL	670328 Kg (1.40×10^6 lb)
LADLE THICKNESS	1.6 m (5.25 ft)
MELTING TEMP. MgO	3073 K (5072°F)
MELTING TEMP. CONCRETE	1473 K (2192°F)

POOL H/T BASED ON DENSITY DRIVEN CORRELATION[9].

POOL SURFACE - STRUCTURES EFFECTIVE EMISSIVITY 0.5.

HEAT TRANSFER BEHIND VESSEL, WALL AND LADLE BY NATURAL CONVECTION.

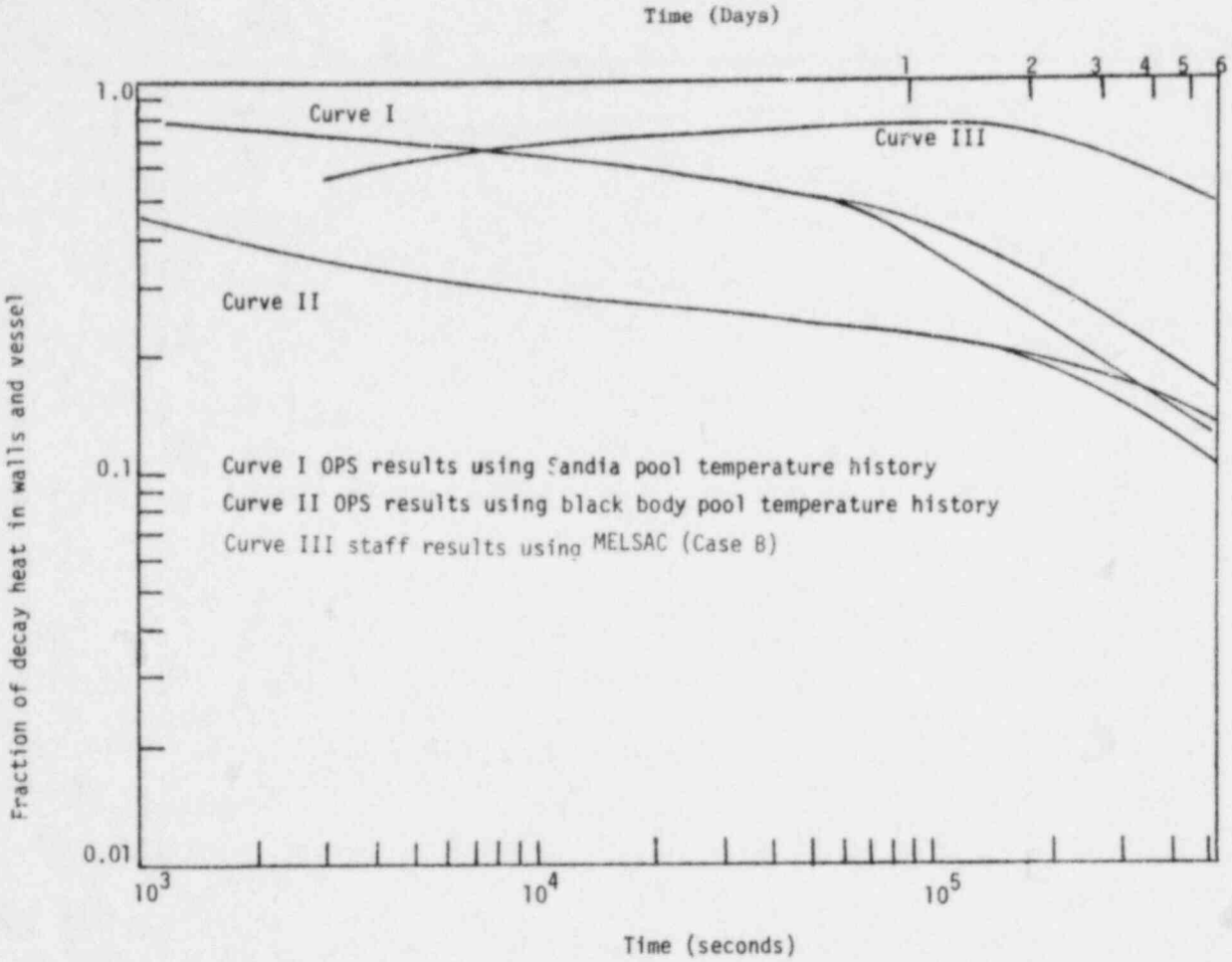


FIGURE 15

Fraction of Integrated Decay Heat Absorbed by Cavity Walls and Reactor Vessel

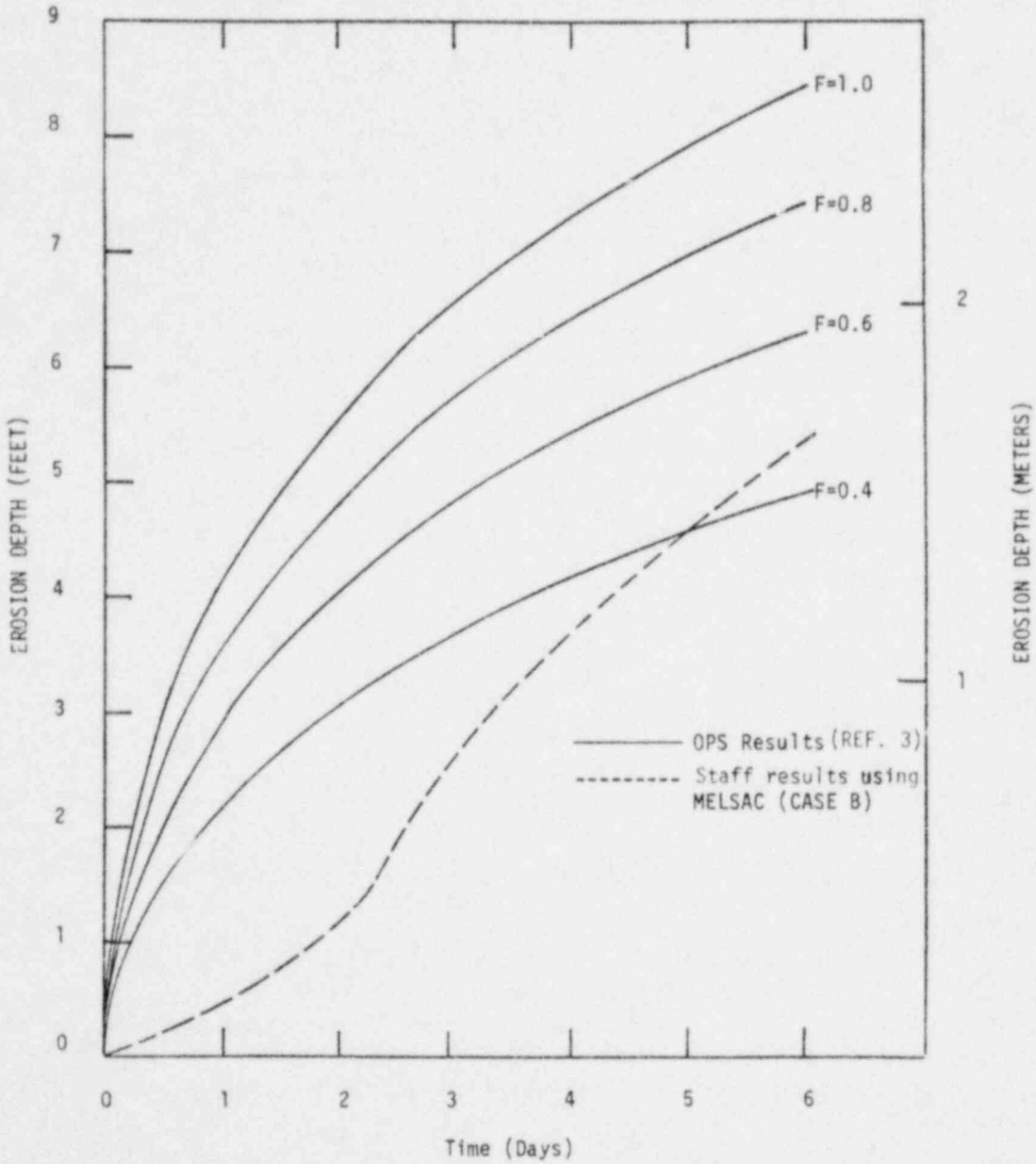


FIGURE 16

Erosion Depth as a Function of Time after Start of MgO Melting

the MELSAC code, and using the above wall configuration, an appreciable thermal transient would be expected in the upper structures in the reactor cavity after the 2 day period and before the molten pool melts through the ladle at 5.79 days.

V.B.3 Thermal Performance Summary and Conclusions

The purpose of Section V.B was to bring together in one place an account of the evaluation that has been performed over the last several months by the staff and its consultants on the thermal response of an FNP core ladle to a core melt event. During this period of time a computer code called MELSAC was developed at the Brookhaven National Laboratory (BNL) for the assessment. It was necessary to develop the MELSAC code because none of the existing melt-front models, References 5 and 6, consider the important feedback effect resulting from the heating up of the structures around the core ladle. We believe that the MELSAC code is the first code to solve, in a coupled fashion, the melting attack of the molten pool on a sacrificial material together with the heating up of structures above the molten pool by thermal radiation from the pool surface. The results obtained from MELSAC are interesting and we summarize them below in subsection V.B.3.a.

V.B.3.a MELSAC Scoping Study

It can be concluded from the scoping study that the penetration of the core ladle is relatively insensitive to the molten pool heat transfer correlations. Apparently, the amount of heat transferred to the structures above the pool controls the penetration of the core ladle. If the lower eutectic temperature of the UO_2-MgO mixture is used as the melting point of the ladle, then the pool is predicted to rapidly cool and freeze. This is because MELSAC considers a two component pool consisting of UO_2 and MgO. At early times the pool properties are close to the properties of pure UO_2. If the lower melting point of the MgO-UO_2 mixture is used as the melting point of the ladle, then the pool temperature rapidly approaches this value and quickly falls below the bulk pool freezing point, which at early times is close to the melting point of pure UO_2. However, the eutectic temperature is not considered to be an appropriate melting point for the MgO over a major part of the core ladle penetration and dilution of the pool with steel and zircalloy would also tend to depress the bulk pool freezing temperature.

The effect of including the additional heat sink associated with the reactor vessel is to increase the core ladle penetration time by about 23 percent. However, the dilution of the pool by the molten steel from the vessel is not currently modeled. The effect of the pool dilution by steel is discussed in subsection V.B.2.b.2. However, a 2000 degrees Kelvin (3140 degrees Fahrenheit) equilibrium temperature, obtained by bringing the total mass of molten steel from the vessel into thermal equilibrium with the molten UO_2-MgO pool, is clearly a lower limit. The pool temperature history, allowing for the effect of steel dilution, cannot yet be predicted but would be between the present MELSAC predictions and 2000 degrees Kelvin (3140 degrees Fahrenheit).

Future work on MELSAC will be directed to addressing the effect of diluting the molten pool with steel and zircalloy. We do, however, consider that the current predictions by MELSAC are convservative with respect to core ladle penetration, heating of the structures in the cavity and melting the reactor vessel. The effect of diluting the pool with steel and zircalloy would tend to increase the time scale of the above events. However, quantifying the changes in time scale at this stage is extremely difficult. The core ladle penetration times predicted by MELSAC are also dependent on the assumed equal lateral and downward heat transfer correlations. If evidence becomes available to suggest that penetration should be faster in either direction, then the MELSAC code could easily be modified.

V.B.3.b FNP Evaluation

The input parameters obtained from the scoping study together with the latest proposed reactor cavity configuration were used to provide a current best estimate of the meltdown progression. This evaluation was used as a comparison with the latest calculations made by OPS and as a means of addressing some of the additional information requested from the NRC staff by the ACRS. A major difference between the OPS and staff results relates to the pool surface temperature histories. OPS assumes the pool surface temperature history drops, whereas the MELSAC code predicts the pool temperature to remain relatively high. The effect of the higher pool surface temperature history predicted by MELSAC is to transfer more heat to the upper structures. After one day the fraction of heat stored in the walls and vessel ranges from 0.24 and 0.4 using OPS assumptions, whereas MELSAC predicts the heat stored to be 0.76. Similarly, after 5.79 days MELSAC predicts a fraction of 0.5 in the walls and vessel compared with 0.11 to 0.17 at 6 days using OPS assumptions.

Calculations using MELSAC indicate that the cavity walls can be protected for over 2 days using a suitable thickness of high temperature insulating brick. The suggestion by OPS that 2 feet of MgO would protect the concrete is based on their assumed pool temperature histories, which are not supported by the results obtained from MELSAC. Based on MELSAC predictions, we consider that at least 3 feet of MgO would be necessary to protect the concrete and steel bulkheads and will be resolved as part of the final design evaluation.

The time to melt the whole of the reactor vessel by thermal radiation from the molten pool surface is estimated to be in the order of 1.8 days using the MELSAC code. Under the assumptions made by OPS, the time to melt the vessel ranges from 0.5 to 4 days. The model used in MELSAC is different from that assumed by OPS and is also exposed to the higher pool temperature. However, the MELSAC prediction is not inconsistent with the range of reactor vessel melting times suggested by OPS.

It should also be noted that MELSAC predicts the core ladle will hold-up the molten pool for about 5.8 days. However, thermal radiation from the pool surface would result in melting of the reactor vessel in 2 days, and using a wall configuration of 3 feet MgO, an appreciable thermal transient would be expected in the upper structures behind the reactor cavity wall after the 2 day period and before the molten pool melts through the ladle after 5.8 days.

The above comments must be qualified as they are based on a first version of the MELSAC code. At present, MELSAC assumes the molten pool to be initially pure UO_2. As the melt front moves into the ladle the code computes the dilution of the UO_2 with molten MgO. However, the dilution of the pool by steel and zircalloy cladding is not presently modeled. For the reasons given earlier, we do, however, consider that the above predictions represent conservatively short estimates of the time for core ladle penetration, heating of the structures and melting of the reactor vessel.

As previously indicated, the FNP core ladle design is still in a preliminary stage and considerable flexibility exists to develop an optimum configuration for protecting the upper reactor cavity structures. As more detailed design information becomes available from OPS, the staff will continue to perform independent evaluations of the core ladle. Also, as noted previously, the applicant is in the process of developing a coupled heat transfer calculational model. Any significant differences between that model and the staff's model can be resolved during the early phases of the final design effort. As better calculational models are developed, the ladle configuration can be optimized for the available space to provide the largest possible core retention hold-up time, including consideration of all of the relevant factors raised in the ACRS letter of July 25, 1979.

V.C Structural Considerations

V.C.1 Core Ladle Structural Design

The structural aspects of the core ladle design are briefly discussed in Section III of OPS Topical Report No. 36A59, "FNP Core Ladle Design and Safety Evaluation," dated April 1979 and as amended by subsequent Revisions 1 and 2.

The applicant has identified the significant structural changes made to the load carrying steel members. The major structural changes required to accommodate the core ladle include: (a) modifications to some vertical frames which serve as stiffeners for a major bulkhead and partially support the primary shield wall, (b) decrease the depth of the transverse girders just below the core ladle, and (c) modification of a longitudinal girder within the reactor cavity, which provides support to the primary shield.

These structures have been redesigned to comply with all of the requirements of the applicable acceptance criteria previously identified in the Safety Analysis Report for this application. For example, in the case of the decrease in depth for the transverse girders, the web thickness was increased to compensate for the decrease in depth.

The changes in the reinforced concrete shield wall are not considered major because: (a) the shield walls are not load bearing structures, (b) the reinforced concrete was replaced with the material of the core ladle, and (c) the applicant provides the same quality of structural supports for the shield wall.

In order to obtain a more in-depth understanding of the effects of this core ladle design modification, the structural reviewer visited the OPS facilities to review and discuss these changes, considering the various models and current drawings of the plant.

In summary, the applicant's position is that the core ladle shall be designed and analyzed to remain functional for all of the operating basis environmental conditions, including the loads imposed by an operating basis earthquake (OBE). For more severe conditions, the plant will be shutdown for inspection of the ladle. In addition, the core ladle shall also be evaluated for design basis environmental loads, including the safe shutdown earthquake (SSE), in order to ensure that there will be no gross failure which might impair the functionality of safety class components.

In view of the unique function of the core ladle, in terms of delaying core melt-through of the FNP barge structure, we consider the applicant's seismic design requirements, as summarized above, to be appropriate and reasonable. The ladle is provided for an event beyond the design basis - a core meltdown accident - in order to provide adequate protection of the environment. For any individual FNP unit, the ladle could be subjected to the event only once during its forty (40) year life. We believe that the design criteria and requirements proposed for the ladle do reflect properly its design functions and are appropriate for protection of the public safety with adequate conservatism. Accordingly, we conclude that the applicant's position on the core ladle design requirements is acceptable. However, as part of the final core ladle design effort, the applicant shall provide the staff with the following information for our review and approval: (a) a proposed method of inspection of the core ladle in the event that the operating basis loadings are exceeded, (b) detailed information pertaining to the manufacture, installation and operation of the core ladle, and (c) detailed information on the analysis and design of the core ladle system and related structures as specified in the Final Environmental Statement. The final design information should include drawings and sketches related to the design and analysis, detailed discussion of the assumptions, method of calculation, calculational procedures, summary of results, codes and standards used and available safety margins.

V.C.2 Impact of Core Ladle on Other Structures

The applicant has addressed the effects of postulated disintegration and collapse of exposed concrete, concrete behind MgO bricks, and steel within the reactor cavity on the integrity of the superstructures, the loss of ladle capacity, the impact resistance of the ladle and its supports, and integrity of steel members.

All of the concrete components that could be exposed to thermal radiation will be protected by a high temperature insulating brick (MgO, ZrO_2, Al_2O_3) or ceramic fiber, such as Fibrefax. No disintegration or melting of these protective shields is expected by the applicant for a 2-day period, although minimal spalling may occur. In addition, it is worthwhile noting that the concrete in question serves only as a shielding material and not as a structural component.

To evaluate the effects of heat radiation, the applicant has analyzed a configuration consisting of 24 inches of MgO, 21 inches of basaltic concrete, and a 1-1/8 inch steel plate using what the applicant considers to be a conservative pool surface temperature - time history. The analysis concluded that the maximum temperature on the concrete would be less than 2200 degrees Fahrenheit, which is below the melting temperature of basaltic aggregate. However, the staff, as indicated in Section V.B., determined that 3 feet of MgO protection would be required to keep the concrete from melting prior to 2 days. In addition, the staff requires that the applicant prove that the melting temperature of basaltic aggregate is the controlling factor and not those of other components of the in-place concrete. All steel members other than the reactor vessel and its contents will be properly shielded with refractory materials to prevent their collapse. All other structures above the reactor vessel cavity will be evaluated by the applicant during the final design stage to determine if they need to be protected by high temperature insulating materials such as those provided for protecting the concrete and steel within the reactor cavity area.

We consider the core ladle support structures to be acceptable since these structures have been redesigned to comply with the structural acceptance criteria as outlined in the Standard Review Plan. However, the applicant should demonstrate in the final design the adequacy of the structural systems supporting the ladle in order to determine their time dependent structural capacity to support the actual ladle configuration. Basically, the applicant should determine that the structural members supporting the core ladle will remain intact during the core debris retention period.

In addition, the applicant performed analyses with the following postulated new loading conditions: (1) reactor vessel bottom head impact, and (2) upper reactor vessel impact, in order to consider the appropriate impactive loads in the evaluation of the core ladle. The analyses indicated that loading condition (1) noted above controls the design. The analysis for loading condition (1) showed that the ladle can resist the impact load without failure. However, the applicant should document in the final design report the adequacy of the structures supporting the ladle for their capability to resist these impactive loads.

With respect to the ACRS concern raised in their subcommittee meeting of November 17, 1979, relative to disintegration of reactor cavity concrete, we previously pointed out that this concrete is for shielding purposes only and is not a load bearing structure. We agree with the OPS design criterion of protecting the inside surface of the reactor cavity concrete with a suitable thickness of refractory material to prevent the concrete from reaching its decomposition or melting temperature during the core debris retention period. Although the concrete will reach temperatures in excess of those necessary to release the free and bound water content, our judgment is that the concrete will not disintegrate in the sense of forming a loose powder, collapsing and falling out of place. Initial experimental evidence from Sandia suggests that as the concrete approaches its melting temperature, it will tend to powder and pack together with the underlying cement retaining some strength. Data does exist which

shows that basaltic concrete still retains 15 percent of its initial compressive strength at temperatures as high as 800°C. The staff will continue to investigate this concern further as part of the final core ladle design evaluation when more experimental information will be available on the behavior of concrete at its melting temperature. It should also be noted that there exist fallback designs which could utilize approaches, such as (1) more refractory protective material, (2) refractory concrete and (3) more effective insulating materials.

Reference should also be made to Appendix F for the staff's evaluation of the OPS response to ACRS Question No. a.3 concerning structural aspects of the FNP core ladle.

V.D. Shielding Considerations

The FNP core ladle and surrounding shielding are designed to reduce the whole body dose rates in the surrounding compartments to acceptable levels. Each layer of MgO bricks in the core ladle will be offset and staggered from the joints of the layers immediately above and below it. This arrangement serves to minimize radiation streaming through the core ladle to the compartments below it.

The applicant's design basis dose rate criteria for the compartments adjacent to the ladle cavity is less than 15 millirem per hour during normal full power operation. These areas contain no equipment and therefore access is generally only required for inspection of the platform structure. A dose rate of 15 millirem per hour allows an occupancy of about 6 hours per week while ensuring that the allowable whole body dose of 1.25 rem per calendar quarter is not exceeded. In the compartment directly below the cavity floor, inspections of the platform structure can be accomplished during plant shutdown. Since access to this compartment is not required during plant operation, dose rates in excess of 15 millirem per hour are permitted here. The design dose rate in occupied areas where equipment is located is less than 2.5 millirem per hour.

Using the following computer codes; DOT-III.W, ANISN-W, NAGS, and MAP, the applicant has calculated the expected gamma and neutron dose rates at the bottom of the core ladle. The calculated dose rate here is about 6 millirem per hour, a factor of 2.5 lower than the design dose rate of 15 millirem per hour for compartments adjacent to the core ladle cavity. Only about 0.1 millirem per hour of the 6 millirem per hour penetrating the floor is due to neutrons.

We find the shielding provided by the core ladle and surrounding walls to be sufficient to maintain doses to personnel in these areas below the quarterly dose limits of 10 CFR Part 20, based on an occupancy of 6 hours per week. Based on this, and the fact that the compartments adjacent to the ladle-cavity contain no equipment and therefore have minimum occupancy requirements, we find the design of the FNP core ladle acceptable with respect to radiological shielding provisions.

V.E. Containment Pressure Response

The addition of the core ladle reduces the free volume in the reactor sump region, thereby increasing the pressure response to LOCA blowdown. In a letter dated July 11, 1979, the applicant analyzed the effects of adding the core ladle on the sump region pressure response. The free volume in the reactor sump region is 20,900 cubic feet without the ladle and 20,031 cubic feet with the ladle. The differential volume of approximately four percent would have an inconsequential effect on the previously calculated peak pressure of 9 pounds per square inch gauge in the reactor sump region since the vent areas of 50 square feet remain unchanged. Addition of the usual 40 percent margin in the calculated peak pressure does not result in the limiting pressure for design of the reactor sump region. The limiting design pressure of 32 pounds per square inch gauge results from the requirement that the reactor sump region withstand the containment design pressure of 15 pounds per square inch gauge plus the

hydrostatic head of water which would result from a break at a reactor vessel nozzle. This limiting design condition is unaffected by the addition of the core ladle.

The applicant has provided sufficient information to show that the containment design pressure would not be violated with the addition of a core ladle. We, therefore, conclude that the design of the containment with the core ladle is in compliance with the requirements of Criterion 16 of the General Design Criteria and is acceptable.

V.F Radiological Considerations

This section addresses in a semi-quantitative sense the impact of the core ladle on airborne radiological releases for core melt type of events. Concern in this area was raised by the ACRS in their subcommittee meeting of November 17, 1979.

The addition to the plant design of a refractory sacrifical heat sink, or core ladle, below the reactor vessel does not affect the previous assessments of the radiological consequences of air-dispersed radioisotopes for accidents within the design basis. For a core melt accident, however, a potential for alteration in gas-phase dispersal derives from the extended period in which molten material would remain within the core ladle, rather than melting quickly through the reactor cavity concrete and then being quenched by water beneath the FNP barge hull. While the length of time available for volatilization of fission products from the melt will be increased by about an order of magnitude with the ladle, the mole-fractions, and hence vapor-pressures, of the fission products will be reduced by about an order of magnitude through dilution with MgO. In addition, in comparing MgO to concrete, there will be essentially no gas sparging of activity from the melt when the core debris interacts with MgO, unlike the situation with concrete where significant gas and vapor generation occurs. With MgO the pressurization rate and driving force for releasing activity from a failed containment will be significantly less than with concrete. Each of these would tend to reduce the airborne releases by using MgO instead of concrete.

Radioisotopes which are concentrated in the gap between the fuel and cladding rather than being dispersed throughout the fuel, such as those of the noble gases, iodine, and cesium, are susceptible to release to the containment atmosphere prior to the melting of the core. In addition, the vapor pressure of cesium species within the fuel is much greater in oxygen-depleted melts, such that the favorable conditions for cesium volatilization exist during the initial melting within the reactor vessel. Species which are volatilized prior to the entry of the core into the ladle are, of course, unaffected by the existence of the ladle.

Basaltic concretes are complex mixtures of silicates and aluminates of calcium, magnesium, and other elements, containing varying amounts of water, both interstitial and as hydrates. Silicates are large families of extended oxyanions having varied shapes and sizes. Silicate mixtures do not crystallize on cooling, but form super-cooled liquids (glasses). Silicates express a vapor pressure as SiO, and are thus oxidizing agents with respect to an oxide melt. Due to the comparatively high mass of silicate anions, they are about an order of magnitude less effective for vapor pressure reduction by dilution than MgO is, and might form additional phases which would further reduce their effectiveness. While cold silicate glasses might have desirable features as fission product solvents, the pyrolysis of basaltic concrete in comparison to the fusion of MgO has no desirable properties with regard to lessening the volatilization and dispersal of fission products. In addition, as previously indicated, the interaction of core melt debris with concrete results in significant generation of noncondensible gases (H_2, CO_2) and water vapor which can sparge activity from the melt for subsequent release to airborne pathways. Gas phase release of a species to the containment atmosphere is proportional to the product of the vapor pressure of the species times the volume of gas that can be saturated with the vapor; hence, even a species having a very low vapor pressure can be gas-dispersed by a sufficiently large flow of gas. The minimization of gas generation

which can sparge a molten core is of as great an importance as the core melt species vapor pressure in minimizing volatilization. Also, the gas and vapor generation from concrete will significantly increase the pressurizaton rate and driving force for releasing activity from a failed containment.

With an MgO core ladle, the fuel, melted portions of the ladle, and most of the fission products and any oxidized cladding would form a single phase at very high temperatures, with additional vapor and molten metal phases. Volatilizations from such molten masses* would be chiefly silver and cadmium control rod material, iron, nickel, and uranium trioxide formed by high temperature disproportionation of the fuel. The evaporation of these materials would be significant heat transport mechanisms from the melt. The addition of a core ladle to the design, therefore, does not remove the potential for containment overpressurization, although it is likely that it will delay and extend the period of time during which such overpressurization might occur.

Table 9 lists the core inventory by groupings of elements having similar chemical properties, with the amounts, fission product beta decay rates, and oxide phase cationic mole fractions of each grouping. Since the molar density of MgO is 2.5×10^4 moles per tonne, 28 tonnes of MgO would be sufficent to dilute the listed cationic mole fractions (and hence vapor pressures) of oxide melt components to one-half of the values given. If very little cladding oxidation were to occur, the ZrO_2 fraction would be reduced, and 18 tonnes of MgO would be sufficient. Several times this amount of MgO would be expected to be dissolved by the melt prior to melt-through of the ladle. While spatial and temporal temperature and concentration distributions are uncertain, it is apparent that the fission product vapor pressures within the molten phases would drop rapidly upon entry into the MgO ladle.

The human inhalation dose conversion factors for the halogens (critical organ being the thyroid) are, with one exception, about an order of magnitude greater than those of other element groupings, and the specific activity of the halogens is the largest of all groupings. The halogens are insoluble in the molten oxide and metallic phases, while the other groupings are soluble in one or another of these phases. As a result, the halogens greatly dominate the airborne radiological consequences of reactor accidents.

The sole exception of halogen dominance of inhalation dose conversion factors is the alkaline earths (Sr-90), which have an order of magnitude lower specific activity than the halogens. Strontium, like magnesium, is an alkaline earth, and as such is very soluble in MgO.

The closest competitor to the high halogen specific activity is the tellurium component of the "A" period grouping. Dose conversion factors for tellurium isotopes (critical organ being the lung), however, are nearly two orders of magnitude less than those of the halogens. Under these circumstances it is reasonable to expect that radioiodine would dominate the airborne radiological consequences to man as a result of a core-melt accident.

Delay in melt-through of the FNP barge would reduce the potential for radiological environmental impact to the hydrosphere by reducing the inventory of short-lived radioisotopes subject to dissolution, and by allowing time for actions to be taken to prevent further dispersal. By diluting the core debris with MgO, impact would be further reduced by decreasing the decay energy density available to drive dispersal mechanisms, and by reducing the concentration (and therefore leach rate) of radioisotopes within the debris.

The tables and discussion in Section 7.3 of Reference 1 indicate that the bulk of the difference between land-based and floating plants in regard to core melt impacts is attributable to aqueous pathway releases during the

*S. D. Gabelnich and M. G. Chasanv, "A Calculational Approach to the Estimation of Fuel and Fission-Product Vapor Pressures and Oxidation States to 6000°K," ANL-78-7 (October 1972).

TABLE 9

RELATIVE COMPOSITION OF MOLTEN CORE

Element Group	Moles/10^4 moles of Fission	BETA Decay Curies	Cationic Fraction
Halogens: Br, I	130	$3x10^8$	Vapor phase
Alkali Metals: Rb, Cs	2,290	$4x10^7$	$3.3x10^{-3}$
Alkaline Earths: Sr, Ba	1,810	3.10^8	$2.6x10^{-3}$
Noble Metals: (includes control rod Ab) Ru, Rh, Pd, Ag	22,670	$3x10^8$	Metallic Phase
"B: Transition: (includes oxidized cladding) Zr, Y, Nb, Mo	261,270	$3x10^8$	.38
"A" Period:* Cd, In, Sn, Sb, Te	290	$2x10^8$	$4x10^{-4}$
Rate Earths: (includes La)	4,880	$1x10^9$	$7x10^{-3}$
Actinides: U, Pu, Np	425,000	------	.61

*Chemical similarities are less pronounced amongst these elements than in the other element groupings. Cd and Sn are likely to be extracted to the metallic phase, while Te has a reported refractory rare earth telluride. Cd and In control rod material has not been included.

first week following the accident. Appendix E of Reference 1 supports the assumption that interdiction methods could be effectively used after that time. The findings illustrated previously in Figure 16 showing erosion depth as a function of time may be interpreted as an estimate that the FNP core ladle is capable of reducing the radiological environmental impacts of the FNP to levels comparable with those of land-based plants.

We conclude that, although recognizing the uncertainties and complexities of fission product behavior during core meltdown events, fission product evaporation, sparging, and subsequent release to both the air and liquid pathways are expected to be less with an MgO core ladle than with concrete. However, as part of a longer-term effort associated with the final design evaluation of the FNP core ladle, the staff will perform more detailed, integrated studies that will involve modeling the containment and core ladle as a system to ascertain the system response to various core meltdown sequences. Such studies will include consideration of the effect that the core ladle has on the airborne radiological releases. In addition, the effect that the core ladle has on the containment building pressure, temperature and hydrogen concentration transients will be evaluated for various core degraded accident scenarios. Such containment systems analyses will also consider hydrogen control and controlled-filtered-vent features that may be required as a result of rulemaking proceedings or other Commission actions emanating from post-Three Mile Island (TMI) related requirements.

V.G. Plant-Site Interface Criterion

The staff concluded in the FES, Part III, that, although the requested manufacturing license did not apply to specific sites, applicants who requested construction permits and licenses to operate FNPs at specific sites would have to comply with certain environmental siting requrements. Those related to a potential core melt accident at an FNP are as follows:

1A. Provide an assessment of actions that will be taken by the owner/operator of an FNP, including source and pathway interdiction methods, that would provide further protection to the public, the operating staff and the environment, in the event of a highly unlikely core-melt accident by taking advantage of the delay in core melt-through provided by the magnesium oxide (or equivalent) pad beneath the reactor vessel.

1B. Proposed FNP sites in estuaries, rivers or near barrier islands must be appropriately modified in an environmentally acceptable maner such that in the event of a core-melt accident, the release of radioactive material into the surrounding water body shall be limited to levels that will not result in undue impact to man or the ecosystem.

The NRC staff position is that "...finding acceptable FNP sites in estuaries, rivers, or near barrier islands, will most likely be extremely difficult, but [the staff] cannot conclude that there are no acceptable estuaries, riverine or barrier island locations for FNP emplacement when appropriate mitigative actions are taken" (Reference 2). Both the staff and the U.S. Environmental Protection Agency (EPA) concluded that siting FNPs in such areas could produce a significant potential for adverse environmental impact, particularly with actions associated with construction and maintenance dredging. Furthermore, in its assessment of the FNP core-melt accident at an estuarine or riverine site, the staff concluded (Reference 2) that a direct release of radioactive material to such areas would result in unacceptable consequences to the environment. As such, the staff formulated environmental siting requirement 1B which must be complied with by an applicant who wishes to locate an FNP at a specific site in an estuary, river, or near a barrier island.

With respect to actions and time periods considered practical to isolate the core melt material for river and estuary sites, the staff concluded that total isolation of radioactive core-debris from open estuarine/riverine

waters, following a core-melt accident would be very difficult to achieve. Furthermore, the staff concluded that total isolation would not be necessary provided the combination of site characteristics, FNP design features, and interdiction methods could provide adequate assurance that a core-melt type accident would not produce risks any worse than the typical estuarine/riverine land-based plant considered in the LPGS Report (NUREG-0440), Reference 1. Thus the staff required (siting requirement 1.B) that an FNP site in such areas must be modified to restrict and allow for interdiction of the potentially widespread and chronic release of radioactivity in the event of a core-melt accident. Siting requirement 1.B is stipulated independently of manufacturing license condition No. 4, Reference 2, which requires that the FNP be redesigned to incorporate a core ladle. The core ladle design would delay core debris melt-through of the FNP barge hull in order to provide additional time to implement interdictive measures; but in the event of an actual melt-through, radioactive debris would undoubtedly be released to the ambient estuarine/riverine environment. This would, in the staff's view, produce unacceptable environmental impacts.

Environmental siting requirement 1.B is intended to prevent waterborne contaminants resulting from core-melt type accients from spreading offsite in an uncontrolled manner. The bases for the requirement included consideration of mitigation and interdiction techniques that could be employed at both land-based and FNP sites to limit the offsite migration of activity into the estuary or river and reduce the long-term environmental consequences of such releases. The environmental consequences in most estuary and river siting situations were judged likely to produce both acute and chronic effects on biota due to the generally very slow natural pollutant flushing capability of such water bodies. Classes of aquatic biota might be destroyed, therefore impacting the ecosystem for years. A direct result of such chronic conditions upon biota would be an indirect effect upon man due to relatively long-term public restriction of water resource related activities on a large scale.

In order to implement this environmental siting requirement, the applicant has proposed an additional plant-site interface criterion in their FNP Core Ladle Topical Report, Reference 3. The criterion requires that site modifications be made at proposed specific FNP sites in estuaries and rivers to ensure that the environmental consequences of an FNP core-melt accident in these areas would be no worse than those for estuary sited typical land based plants considered in the LPGS Report. The staff has accepted this criterion, noting that the consequences of core-melt type accidents would be assessed for any proposed estuary or river FNP site. The assessment will compare specific FNP site and plant design information with the typical estuarine/riverine land based reactor sites assessed in the LPGS Report. If the specific FNP estuarine site characteristics fall outside of the range of the typical land based estuarine site parameters assumed in the LPGS Report, then a specific core-melt accident analysis will be performed for the proposed FNP site. It must be emphasized that the staff has not yet developed firm criteria that could be used for purposes of specifications on such site modifications, but has in mind the objective of lowering the risk to levels comparable to those for land-based power plants. Candidate criteria that require consideration include, but are not necessarily limited to, radiation protection standards as set forth in 10 CFR Part 20, protective action criteria as set forth by the U.S. Environmental Protection Agency and/or the Food and Drug Administration, or other independently established individual and/or population dose criteria. Such criteria may also include considerations of the expected effectiveness of interdictive actions.

V.H. Related Steel Industry Experience

V.H.1 Introduction

There is considerable industry experience in steel-making operations which is relevant to the FNP core ladle. Refractory materials are used to contain melts at high temperatures in the metal refining industry. In steel-making operations, crucibles and hearths of large furnaces are basically large shallow steel containers lined with a refractory material to provide a heat resistant liner capable of withstanding severe thermal shock and physical impact. For the past century, crucibles and hearths lined with a refractory material have been used successfully to contain molten iron and steel for long periods of time during the refining processes. In addition, the molten metal at the time of discharge is poured into large capacity ladles, which are also lined with

a refractory material. The ladles, therefore, are subjected to the same extreme service conditions (intense heat
and impact loads) and requirements as the crucibles and furnaces themselves.

As a part of the review of the FNP, the staff and its consultants have investigated relevant experience at the U.S. Steel Corporation. At the Edgar Thompson Works in Pennsylvania, U.S. Steel operates three blast furnaces, two basic oxygen furnaces (BOF) and a continuous casting mill. The BOF units each have a capacity of 200 tons of molten metal per heat. We have independently observed and discussed the current practices of the steel industry. These are of interest because OPS has proposed to contain a molten core in a postulated core meltdown in a ladle which will be similar in design to ladles employed to contain molten steel. Furthermore, since molten steel will be a major constituent of the expected molten core debris, the experience of the steel industry in handling and containing molten steel is highly relevant.

V.H.2 Chemistry of Steel Refining Process

Before considering the applicability of the steel industry's experience to the FNP core ladle, it is useful to briefly review the chemistry of the steel refining process. Iron ore is reduced to a high-carbon alloy of iron, called pig iron, in a blast furnace. Coke is burned in the blast furnace forming carbon monoxide which reduces the ore:

$$Fe_2O_3 + 3CO = 2Fe + 3CO_2 \qquad (1)$$

Some of the freshly reduced iron, in turn, reacts in the blast furnace to form pig iron, either by physical contact with hot carbon or by the reaction:

$$3Fe + 2CO = Fe_3C + CO_2 \qquad (2)$$

Impurities present in the ore and coke ash must be removed. Limestone is added to the materials in the blast furnace to "flux" or convert the impurties into slag. Slags may be basic if they contain a preponderance of lime (CaO) or magnesia (MgO) or acidic if they contain a preponderance of silica (SiO_2) or alumina (Al_2O_3).

At the next stage of the steel refining process, the metal enters the BOF unit where it is converted into steel. Oxygen from a lance or probe is injected at supersonic velocity into a mixture of iron scrap and molten iron from the blast furnace. No heat is added to the metal at this point; instead, chemical reactions supply all of the energy required to melt the scrap and burn off impurities. Oxygen entering the BOF unit burns the first pig iron it encounters, forming FeO which is quickly distributed throughout the molten pool. Initially, the silicon and manganese impurities are removed by the reactions:

$$Si + 2FeO = 2Fe + SiO_2 \qquad (3)$$

$$Mn + FeO = Fe + MnO \qquad (4)$$

These oxides are insoluble in molten iron and accumulate, forming a layer of slag. Some 5 to 30 percent of the iron in the BOF unit is converted to FeO and collects in the slag layer. The reactions generate much additional heat and raise the temperature of the remaining metal. Gradually the carbon begins to burn off also:

$$FeO + Fe_3C = 4Fe + CO \qquad (5)$$

In a few minutes, the carbon is gone. Inside the BOF unit, the molten material consists of a layer of iron, practically free of carbon, silicon and manganese, covered by a layer of slag.

Once all of the impurities have been removed, charcoal and manganese can be added to produce steel with the properties desired. For example, high carbon steels can be made by adding charcoal to molten steel in the ladle.

V.H.3 Mechanical Shock

During each heat in a BOF unit, 20-30 percent of the initial charge, or 40-60 tons, consists of scrap metal. The remainder is molten steel from the blast furnaces. Pieces of scrap metal as large as ten tons in size are dropped from a height of 20 feet onto the refractory in the BOF unit.

At one time, there was concern about the effects of mechanical shock on the MgO refractory. Twenty years ago, a bed of smaller sized scrap metal was placed on top of the refractory to protect it from the impact caused by large pieces. With the improved refractories available today, there is no concern about mechanical shock caused by the impact of scrap. Areas of the furnace impacted by scrap do wear slightly faster but the hot strength of the material is quite good.

It also should be noted that supersonic velocity oxygen from the oxygen probe in the BOF unit blows the steel scrap around in the furnace. Even so, wear on the furnace walls from mechanical shock is minimal.

V.H.4 Preheating of MgO, Cracking and Thermal Shock

Usually, MgO liners are preheated for 3-4 hours with gas jets prior to use. After preheating, the liner is at a temperature of 2000 degrees Fahrenheit. No attempt is made to heat the liner to the approximately 3000 degrees Fahrenheit temperature of the metal.

In certain applications, high purity MgO is used successfully without preheating. An example is the "slide gate" used to control the flow of molten steel from a ladle. A diagram of the slide gate is shown in Figure 17. When the MgO blocks are moved so that the apertures coincide, molten steel can pass through and leave the ladle. The design and required conditions of operation make preheating of the slide gates an impossibility. The temperature of the slide gate prior to contact with molten metal is about 100 degrees Fahrenheit.

No attempt is made to preheat the BOF unit prior to use, even when it is cold. During normal operation, the furnace cools to room temperature 2-5 times during the life of a liner. There have been periods when the BOF units have been operated only one or two shifts per day due to a lack of demand for steel. Industry studies suggest that liner life will be reduced by 25 percent if the liner is allowed to cool to room temperature every day and is not preheated prior to use.

Some steel industry experience exists for situations where molten steel has been poured into cold MgO ladles. This has occurred accidently when personnel have forgotten to preheat the ladles. In these situations, damage to the ladle has been minimal rather than catastrophic. The only damage that has been observed is cracking which occurs down to a depth of 2-3 inches. Below 2-3 inches, the liner remains undamaged. Since the cracked material remains in place, it continues to contain the molten metal.

Emergency ladles, used to hold molten steel when "breakouts" or leaks occur, are lined with inexpensive fireclay and are not preheated. Even so, the ladles will hold molten metal without difficulty.

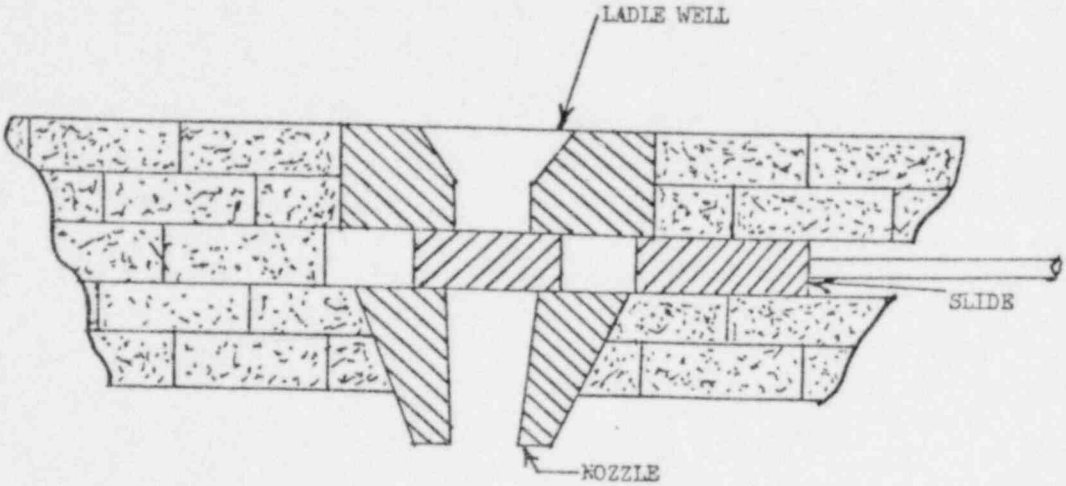


Figure 17: Slide Gates (not to scale)
Molten steel can leave the ladle when the MgO slide gates are aligned. (Artists conception)

FIGURE 17

The cracking noted in cold MgO ladles in the steel industry, from molten steel pours on cold liners, should not be a problem in the case of the FNP core ladle. Unlike the steel industry ladles, which must be used repeatedly, the core ladle will be used only once. If the molten core debris affects the core ladle along the lines of steel industry experience, the loss of a few inches of MgO out of a five foot thick layer, due to cracking, should not pose a problem. Furthermore, to the extent that steel industry experience is relevant, it should be mentioned that the cracked bricks remained in place and, thus, continued to provide protection even though they were cracked.

Although the UO_2-steel mixture in a core meltdown will be at a higher temperature than normally encountered in the steel industry, it should be noted that the UO_2, at least, is a poorer conductor. There is evidence from experiments performed several years ago at The Aerospace Corporation, References 11 and 12, that cracking induced by molten UO_2 will not be worse than the steel industry experience. In those experiments, there was little evidence of significant damage from cracking induced by pours of molten UO_2. Whether molten steel at temperatures higher than those encountered in the steel industry would be significantly worse than molten UO_2 as far as thermal shock is concerned is not known. Experiments with molten steel-UO_2 mixtures were also performed at Aerospace Corporation, but the yields were too small to provide a good test of thermal shock resistance.

From many years of experience, the steel industry has learned by trial and error which compositions, grain sizes and densities provide the optimum for thermal cycling. With a controlled microstructure, better thermal resistance can be achieved. The industry experience has been that 98 percent pure MgO is more shock resistant than either 93 percent MgO or 100 percent pure MgO. A combination of grain sizes is used typically, with both larger and smaller grains present. It has been found that the best shock resistance is obtained with a mixture of 60 percent course grains (-0.5 inch down to -20 mesh) and 40 percent fine grains (-60 mesh down to micron size). The porosity is maintained in the region between 15-20 precent as fully dense MgO cannot withstand thermal shock. The pores are believed to serve as crack terminators. Chrome-periclase bricks have been found to have a better thermal resistance than MgO and are sometimes used in applications where thermal shock is felt to be important.

V.H.5 Brick Flotation

Since MgO bricks have a lower density than the molten core debris, it is possible that MgO bricks could be floated away by the more dense molten steel and UO_2. Consequently, there was interest in learning whether the steel industry has had any adverse experience with brick flotation. According to U.S. Steel, instances of brick flotation have been extremely rare. Usually an inverted arch design has been employed in refractory brick structures which has proved successful in the restraint of the bricks. Brick structures with flat bottoms and diameters in the 15-18 foot range have been constructed without the benefit of an inverted arch design. Flotation of bricks has generally not occurred with these designs either.

In some blast furnaces constructed 20 years ago, carbon refractories, less dense than MgO, were employed. The flat bottoms were 40 feet in diameter and were constructed with a tongue-in-groove design similar to the OPS proposal (see Figure 5) due to concern about the potential for brick flotation. No flotation problems were observed wtih this tongue-in-groove design.

V.H.6 Fusing of Joints

It has been suggested by representatives of Harbison and Walker Refractories and OPS that the joints between bricks in the core ladle will fuse together during contact with molten core debris. In the steel industry, with molten steel at 3000 degrees Fahrenheit, the joints remain after contact with molten metal. When the metal cools, the bricks are still individual bricks. Since the molten core debris would be at a much higher temperature, fusion of bricks may occur. The experience of the steel industry here is not applicable.

V.H.7 Slag Line Attack

The lining used in a BOF unit is determined by the chemistry of the expected slag. In the units at U.S. Steel, the slags are basic with lime to silica ratios greater than 1.5. Lime and MgO are added to the molten mixture to control basicity and slag attack; the MgO added assists in the saturation of the slag layer with MgO. MgO bricks are selected with lime to silica ratios similar to the slag for compatibility. A compatible brick is more resistant to slag line attack and is harder for the slag to wet.

Eighty percent of the liner of a BOF unit consists of "tempered" brick in which the MgO grains have been impregnated with pitch and baked at a low temperature (500 degrees Fahrenheit). The tempered brick contains 96-97 percent MgO and contains some intentionally added carbon to reduce penetration by slag. In addition to a 2 percent carbon contribution from the pitch, very fine carbon, totaling 2 percent, is added to the MgO. Although it is known that the carbon is effective, the reasons are not understood. According to one theory, the slag cannot wet or penetrate the MgO pores when carbon is present. It has also been suggested that the permeable MgO can volatilize, combine with carbon and redeposit forming an impermeable layer.

The remaining 20 percent of the liner consists of a more expensive "burn-impregnated" brick prepared by passing the MgO through a firing process prior to the addition of pitch. This material has a greater hot strength than the tempered brick and is used in regions where it will be subject to impact from falling iron scrap. The composition of the burn inpregnated brick is similar to the tempered brick. Again, carbon bearing material is added to the brick to reduce slag attack. The presence of carbon reduces the slag penetration from approximately two inches to a few millimeters per heat. Carbon cannot be employed in an oxidizing environment. The BOF environment is reducing due to the CO and CO_2 formed as carbon is oxidized.

In the BOF units at U.S. Steel, the slags contain a high concentration of iron oxides. The slag thickness tends to be 12-18 inches in a 40-48 inch deep layer of molten material. The slag is quickly saturated with MgO and penetrates the MgO liner to a depth of only a few mm. The usual operating temperatures is about 3000 degrees Fahrenheit. No heat is added to the molten metal in the BOF unit. Rather, energy is supplied by chemical reactions with the supersonically introduced oxygen. Oxidized particles of iron in contact with the MgO walls are thought to be at temperatures as high as 4500 degrees Fahrenheit due to the oxidation reaction. In this environment, the liners last for 1200-2000 heats until they eventually must be replaced due to slag attack. A typical heat lasts about one hour and thus a total of 24 inches of MgO are eroded over 1200-2000 hours.

It should be noted that slag is never completely eliminated. It will be present to some degree even when refined metal is heated. The slag from melting refined metal in air will amount to less than one percent of the total. In an environment containing water, or where air bubbles through molten iron, additional slag can be generated from refined steel.

As temperature increases, the amount of slag attack rises substantially. At 3200 degrees Fahrenheit, the highest tap temperature used in steel mills, the erosion rate is twice what it is at 3000 degrees Fahrenheit. The industry tries to avoid temperatures above 3000 degrees Fahrenheit because "breakouts," or leaks, in ladles and crucibles rise dramatically at higher temperatures. Also, continuous casting equipment is designed to handle lower temperature metal. With excessive superheating the skin on the castings will break out. In addition, alloys from superheated steel will not have the desired properties. The industry has no experience with near boiling iron in the temperature range that may occur in a core melt down event.

Harklase MgO is somewhat less slag resistant than the high lime to silica ratio MgO bricks used in the working liner of the ladle. However, the difference is small, on the order of 10 percent.

Chrome-periclase bricks are occasionally used in steel mills in place of MgO. These bricks are more resistant to thermal shock than MgO. They are also better insulators and have a lower thermal expansion constant and thus are better suited for certain applications, such as the roofs of electric furnaces. However, these bricks are considerable less resistant to attack by basic slags than is MgO. In a neutral or acidic slag environment, chrome-periclase bricks are more resistant to slag attack than MgO.

V.H.8 Hydration Resistance

When MgO is hydrated, it swells and becomes less dense. Substantial differences have been observed in the hydration resistance of various types of commercially available MgO bricks. These differences appear to be caused principally by the quantity of lime present in the bricks. In water, a high lime MgO brick will swell within 48 hours. Harklase, which is a high purity magnesite brick (~ 98 percent MgO) and which has a relatively low lime concentration, is considerably more resistant to swelling and will not swell after months of exposure.

Hydration resistance is a principal advantage of the high purity magnesite brick (such as Harklase). The superior resistance of Harklase to hydration was probably the motivating factor in its selection as the core ladle material. Hydration resistance is important in the FNP as moisture may accumulate in the reactor cavity area over long periods of time. In the steel industry, Harklase bricks are generally used only in the safety liners of ladles because of the relatively high cost of the brick. A ladle typically has 12 inches of Harklase as a safety liner next to the metal wall with a 24-27 inch thick working liner of a cheaper grade of MgO in contact with the molten steel and slag. Harklase is used for the safety liner so that the ladle can be sprayed with water to cool it quickly when it becomes necessary to reline the working liner.

The MgO bricks that have been impregnated with pitch tend also to be more resistant to hydration. Some of these bricks have been stored at U.S. Steel for periods as long as 5 years without becoming hydrated. Impregnation of bricks with boric acid has also been used to improve hydration resistance. Acids generally increase the hydration resistance of MgO bricks for reasons that are not understood.

V.H.9 The Effect of Molten Steel on Wet MgO Bricks

The effect of molten steel on wet MgO bricks is of interest because of the potential for water to enter the lower reactor cavity region, even though provisions for water removal are available. Although a steel shell will completely enclose the ladle, imperfections in the steel shell could admit water to the top layer of bricks.

Molten steel has been poured on wet MgO bricks at U.S. Steel without the serious consequences that might be expected. Steam forms in the brick and is driven back through the refractory. It leaves the ladles through vent holes or drains provided for this purpose at the bottom of the ladles.

In situations where molten steel is poured on top of pools of water in crucibles, trapping the water, serious consequences can result. Steam can be released with explosive force and can cause sprays of molten metal as it escapes from the molten pool.

At the subcommittee meeting of November 17, 1979, the ACRS expressed concern regarding the potential for water gaining access to the ladle region prior to the core melt material, and perhaps resulting in a steam explosion that could damage the ladle and reactor cavity structures. We consider it to be highly unlikely that large

quantities of water would be introduced into the ladle region prior to the core debris melting through the reactor vessel. For this to happen, a break would have to occur at a very specific location in the small length of piping inside the reactor cavity, close to the reactor vessel. Since this length of piping is very small compared to the total heat transport system piping, we consider the probability of a piping break inside of the reactor cavity to be very low. Additionally, to express the probability of this potential steam explosion occurring, the probability of a piping failure at a specific location has to be combined with (i.e., multiplied by) the probabilities of (1) a loss-of-coolant accident (LOCA), (2) failure of the emergency core cooling system (ECCS), and (3) a specific type of core melt sequence that results in a relatively large quantity of molten core material pouring in a coherent fashion into water, as contrasted with core melt scenarios where the core debris melts through the reactor vessel in a gradual manner and small quantities of core debris are introduced to the lower reactor cavity in a non-coherent manner. Therefore, we consider the combined probability of the above sequence of events to be very low. However, if this sequence did occurr, the potential exists for a steam explosion. As was discussed in the Liquid Pathway Generic Study (NUREG-0440, pages A-20 to A-24), the probability of a steam explosion occurring and also being able to damage the FNP barge hull is very low. We found that in order to get a high thermal to mechanical energy conversion that could damage the barge hull, the sequence of events would require (1) a coarse premixing of the core debris and the water masses with essentially no energy transfer, (2) an ideally timed trigger of sufficient magnitude to set off the explosion, and (3) extremely fine core debris fragmentation and interdispersion with water during the explosion propagation phase. Considering all of these factors, it is very difficult to see how all of these conditions can be met for a large fraction of the core debris material. However, as part of our longer-term effort associated with the final design evaluation of the FNP core ladle, we will assess whether steam explosions (taking into account the latest R&D in this area) could result in significant damage to the ladle and reactor cavity structures, such that the requirement of providing increased resistance to core melt-through is defeated.

V.H.10 Gas Evolution

Noncondensible gases are generated when molten steel is poured on pitch-impregnated MgO bricks. This may be of concern if OPS proceeds with plans to employ a top layer of shock resistant TOPEX-S brick. This material is impregnated with the waste liquor from paper mills and appears to generate hydrocarbon gases. This gas is at least potentially flammable and will increase the pressure in the containment building slightly (less than 1 pound per square inch gauge).

V.H.11 Conclusions

It should be noted again that the temperatures employed in the steel industry are 3000 degrees Fahrenheit, with an indication of the presence of 4600 degrees Fahrenheit iron oxide particles in contact with the MgO liner in the BOF unit. At the postulated maximum 5200 degrees Fahrenheit temperatures in a core melt down event, the rate of slag line attack will initially be much greater than at 3000 degrees Fahrenheit until the slag is saturated with MgO. At temperatures between 3000 degrees Fahrenheit and 5200 degrees Fahrenheit, the amount of MgO needed to saturate the slag will increase as indicated by the iron oxide-MgO phase diagram. It is this slag attack that is the main concern about the use of MgO in the core ladle and will be the subject of future research.

Several more positive points should be made concerning the selection of MgO as the material for the core ladle. The steel industry has been satisfied with the response of modern MgO refractory bricks to mechanical shock. The MgO liners appear to be reasonably resistant to thermal shock, even when cold. Brick flotation does not appear to be a problem. It appears that MgO bricks, such as Harklase, with a low lime concentration, are reasonably hydration resistant in water although measurement of hydration rate as a function of temperature and humidity must be determined to assure that the expected service life of the ladle can be achieved. Also, with proper

design, some superficial hydration of the MgO could be tolerated without significant deleterious effects, such as molten metal interactions with wet MgO bricks.

In summary, although there may be a slag attack problem with MgO, it still seems to be the best material for the core ladle application. If the amount of available MgO is sufficiently large compared to the slag layer, the slag layer will eventually be saturated with MgO. None of the other materials suggested for this application appear any more promising than MgO.

VI. RESEARCH AND DEVELOPMENT NEEDS

The staff concluded in Reference 2 (FES-III, NUREG-0502) that, following additional research, there is reasonable assurance that a material can be selected that will perform satisfactorily to substantially increase the melt-through delay time. This section delineates the informational needs which must be obtained through the conduct of appropriate research and development (R&D) prior to the staff taking approval action on the final core ladle design as part of construction of major elements of the FNP hull structure.

There is a scarcity of data in the area of material interactions between core melt debris and refractory sacrificial materials. Most of the work, References 11 and 12, in this area has been associated with studies of core retention systems (or "core catchers") for Liquid Metal Fast Breeder Reactors (LMFBR). The experimental studies so far have been confined mainly to transient pours of either molten steel or UO_2 onto single bricks or large crucibles composed of a refractory sacrificial material, such as magnesium oxide (MgO). There is essentially no data wherein the melt material (either steel or UO_2) was sustained in a molten state for a considerable period of time while in contact with a sacrificial material, such that quasi-steady state conditions are obtained which are nearly representative of nuclear decay heating. Such steady state conditions are required in order to adequately evaluate the uncertainties associated with the thermal, mechanical and chemical performance of various candidate sacrificial materials and to develop and/or validate proposed analytical models of melt front penetration into such materials. In addition, it is particularly important to perform a test that is a scale-model demonstration of an integral sacrificial bed design, as contrasted with separate effects tests coupled only by analysis. It is our understanding from reviewing the applicant's proposed R&D plans that scale effects are not considered to be important and will not be examined. The NRC plans to support some large-scale tests in this area. The staff will have to await the results of such tests before we can determine if the applicant's contention is correct.

In order to adequately address the technical uncertainties, we consider that the following specific informational needs should be obtained by the performance of R&D in the area of molten core debris materials interacting with the selected refractory sacrificial material:

(1) Rate and extent of core melt penetration into the sacrificial material, including penetration into cracks between bricks of the sacrificial material;

(2) Extent of thermal shock cracking and/or spallation of the sacrificial material;

(3) Quantity and composition of vapors and gaseous products released from the sacrificial material;

(4) High-temperature thermophysical properties, especially the lowest melting point eutectic temperature of the sacrificial material after exposure to core melt debris;

(5) Determination of any significant chemical interactions or dissolution processes, including characterization of the reaction products, phases present and extent of slag-line attack;

(6) Determination of a suitable layered brick configuration to prevent flotation of the sacrificial material by core melt debris and also withstand the mechanical impact if the lower reactor vessel head should drop;

(7) Measurement of the heat transfer split (upward/downward/sideward) from the core melt material.

The applicant plans on obtaining its R&D information needs through three sources: namely, (a) applicable experience and technology from the metals refining industry, (b) test programs already planned by the Nuclear Regulatory Commission (NRC), the Department of Energy (DOE), Electric Power Research Institute (EPRI), or in foreign countries related to interaction between molten oxides at high temperatures and refractory materials, and (c) additional testing as may be required to obtain information not available from the first two sources. The applicant will rely on existing sources of information to the extent possible both for satisfying requirements for information and for defining the scope of any future testing required.

Irrespective of where the research information is obtained, we must emphasize that it will be the applicant's responsibility to provide sufficient information to satisfactorily resolve the areas of uncertainty as specified above. This information shall be provided prior to the staff taking approval action on the final core ladle design as a prerequisite to the construction of major elements of the FNP hull structure.

The NRC Division of Reactor Safety Research (RSR) is supporting a confirmatory research program at Sandia Laboratories on materials interactions between core melt debris and containment materials, including candidate core retention system materials. Sandia researchers were requested to review Reference 3 (OPS Topical Report No. 36A59), and their comments are provided in Appendix G. The areas of technical uncertainty raised by Sandia were addressed by OPS in Revision 2 (dated September 21, 1979) of References 3 and will also be the subject of future research in this area. The suggestions for future evaluations made by Sandia, such as reviewing applicable experience from the glass-making industry, will also be taken into account as part of the final core ladle design and evaluation effort by both the applicant and the staff. It is the staff's judgment that the Sandia comments do not negate the feasibility of designing a core ladle which will provide significant delay times before the core melt debris penetrates the FNP barge structure.

VII. REMAINING TECHNICAL ISSUES

This section provides a discussion of each of the remaining technical issues which need to be resolved as part of OPS's final core ladle design effort and further research and development in this area. Each of these areas should be evaluated and included as part of OPS's final core ladle design report, which requires NRC approval prior to start of construction of major elements of the FNP hull structure.

VII.A Ladle Instrumentation

As indicated in the staff's recent meeting summary with OPS, Appendix B, we believe that it is highly desirable to provide instrumentation in the core ladle region. The instrumentation would consist of moisture monitors and thermocouples.

Although the core ladle is entirely enclosed in a steel shell to prevent water from coming into contact with MgO bricks during normal operation, the staff is concerned about the effects of water moisture on the performance of MgO refractories, including what can be done to minimize degradation. Our concern stems primarily because of

the existence of the potential for either high atmospheric humidity conditions or water getting into small cracks and flaws in the steel liner. Magnesia will hydrate quite rapidly under steam conditions. Under ambient conditions and moderate humidity, magnesia brick normally hydrates quite slowly. Hydration will result in expansion and swelling. Bricks have been stored inside for several years with no significant effects. MgO bricks can be treated to improve hydration resistance. Further research into the hydration related effects, including the effects of treatment to improve hydration resistance, will be necessary before the staff can take a final position on what the necessary environmental control shall be. The applicant has committed to evaluating the need for moisture monitors, and if they are found not to be necessary, sufficient justification will be provided.

With respect to measuring temperatures in the core ladle region, OPS plans to install thermocouples on the very outside top and bottom surfaces where they would be accessible for replacement. The staff believes that it would be highly desirable to have thermocouples installed inside the core ladle MgO to monitor the approximate extent of the core melt front penetration. We consider it important to know about when melt-through was going to occur in order to take appropriate actions. OPS has indicated concern as to whether the thermocouples would last for 40 years and whether they could be replaced. It is not clear to the staff that the thermocouples either would not last for 40 years, or that it would be very difficult to replace them. The applicant has committed to evaluating the need for installing thermocouples inside the core ladle; if this is found not to be necessary, sufficient justification will be provided. There may be a problem related to reading the thermocouples in the control room because of high radiation levels associated with the core melt debris being in the ladle. This will be investigated further as part of the final design effort.

VII.B Slag-Line Attack

Slag-line attack, which is the preferential attack of MgO by an iron oxide slag layer on top of molten iron, has been observed in steel making operations. In some furnaces like the blast furnace, the slag layer is about one-third of the total height (or approximately 1-1/2 to 2 feet) of the molten material height and increased erosion by about 25 percent at the slag-line has been observed. Discussions with U.S. Steel have indicated that for a 24 to 27 inch MgO lining in a basic oxygen furnace, it would last about 3000 heats (at 1 hour each) or 3000 hours of operation before slag line erosion would have to be repaired. It may be that a special brick composition, such as one composed of chromium, alumina and magnesia, could be used in the ladle design at the slag line to mitigate this particular erosion mechanism. Another possibility would be to increase the lateral thickness of the ladle to compensate for slag-line attack. More quantitative information on the extent of slag-line attack of MgO will be required.

The three principal oxides formed during a core melt accident are UO_2, ZrO_2 and iron oxide. The phase diagrams for such a three component oxide mixture interacting with MgO are unknown and could be quite different than that for the simpler iron oxide - MgO system. Also, the potential exists for formation of different lower melting point eutectics. Since no direct evidence exists for the three component core melt oxide mixture interacting with MgO, this will need to be the subject of future research.

VII.C Core Ladle Configuration

Pursuant to a staff request (see Appendix B), OPS has proposed a deeper core ladle cavity configuration that would result in increased MgO sidewall thickness and decreased MgO floor thickness as compared to the previous configuration. This design which increases the sidewall thickness of MgO should also increase the melt-through delay time because it provides greater margin for: (1) lateral melt front erosion which is likely to be equal to the vertical erosion, (2) slag-line attack, (3) barge list or tilt, (4) increased quantities of core melt debris, and (5) heat being generated primarily in the oxide phase and not the metal phase which is likely to be

on the bottom. At the November 17, 1979 meeting of the ACRS subcommittee, OPS provided their latest core ladle design configuration (see Section IV) which has a deeper cavity and increased sidewall thickness of MgO. OPS has considerable flexibility to change the configuration of the core ladle as the design information develops; the only constraint on the ladle configuration is the present outside dimensions, which are determined by the surrounding structures.

The staff has also requested (see Appendix B) OPS to examine further the need to protect the upper reactor cavity walls with MgO so that melting of the concrete would be precluded for times comparable to the ladle melt-through delay time. As previously mentioned, the heat transfer calculations performed by OPS for the upward and downward heat transfers were made separately and independently from one another. Future analyses will couple the upward and downward heat transfers from the pool, including an allowance for heating up the reactor cavity structures. Recent coupled calculations made by the staff are showing that the upper reactor cavity will rapidly heat up and about 50 percent of the core melt decay heat will be deposited in the reactor cavity walls at the time of core ladle melt-through (about 6 days). For this amount of upward heat transfer, the staff has calculated that about 3 feet of MgO on the inside of the upper reactor cavity surface may be necessary to prevent (1) the concrete from melting, and (2) the primary steel structures from reaching 1000 degrees Fahrenheit for 2 days. As part of the detail design, OPS has committed to evaluating the need to protect the upper reactor cavity concrete with sufficient MgO or other refractory materials, such that the primary steel structures necessary to maintain the integrity of the reactor cavity or necessary to support the core ladle will remain intact during the core debris retention period. In addition, as part of more detailed heat transfer analyses, OPS will examine other core ladle configurations to determine if the melt-through delay time can be extended.

VII.D Flooding Core Melt With ECCS Water

Although the sump in the FNP containment is higher in elevation and is isolated from the lower reactor cavity by a wier of sufficient height that will preclude overflowing of sump water into the region below the reactor vessel, the potential exists for sump water to reach the lower reactor cavity by being pumped by the ECCS recirculation system into the reactor coolant system where it can exit out the reactor vessel melt hole following a core meltdown event. Specific reactor operator action would be required to secure the ECCS recirculation system. Any ECCS water which is introduced into the core ladle region following a meltdown should only come into contact with the core melt debris and not with the MgO.

There are advantages and disadvantages of pumping ECCS water on top of the core melt debris. Pumping water on the core melt debris will dissipate a considerable amount of heat by evaporation of water, resulting in less energy being available to erode the MgO bed in the downward direction. This would serve to increase the melt-through delay time. In order to be conservative with respect to melt-through delay time, OPS and the staff have only considered dry core melt scenarios where the heat transfer in the upward direction is primarily by radiation.

On the other hand, pumping ECCS water on top of the melt creates the potential for a steam explosion (see also Section V.H.9). More importantly, however, the pumping of ECCS water on top of the melt would probably result in significant sparging of activiity from the core debris because of the copious quantities of steam which would be generated by the interaction. The sparged activity from the core melt debris would be released into the upper containment building where it can be subsequently released via cracks in the containment building (assuming the upper containment has failed earlier in the accident scenario by, for example, hydrogen burning). This additional activity from the core melt debris would increase the airborne releases (see also Section V.F). In addition, the sump water, which is the supply for ECCS recirculation, would contain a significant amount of activity. Following melt-through penetration of the core ladle and barge structure, continuing to pump sump water would result in significant releases to the liquid pathway as discussed in Reference 1, the Liquid Pathway Generic Study,

and Reference 2, FES-III. However, the significant hold-up time provided by the FNP core ladle would provide sufficient time for operator action to prevent sump water release, thereby reducing or eliminating the release of a large source of radioactivity to the hydrosphere.

The staff's position on the desirability of pumping ECCS water on the core melt debris will be made as part of the final core ladle design evaluation when more analysis and research information will be available to better judge the advantages and disadvantages of this action.

VIII. CONCLUSIONS

Based on our analysis of the proposed FNP core ladle design as described in Reference 3, OPS Topical Report No. 36A59, dated April 1979, including Revisions 1 and 2 and the material presented at the November 17, 1979 ACRS subcommittee meeting, we conclude that the applicant has satisfactorily met the staff's Condition 4 of FES-III (NUREG-0502, dated December 1978) of providing a pad constructed of magnesium oxide, or other equivalent refractory material, that will provide increased resistance to melt-through by the molten reactor core in the event of a highly unlikely core melt accident, that will not react with core debris to form a large volume of gases and that will not have any deleterious effects on safety. We also conclude that the core ladle design concept as proposed by OPS is feasible and can be engineered to provide retention of molten core debris for a period of time in the range of 2 days to 1 week. Furthermore, we find that, although recognizing the uncertainties and complexities of fission product behavior during core meltdown events, fission product evaporation, sparging, and subsequent release to both the air and liquid pathways are expected to be less with an MgO core ladle than with concrete. As a condition to a manufacturing license, we will require that the applicant obtain NRC approval of the final core ladle design prior to the start of construction of any major element of the FNP hull structure.

IX. REFERENCES

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APPENDIX A

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 9, 1978

Mr. Lee V. Gossick
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: LIQUID PATHWAY GENERIC STUDY

Dear Mr. Gossick:

During its 216th and 217th meetings, April 6 and 7, and May 4 and 5, 1978, the Advisory Committee on Reactor Safeguards reviewed the Liquid Pathway Generic Study (NUREG-0440), which discusses the impacts of accidental radioactive releases to the hydrosphere from floating and land-based nuclear power plants. NUREG-0440 was also considered during a Subcommittee meeting in Washington, DC on March 22, 1978. This matter and an earlier draft version of this report have been the subject of ongoing review by the Committee since 1975. During its review, the Committee had the benefit of presentations by representatives of the NRC Staff and Offshore Power Systems (OPS).

Although the discussions at these meetings indicated that there are still areas of disagreement between the NRC Staff and OPS on specific details of the various assumptions entering the calculations, it appears that a reasonable perspective has been developed on the range of potential accidents, the identity and sources of the critical radionuclides, and the accompanying dose patterns and consequences. The current study also appears to provide a basis for evaluating possible design modifications. Although the Committee recommends that the NRC Staff and OPS continue to work to resolve disagreements and to eliminate weaknesses in these assessments, the ACRS believes that the perspective provided by the study can serve as a useful basis for current evaluations of the health and safety risks of radionuclide releases to the hydrosphere from land-based and proposed floating nuclear power plants.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

APPENDIX B

SACRIFICIAL MATERIALS TO DELAY CORE MELT-THROUGH IN A FLOATING NUCLEAR PLANT

E.1 Introduction

The proposed floating nuclear plant (FNP) design developed by Offshore Power Systems consists of a 1150 MWe pressurized water reactor power plant mounted on a large floating platform. The platform has been designed to be sited at offshore and shoreline sites that meet specific site envelope parameters. In the unlikely event of a core meltdown accident, molten core debris could melt through the reactor vessel and through the structure of the platform. A 4-foot thick concrete radiation shield and the steel hull of the platform are the last barriers that the molten core debris would encounter before entering the surrounding waterbody. On the basis of analyses (References 1 through 5) and experiments (References 6, 7, 8), it appears that the penetration of the molten core debris into this concrete layer would be quite rapid. Concrete penetration rates as high as 130 cm/hr (4.3 ft/hr) have been reported for high-temperature (2800°C) molten steel. Unless design changes are made, molten core debris from a meltdown accident would be released to the surrounding waterbody within hours. A delay in the penetration of molten core debris would provide time so that steps might be taken to mitigate the radiological consequences of such an event. The purpose of this assessment is to determine the feasibility of achieving significant delay times through the replacement of the 4-foot thick concrete radiation shield with a layer of a sacrificial material, such as magnesium oxide.

E.2 Materials Considerations

There are a number of considerations in the choice of a sacrificial material to delay melt-through by a molten core. Desirable characteristics for a sacrificial material include the following:

(1) High melting point - This property will slow down the advance of the melting front of core material into the sacrificial bed.

(2) High specific heat and heat of fusion - These properties will slow down the advance of the melting front by absorbing heat from the molten core debris. The heat absorbing capacity of the material can be estimated from the sum of the sensible and latent heats; this combination is sometimes referred to as the volumetric heat absorbing capability of the material.

(3) Low thermal conductivity - This property will slow the advance of the melt front into the sacrificial material and will decrease the heat load on the outer extremities of the sacrificial bed.

(4) High density - If the sacrificial material has a high density, it will tend to resist floating to the surface if core material gets under it by way of cracks, seams or flaws.

(5) Miscibility of molten fuel and sacrificial layer - If the fuel and sacrificial materials are miscible, the fuel volumetric heating rate will be decreased substantially by dilution. If the fuel and sacrificial materials are not miscible, a high density sacrificial material which would not float on the surface is preferred.

(6) Chemical compatibility with molten core debris - The sacrificial material should not react chemically with molten UO_2 or steel. It is especially important that no significant exothermic reactions occur between the sacrificial material and the molten core debris in order to keep the chemical heat addition to a minimum. It is also important that the sacrificial material not form very low melting temperature compounds when exposed to the molten core.

(7) No gases should be generated - Gaseous reaction products, such as carbon dioxide, carbon monoxide, and hydrogen, should not be generated by chemical reactions between the molten core debris and sacrificial materials. The generation of these gases can create an explosion hazard, increase the sparging of fission products from the core melt debris, and increase the gas available to sweep activity out of the failed containment.

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

(9) Pre-accident stability - The sacrificial bed should be stable in the environment of the reactor cavity for the expected life of the reactor plant.

(10) Limited neutron moderation - The sacrificial material must not increase the potential for recriticality of the core material. Poison materials, such as tantalum and boron, could be added to alleviate this concern.

(11) Low cost and availability - It is highly desirable that the sacrificial material be of low cost and that a technology exist for manufacturing mechanically stable structures, such as bricks, from this material at reasonable cost.

E.3 Available Materials

Of known high-temperature materials, relatively few have been studied sufficiently to accumulate an extensive body of engineering data. While in many cases there is insufficient

experimental data to make generalizations, it can be stated that oxides are more likely to be stable chemically in the presence of molten UO_2 than carbides, borides and nitrides. These latter materials are likely to oxidize, especially in the presence of excess oxygen. In general, oxides have only minor chemical reactions with molten core debris. The oxides typically form eutectic mixtures with UO_2 which have melting points below that of either pure constituent. The oxides of magnesium (MgO) and aluminum (Al_2O_3), which are used for crucible liners in the steelmaking industry, are excellent for the containment of molten steel. Considering the desired material characteristics discussed in the preceding section, a few of the better refractory materials are discussed in the following paragraphs.

Magnesium oxide (MgO) has the advantages of a very high specific heat (0.31 cal/g°C), a high melting point (2850°C), is miscible with UO_2, is stable with respect to molten UO_2 and steel, is easy to fabricate, and has a relatively low cost. In large quantities (lots of 1000 bricks), Harklase MgO bricks are available at a cost of approximately $1.80 each.

The principal disadvantage of MgO, which is also a disadvantage for titanium carbide (TiC), aluminum oxide (Al_2O_3), and graphite, is that it has a low density compared to UO_2 and would tend to float in the molten core debris. Some method has to be developed to prevent the solid MgO from floating away in molten core material. The MgO could probably be held in place by constructing a bed of many layers of bricks. The bricks could be wedged together to form an inverted spherical arch with an interlocking, tongue-in-groove design.

Graphite has the advantages of a high sublimation temperature, good specific heat, easy fabrication, and low cost. It has two serious disadvantages, however. It acts as a neutron moderator and thus increases the potential for recriticality. The reactions of graphite with UO_2, generating CO, CO_2 and UC, are not well understood and are potentially serious problems

Titanium carbide (TiC) has a very high melting point (3076°C), good specific heat (0.21 cal/g°C) and is available at reasonable cost. However, it has a low density, marginal chemical compatibility with respect to molten UO_2, and it is difficult to fabricate, requiring high temperature inert atmosphere facilities for the production of bricks. These production facilities do not currently exist.

Zirconium oxide (ZrO_2) has the advantages of a high melting point (2730°C) and density. It is chemically stable with respect to molten core debris. Its disadvantages include a low specific heat, structural instabilities, and high cost.

Aluminum oxide (Al_2O_3) has a high specific heat, is chemically stable with respect to molten core debris, and is available at relatively low cost. However, it has a low density and melting point (2037°C).

The oxides of uranium (UO_2) and thorium (ThO_2) have very high melting points and low thermal conductivities. There are no problems with chemical compatibility. A sacrificial bed of UO_2 eliminates the problem of low melting temperature eutectic formation. These oxides are

poor materials from which to fabricate bricks or other structures. Usually, they must be sintered or mixed with other materials that act as binders and even then they are quite fragile. Use of UO_2 or ThO_2 may be prohibitively expensive.

E.4 Magnesium Oxide

After considering a number of sacrificial materials in view of the design requirements, favorable materials considerations and results of compatibility tests with molten UO_2 and steel, MgO appears to be an excellent candidate material. It is possible that some lesser known materials might serve as better sacrificial layers, but their identification would probably require an extensive research and development program. It is not obvious at this time that any other material would offer any significant advantages.

Table E-1 lists the properties of a good quality MgO brick; it is extremely important to obtain a high purity MgO material (~ 98% MgO) because high impurity levels in the MgO can significantly degrade its high temperature performance. This particular MgO material was manufactured by Harbison and Walker Refractories and is sold under the name "Harklase." The advantages of MgO, as mentioned earlier, include a high melting temperature, high heat capacity, low thermal conductivity, high degree of chemical compatibility, resistance to thermal shock, ease of fabrication, and low cost. There are no adverse chemical reactions, and there is limited gas evolution, consisting mainly of gas forced out of the pores in the material. MgO has been used in the steelmaking industry for many years and, consequently, its properties at high temperatures are relatively well understood.

MgO (melting point ~2800°C) and uranium oxide (melting point ~2850°C) are miscible in the liquid state and form liquid solutions with a eutectic composition of about 50 mol % MgO in UO_2 (13 wt.%). The reported eutectic melting temperatures range from about 1800°C to 2300°C, with the higher temperatures occurring in oxygen-free surroundings.

Some differences have been observed in the rate of solubility of UO_2 into MgO. Meacham (Reference 9) reported formation of a UO_2-MgO eutectic at 2375°C with rapid, complete solution of all of the UO_2 present in three minutes. The UO_2 and stainless steel in his experiment were placed in a cavity only 0.64 cm in diameter by 1.27 cm deep. In his relatively small experiment, diffusion and stirring effects would not be expected to be important.

In a larger scale experiment, Stein, et al. (Reference 10), reported that part of a magnesia specimen dissolved smoothly into molten UO_2 in the range 2200°C to 2300°C over a period of 26 minutes. In this experiment, 9.5 kg of UO_2 was used and the MgO specimen was a rectangular solid with a thickness of 2.5 cm and a length and width of 10 cm.

In experiments at the Aerospace Corporation, Swanson, et al. (References 7, 8), found that the rate of erosion of MgO seemed to be influenced by the degree of stirring of the molten UO_2. Penetration of 0.6 cm in 5 minutes was observed when 100 grams of UO_2 were vigorously stirred by an electric arc. Under more quiescent conditions, with 2 kg of UO_2 on an 8.9 cm diameter cylinder with a thickness of 7.6 cm, erosion occurred much more slowly with a

TABLE E-1

CHARACTERISTICS OF HARKLASE BRICK
(Measurements Taken from Reference 7)

Quantitative Chemical Analysis		Semiquantitative Chemical Analysis	
Constituent	Percent*	Element	Percent
SiO2	0.7	Al	0.1
Fe_2O_3	0.4	Ca	0.1
MgO	96.3	Mn	0.01
SO_3	0.03	Ti	0.05

Bulk density = 2.72 g/cm^3

Average void volume = 16.94%

Melting point: 2850°C

Specific heat: 0.31 cal/g°C

*Information provided by the manufacturer, Harbison and Walker Refractories, as compared to the measurements reported in Reference 7, is as follows:

Information Provided by Manufacturer

Chemical Analysis

Silica (SiO_2) = 0.8%
Alumina (Al_2O_3) = 0.4%
Ironoxide (Fe_2O_3) = 0.4%
Lime (CaO) = 0.6%
Magnesia (MgO) = 97.9%

Reported Density = 2.84 to 2.90 g/cm^3

Reported Appar[illegible] Porosity = 15.5 - 19.0%

Measured Bulk D[illegible]ity = 2.849 g/cm^3

Measured Porosity = 16.94%

diffusion coefficient of only 1.5 10^{-5} cm^2/sec. In the latter experiments, the UO_2 was cooler than in the vigorous stirring experiment, but it is believed that it was molten in the region in contact with the MgO. The differences between the experiments seems to be a consequence of stirring in the first experiment bringing fresh molten UO_2 in contact with the MgO.

In both of the Aerospace experiments (References 7, 8), three layers were observed consisting of a mixture of a eutectic composition of UO_2 and MgO in UO_2, a layer of UO_2 attacking the MgO binder and grains, and finally a layer of MgO. Molten UO_2 preferentially attacked the low melting phase binding the MgO grains and then formed a hypoeutectic mixture with melted MgO.

When molten UO_2 was poured on a rectangular MgO brick (9 in. x 4-1/2 in. x 2-1/2 in.) with a cylindrical cavity (3-1/2 in. diameter x 2 in. deep) in the Aerospace experiments to simulate thermal shock, there was little sign of interaction aside from a discolored region extending 1 cm into the brick. Thermal shock cracks that formed on cooling, after solidification of the UO_2, extended 2 cm into the material. Although MgO is not usually considered a thermal shock-resistant material, under these conditions it performed acceptably.

It should be noted, however, that the MgO brick was coated with UO_2 dust, vapor-deposited by the electric arc employed to heat the UO_2. This coating may have absorbed some of the thermal shock and heat. The possibility of such an effect suggests that placing layers of powders or gravel composed of such materials as MgO on top of a layered brick configuration could be used effectively to absorb the initial thermal shock.

Under the experimental conditions of the Aerospace experiments, the MgO performed adequately as a container material for molten UO_2. There was some attack, but it was relatively minor, and thermal shock effects were small.

As previously indicated, the low density of MgO relative to molten UO_2 gives rise to the possibility that molten UO_2 may undermine a sacrificial layer by flowing under the bricks and floating them away. However, in typical installations, such as in steel mills, the bricks are placed with spaces between them to allow for thermal expansion. When hot metal comes in contact with the bricks, the bricks expand forming a tight structure which the metal is unable to penetrate. Consequently, movement of molten metal through the joints into successive layers of bricks is not considered to be a problem.

Design of a sacrificial bed composed of layers of these bricks may be complicated by the variation in the temperature of the constituents of the molten core debris. Normally, in the design of a steelmaking furnace, the joint spacing is designed to contain a hot material at one specific temperature. In order to retain molten core debris, the joint spacing between bricks must be large enough to accommodate expansion of the bricks at the highest expected temperature, probably about 2800°C, the temperature of molten UO_2. However, some constitutents of the molten core debris may be much cooler, consisting, for example, of molten iron and solidified UO_2. Such a mixture might leak through a structure with joints

designed to retain molten UO_2 and cause flotation of the bricks. On the other hand, if the joints are spaced with lower melting core debris in mind, the core retention structure may be damaged by thermal expansion if higher than expected temperatures are encountered. A design consisting of many layers of bricks wedged together (such as by a "tongue-in-groove" design) into an inverted spherical dome would probably ensure that they remain in place. Further investigation is needed in this area.

With any core retention material, there is a question about what should be done at locations where penetrations are required or at corners, edges, etc. A mortar, "Oxybond," has been developed for use with "Harklase" MgO bricks in high-temperature applications that should work well in these regions, based on information supplied by the vendor. The mortar is not intended for use between bricks. Introduction of mortar into the joints between bricks would cause the brick structure to be destroyed by the pressure generated during thermal expansion.

Oxybond has a somewhat lower melting temperature than MgO (about 50°C lower) and is made from a dry mix consisting of hard fired magnesia fines, or from crystals, below Number 325 mesh (.0017 in). Hard fired MgO is material that has been heated to the point where amorphous MgO is converted to small periclase crystals. Oxybond sets in place or sinters when heated to above 2000°C. Since the mortar employed around any penetrations has a lower melting point than the MgO bricks, it would be preferentially attacked by hot core debris. Consequently, minimal use should be made of the mortar and penetrations in the structure should be kept to a minimum. No experiments have been performed to date with molten UO_2 and this mortar.

E.5 Melt-Through Time Estimates for a Sacrificial Bed of MgO

A thermal analysis was performed to estimate the time for a molten pool of core debris to penetrate a sacrificial bed composed of MgO. The calculations were performed to indicate the holdup time that could be afforded by installing a sacrificial MgO bed to cope with a core meltdown accident in a floating nuclear plant (FNP).

The analysis was done for a rectangular MgO cavity located in the 20 foot by 30 foot reactor cavity area below the reactor vessel in the FNP. It was assumed that the MgO material replaced the existing reactor cavity floor concrete; however, on the reactor cavity side walls, MgO material was added to the inside of the existing concrete walls up to a height of 14 feet. A molten pool mass of 300,000 pounds was assumed (215,000 pounds UO_2, 40,000 pounds steel, and 45,000 pounds Zircaloy) with an average density of 555 lb/ft^3 resulting in an initial pool volume of 541 ft^3. The time-dependent integrated decay heat source used in the analysis is provided in Table E-2. The calculations were based upon assuming that 30% of the core decay heat is associated with noble gases and volatile fission products, so that only 70% of the decay heat is available from the core melt material.

Further simplifying assumptions are listed below; these were made to allow calculations to be performed by hand and to bracket the directional heat transfer uncertainties that presently exist in this area:

TABLE E-2

INTEGRATED DECAY HEAT LEVEL (Based On 3400 MWt)

Time	Decay Heat
(Hours)	(Btu x 10^8)
1	2.325
2	3.822
3	5.105
4	6.279
5	7.372
6	8.398
7	9.368
8	10.290
9	11.170
10	12.020
(Days)	(Btu x 10^9)
1	2.190
2	3.584
3	4.814
4	5.936
5	6.983
6	7.976
7	8.926
8	9.843
9	10.730
10	11.600

(1) Changes in the sensible heat of the molten pool were neglected.

(2) The melting process is ablative; that is, heat is not conducted away from the melt front into the MgO.

(3) The sidewards and downwards heat flux from the molten pool were equal.

(4) The sensible heat plus the latent heat of fusion for MgO was taken as 4.5×10^5 Btu/ft^3.

(5) Three directional heat split ratios were considered; namely, 20%, 50% and 70% of the available decay heat from the core melt debris was directed into the MgO (sidewalls and floor); the balance was lost in the upward direction.

The understanding of the heat transfer characteristics for a molten fuel pool melting into a sacrificial bed material is a developing area of knowledge. Previous heat transfer studies in this area, such as those in References 11 through 15, indicate that a range of downward energy transfer (i.e., into MgO) from 20% to 70% should bracket the present heat split uncertainties.

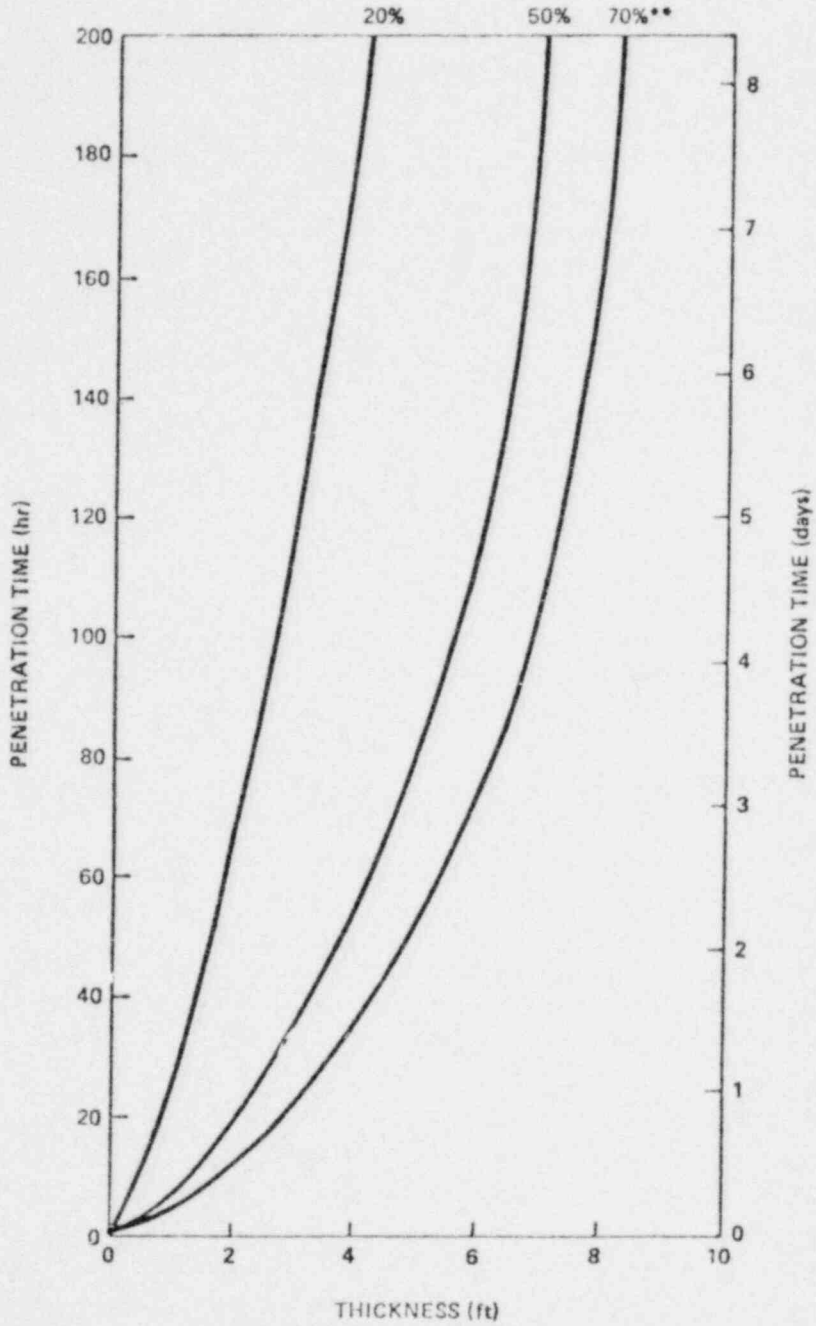
Calculations were performed using the above heat splits for a range of sacrificial bed thicknesses (the MgO cavity sidewalls and floor thicknesses were assumed to be equal). The penetration times as a function of the MgO bed thickness are shown in Figure E-1 for various heat splits. For example, the results show that if 50% of the core melt energy (or 35% of total decay heat) is directed into the MgO, a bed thickness of 4 feet would delay core melt-through by about 2.3 days. If only 20% of the core melt energy is directed into the MgO, a 4-foot bed thickness would delay melt-through by about 8 days. However, if 70% of the core melt energy is transferred into the MgO, then 4 feet of MgO would only delay melt-through by about 1.5 days.

The use of up to an 8-foot thickness of sacrificial material in the lower reactor cavity region appears to be feasible without significantly affecting the FNP barge design. For this thickness, melt-through is expected to be delayed by at least one week.

In addition to hand calculations, predictions of core melt penetration into a sacrificial bed of MgO were made with the use of the GROWS computer model (Reference 14). This code accounts for the spatial and time dependent variation of thermophysical parameters as the melt front progresses. Presently, the code is only able to handle a cylindrical geometry, so that direct application to the FNP rectangular reactor cavity could not be made. However, for the same sacrificial bed thickness and floor area, the results for penetration time as determined by the GROWS code and hand calculations (for a rectangular cavity) were found to be in reasonable agreement.

FIGURE A-1

MgO Sacrificial Bed Penetration Time vs. Bed Thickness*



*Assumes 30% of the core decay heat is associated with noble gases and volatile fission products so that only 70% of the decay heat is available from the core-melt material.

**Percent values represent fraction of total melt energy (70%) directed into MgO, e.g., for the 70% case, 49% of the total core decay heat is directed into MgO.

E.6 Conclusions

In the unlikely event of a core meltdown accident in an FNP, for the present design utilizing a thin (4-foot) nonstructural concrete pad below the reactor vessel, the molten core debris would very quickly (within a few hours) penetrate the concrete layer. However, with the use of a sacrificial refractory material such as MgO in place of the concrete, it is estimated that several days would be required before melt-through penetration occurs. This time delay could allow for steps to be taken to mitigate the radiological consequences of such an event. Furthermore, as previously indicated, MgO is relatively inexpensive (it is estimated that an MgO sacrificial bed in an FNP would cost approximately one million dollars), readily available and easily fabricated. Its properties are relatively well understood. Although additional research would be required, MgO appears to be an excellent choice among the available candidate sacrificial materials when all of the design and material considerations are taken into account.

On August 29, 1978, the NRC staff organized and held a meeting with experts in the field of sacrificial materials and core meltdown. The purpose of the meeting was to provide the NRC staff with relevant information and expert opinion on the use of sacrificial materials to delay melt-through in the event of a core meltdown accident in a floating nuclear plant. The staff's meeting summary is provided as an enclosure to this appendix. Based on an evaluation of the information presented at this meeting, the staff reached the following conclusions which are reaffirmed herein... "There are a variety of materials available which might be used to replace the concrete pad below the reactor vessel in the FNP. Some of these would provide a significant increase in the time required for the core debris to melt through to the bottom of the barge. There is, however, no clear choice of one material which is best in all respects, and there are important questions about the physical, chemical and mechanical properties of all of these materials at these extreme temperatures. Additional research is required before the staff could determine the suitability of a particular material. We conclude that there is reasonable assurance that a material can be selected (following additional research) that will perform satisfactorily to substantially increase the melt-through time and reduce the airborne release by reduced gas generation in the melt. We believe that these materials can be incorporated in the FNP design as it exists or with minimum alterations."

Enclosure 1

to Appendix E

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SEP 29 1978

Docket No. STN 50-437

APPLICANT: Offshore Power Systems

FACILITY: Floating Nuclear Plants 1-8

SUBJECT: SUMMARY OF MEETING HELD WITH EXPERTS ON THE USE OF SACRIFICIAL MATERIALS TO DELAY CORE MELT-THROUGH IN A FLOATING NUCLEAR PLANT (FNP)

On August 29, 1978, members of the NRC staff met in Bethesda, Maryland with a number of experts in the field of sacrificial materials and core melt down. The purpose of the meeting was to provide the NRC staff with relevant information and expert opinion on the use of sacrificial materials to delay melt-through in the event of a core melt down accident in the Floating Nuclear Plant. This information pertains to the requirement (ref.: Draft Environmental Statement, Part III, NUREG-0127, May 1978) by the NRC staff that the concrete base mat beneath the reactor vessel of the FNP be replaced by a material which would increase resistance to a melt-through by the molten reactor core and which does not react with core debris to form a large volume of gases. A list of the meeting attendees and the agenda are attached as enclosure 1.

BACKGROUND

The meeting began with introductory remarks by members of the NRC staff. Dr. Speis opened the meeting and explained that the objective was an exchange of technical information, and that there would be no policy decisions made as part of the meeting. Mr. Chipman presented the staff's viewpoint and a short history of the review of FNP by the NRC staff. The pre-application review was initiated in 1972. The staff concluded that, in accordance with its responsibilities under the National Environmental Protection Act, the consequences of a core melt accident might be significantly different for the FNP than for a land based plant. As a result, a detailed study of the comparability of the consequences of accidents in the liquid pathway for floating and land based plants was undertaken. This led to the Liquid Pathway Generic Study (NUREG-0440) which the staff completed in February 1978.

The study found that the risks associated with releases to the hydrosphere at an FNP are greater than those for a land based plant (LBP) for core melt accidents. The staff then asked the applicant to make design changes in the plant to mitigate the consequences of this kind of accident; specifically, the staff in the Draft Environmental Statement, Part III (NUREG-0127), May 1978, and in a subsequent letter to Offshore Power Systems (OPS) (R. P. Ballard to A. P. Zechella, July 25, 1978) requested that the concrete pad beneath the reactor vessel be replaced by some material that provides increased resistance to a melt through by the reactor core. This letter is attached as enclosure 2.

Mr. Marchese of the staff outlined specific objectives for the meeting. These objectives included determining the present state of the technology of sacrificial materials, reviewing the present research efforts in the field, and attempting to develop appropriate criteria for judging the effectiveness of a sacrificial material in this specific application. Mr. Marchese also provided a list of items which should be considered while the technical presentations are being made and is included as enclosure 3 along with a sketch of the approximate geometry of the space below the reactor vessel.

DISCUSSION

A brief summary of the presentations made by each of the participating organizations is provided in the following paragraphs.

1. AEROSPACE CORPORATION, DR. D. G. SWANSON

Dr. Swanson discussed a series of experiments that were conducted by Aerospace Corporation during 1976-1977. In these experiments UO_2 and/or stainless steel were heated to their melting point in an electric arc furnace and then poured onto the sacrificial test material's surface. The tests were done with Harbison and Walker's "Harklase" MgO brick and two types of mortar. The composition of these bricks and mortars are included as enclosure 4 along with photographs of the experimental apparatus.

Their large arc furnace can melt about 3 kg of a UO_2/steel mixture. The melt is then poured from a graphite crucible on to the test specimen. Sustained heating experiments in which case the melt was maintained for about 30 minutes, were also conducted. These experiments showed very little attack of molten UO_2/steel on MgO. The initial thermal shock was found not to be a problem during the pour, but the bricks did show signs of cracking as they cooled when the pour was over. Dr. Swanson pointed out that constructing

layers of bricks may be a problem. Traditionally the bricks, which are used extensively in the steel industry, are stacked so that the gaps between them will close when they reach a certain temperature. In the FHP application though, the sacrificial material could be subjected to a range of temperatures, and gaps may reappear as the mixture cools allowing material to flow under the bricks and float them out. Based on the thermal and physical properties and their test experience, Swanson's preliminary conclusion was that this specific type of MgO brick would make a satisfactory material provided that some adequate method could be found to construct the bed. He cautioned, however, that there are many types of MgO bricks and particular attention must be paid to the impurity level.

2. ARGONNE NATIONAL LABORATORY, DR. L. BAKER

Dr. Baker described the principal activities in the ANL program. These include prototypic material experiments, analytical development work, simulation experiments, and modeling. The prototypic material experiments are conducted in a cube shaped pot (shown in the second figure of enclosure 5). The material is heated by direct electrical conduction through the tungsten electrodes. This device can hold UO_2 above its melting point for indefinite periods.

Dr. Baker made some observations about a number of the experiments they have conducted. In steel/UO_2 mixtures the steel did not float to the top - probably because of crust formation. MgO and UO_2 form a eutectic at about 2200°C. The UO_2 diffused about 1/4 inch into the MgO matrix.

Experiments are also being conducted on melt-concrete interactions and related heat transfer and spalling characteristics. These have led to temperature dependent heat transfer correlations for concrete. In graphite - UO_2 interaction experiments, the amount of steel present and type of graphite affects the rate of formation of UC and CO.

Analytical work continues on the GROMS-II melt front penetration code. Improvements include modeling the corners of the melt front and applying more appropriate heat transfer correlations. The GROMS-II version of the GROMS code should be ready for release by the end of the year. Simulant material experiments are being

conducted to determine the effect of gas evolution on downward heat transfer, and the shape of the melt front for various density materials. Results show that the bottom heat transfer rate depends upon whether the pool is increasing or decreasing in density as the sacrificial material combines with the molten fuel pool.

3. ATOMICS INTERNATIONAL, DR. H. A. MOREWITZ

Dr. Morewitz described the experiments being conducted by AI. Their facilities include an arc furnace with a total melt capacity of 100Kg of UO_2. Experiments are now yielding about 10Kg of molten UO_2. All of AI's tests thus far have been transient tests. There has been no sustained heating of the melt. The melt is observed to pour very rapidly, like water. UO_2 poured directly on to a graphite surface did not show signs of attacking it. Dr. Morewitz noted that the cloud of UO_2 vapor formed by the electric arc had a density of .5 to 1kg per cubic meter. Although this interfered with observation and optical measurements during the experiment, it might be useful for aerosol release experiments.

4. BATTELLE MEMORIAL INSTITUTE, P. CYBULSKIS

Mr. Cybulskis reviewed the sequence of events which lead to containment failure and release of fission products for various core melt accidents in LWRs. The viewgraphs from his presentation are attached as enclosure 6. They outline the important processes and failure modes. For the ice-condenser PWR containment, which is the type planned for the FNP, Mr. Cybulskis predicts that H_2 burning or overpressurization due to steam or non-condensable gases will be the most likely mode for containment failure (design pressure for this containment is 15 psig). This will probably occur about the time the core debris melts through the bottom of the vessel. Melt-through penetration of the containment base mat is a secondary failure mode. Water in the reactor cavity will provide the driving force for fission product release and hydrogen generation. Mr. Cybulskis made the following observations about delaying core melt-through. Increasing the base mat thickness will increase the quantities of concrete decomposition products. The use of a basaltic aggregate in the concrete will eliminat the CO_2 production but will not eliminate the water and hence th hydrogen. Using an inert base mat will eliminate the decomposition products and may reduce vaporization fission product release but it does not eliminate the effect of water in the reactor cavity. Mr. Cybulskis pointed out that ECCS systems are designed to pump

water into the vessel no matter what has happened - even though this may not always be desirable. Because of this the analysis of the core melt accident should include the presence of water in the reactor cavity.

5. GENERAL ELECTRIC, DR. EMIL GLUEKLER

Dr. Gluekler presented a review of various materials which might be considered for use as a sacrificial bed. He began with an outline of the properties required for analysis of the penetration of the core debris into the sacrificial material. These included the factors which affect the melt penetration rate, such as phase change, eutectic formation, and the release of vapors and gases, the effects of thermal shock and spallation, and possible chemical reactions between the various phases. Dr. Gluekler then presented a table of physical properties for a wide variety of candidate materials. These tables are attached with the other viewgraphs from his presentation as enclosure 7. He discussed heating rate distributions for fuel-steel-material combinations and the effect of the gas release rate on melt penetration into concrete. He also pointed out the importance of structural properties such as thermal shock resistance since the penetration rate may be a function of structural failure. Materials which contain evaporables such as water can be subject to spalling. The porosity of the material is also important, as are the radiation shielding properties.

Taking all these factors into account, Dr. Gluekler concluded that MgO, Al_2O_3 and perhaps some carbides such as TiC are good candidate materials for use to delay core melt-through.

6. HANFORD ENGINEERING DEVELOPMENT LABORATORY, DR. D. D. STEPNEWSKI

Dr. Stepnewski described the analysis HEDL performed as part of their evaluation of containment margins for the Fast Flux Test Facility (FFTF). This included calculations on the penetration of the concrete base mat below the reactor cavity in the event of a core meltdown. For these calculations it was assumed that fifty percent of the decay heat went into the concrete and that the radial and downward penetration rates were equal. The melting point for basalt concrete was assumed to be 2100° F, the heat of fusion 168 BTU per pound.

7. BROOKHAVEN NATIONAL LABORATORY, DR. W. T. PRATT

Dr. Pratt described some of the work done by BNL in the area of evaluating core melt-down accidents for FFTF, including the effect of using sacrificial materials to delay the melt-through of the reactor cavity. The viewgraphs from this presentation are attached as enclosure 8. He explained the role of BNL in the safety review of FFTF by the NRC staff. The computer codes used at Brookhaven for core melt-down calculations are GROWS and INTER. The GROWS code was written at ANL to study the growth of a molten fuel pool into a sacrificial material. This code has since been modified by UCLA. At UCLA improvements were made in the radial and downward heat transfer correlations and some modelling was added specifically for interactions with concrete. INTER was developed at Sandia to analyze experiments where melts were poured into concrete crucibles. INTER has also been modified by UCLA.

Dr. Pratt outlined some of the differences between the FNP and FFTF. These are summarized in a table in enclosure 8. BNL performed a parametric study on the core melt-through accident in FFTF using GROWS and INTER. The accident was analyzed first for the three feet of concrete below the FFTF reactor vessel, and again assuming 1 1/2 feet of MgO above the concrete. The conclusion from the GROWS calculations, which give no consideration to structural aspects, was that the use of 18 inches of MgO could delay the time at which the FFTF molten core debris would contact the concrete floor by several days. From this analysis, MgO appears a reasonable choice as a sacrificial material.

8. OAK RIDGE NATIONAL LABORATORY, DR. G. W. PARKER

Dr. Parker proposed a plan for maintaining a coolable configuration for the core debris. He would use iron oxide as the sacrificial material in order to dilute the molten core debris and lower the temperature of the mixture. This mixture would eventually end up inside a steel annulus. The outside surface of the annulus would be exposed to the lagoon where natural circulation would keep it cool indefinitely. By choosing the proper iron oxides for the bed, enough oxygen could be made available to oxidize all of the core debris. Some data on the melting temperature and a diagram of the changes which would be made to the hull of the FNP barge are attached as enclosure 9. Some preliminary heat transfer calculations have been done on this design, and they indicate the annulus can be cooled without boiling.

9. SANDIA LABORATORIES, DR. D. A. POWERS

Dr. Powers presented a summary of the experimental and analytical work now underway at Sandia Labs. Studies of core melt/concrete interactions have led to interest in the potential use of other materials in reactor cavity sumps. These alternate materials may cause either crucibilization or dilution. Potential crucibilizing materials are MgO, UO_2, carbon, or high alumina cement. Some diluent materials are borax, lead, or zinc chloride. The present program at Sandia is focusing on MgO and high alumina cement. A brief description of the program and the viewgraphs are attached as enclosure 10. Both small scale transient tests using metallothermically generated melts and large scale sustained interaction tests using inductively heated melts will be conducted using MgO crucibles. Results from past melt experiments where MgO was used as the material to line the furnace where the melt was prepared, indicate several potential problems with the MgO bricks. They undergo severe fracturing when they begin to cool, impurities in the brick may be susceptible to attack by metallic oxides, and the bricks may undergo creep at high temperatures (>1600°). These and other potential problems, such as floating out of the bricks or mechanical failure, point out the need for an experimental program. Large scale experiments must be conducted at temperatures above 1600°C for long periods with prototypic materials and designs. Some large scale experiments are planned for September and October of this year using MgO and high alumina cement. The high alumina cement may provide a compromise between high temperature properties, and low cost and ease of fabrication. The principal ingredient is Al_2O_3 but little is presently known about the high temperature properties of the cement.

Dr. Powers also indicated that while thermal effects were found to dominate the melt erosion of concrete, he expected that chemical effects will dominate the erosion of MgO. He also suggested that placing MgO powder or gravel on top of a layered brick configuration should ameliorate the effects of thermal shock.

SUMMARY AND CONCLUSIONS

Several materials were discussed which might be acceptable for use in delaying core melt-through in a floating nuclear plant. Most of the research in this area has been done with MgO in connection with the LMFBR program. This research has revealed that although some problems exist there are no theoretical barriers to its use as a sacrificial

material in the FNP. Problems, such as methods of construction of the sacrificial bed and the effects of impurities in the brick, must be solved by additional research.

The computer codes presently available for analyzing the penetration of molten core debris into base mat beneath the reactor are GROWS, INTER, and WECSL. INTER and WECSL are not applicable when there is no gas evolution. Both Brookhaven and Argonne National Laboratories are working to improve the modeling in GROWS and development continues at Sandia on INTER.

The staff has evaluated the information presented at this meeting and reached the following conclusions. There are a variety of materials available which might be used to replace the concrete pad below the reactor vessel in the FNP. Some of these would provide a significant increase in the time required for the core debris to melt through to the bottom of the barge. There is, however, no clear choice of one material which is best in all respects, and there are important questions about the physical, chemical and mechanical properties of all of these materials at these extreme temperatures. Additional research is required before the staff could determine the suitability of a particular material. We conclude that there is reasonable assurance that a material can be selected (following additional research) that will perform satisfactorily to substantially increase the melt-through time and reduce the airborne release by reduced gas generation in the melt. We believe that these materials can be incorporated in the FNP design as it exists or with minimum alterations.

Original Signed By
James Carter

James L. Carter, Nuclear Engineer
Advanced Reactors Branch
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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APPENDIX C

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 17 1979

DOCKET NO.: STN 50-437

APPLICANT: Offshore Power Systems

FACILITY: Floating Nuclear Plants 1-8

SUBJECT: SUMMARY OF MEETING HELD ON MAY 7-8, 1979

On May 7-8, 1979, members of the NRC staff and their consultants met with representatives of Offshore Power Systems (OPS) and Harbison-Walker Refractories in Jacksonville, Florida. The primary purpose of the meeting was to discuss materials and design considerations related to the Floating Nuclear Plant (FNP) core ladle as described in OPS Topical Report No. 36A59, dated April 1979. A list of attendees is provided in Enclosure 1.

Generally, the staff and its consultants requested additional discussion and clarification of information provided in OPS Topical Report No. 36A59. Detailed discussions focused on areas related to: steel-industry experience, core ladle design aspects, water moisture effects, gas evolution, slag-line attack, ladle cavity configuration, core melt constituents, ladle instrumentation, heat transfer calculations, gap size, wet core melt scenarios, seismic conditions, concrete type, R&D program, loading conditions, and the plant-site interface criterion. A summary of the discussion on each of these items is presented below.

Steel Industry Experience

The meeting opened with discussions of steel industry experience which is relevant to the FNP core ladle. Refractory materials are used to contain melts at high temperatures in the metal refining industry. In steel making operations, crucibles and hearths of large furnaces are basically large shallow steel containers lined with magnesium oxide (MgO) to provide a heat resistant liner capable of withstanding severe thermal shock and physical impact. For the past century, crucibles and hearths lined with MgO have been used successfully to contain molten iron and steel for long periods of time during the refining processes. In addition, the molten metal at the time of discharge is poured into large capacity ladles, which are also lined with MgO. The ladles, therefore, are subjected to the same extreme service conditions (intense heat, impact loads) and requirements as the crucibles and furnaces themselves. Because MgO has a melting point of about 5100°F and is compatible with the chemistry of molten steel, it is used almost exclusively as the inner linings of the large furnaces and ladles in the steel industry. The temperature of the molten materials in contact with the linings of the furnace crucibles and hearths in steel making operations

is typically in the range of 3000°F to 3500°F, while use of "oxy-fuels" have increased the flame temperature from about 3500°F to 4500°F. Harbison and Walker mentioned that there is experience with nickel melts at 4500°F in contact with MgO. They will provide us with a contact to get more information on experience with nickel melts. Some furnaces, such as the open hearth furnace, are operated almost continuously (22 hours per day) at flame temperatures up to 4500°F. The MgO lining in some of the more modern furnaces, which use hard burned MgO on top instead of a magnesia ramming mix, has been subjected to up to 3000 heats (each heat or the time to melt the steel lasts about one hour) before some repair is needed. The Japanese have exposed similar furnaces for up to 6000 heats. Induction furnaces are operated continuously (i.e., never turned off) with molten steel at about 2700°F in contact with MgO. The physical impact and thermal shock which the steel making furnaces and ladles are repeatedly subjected to represent conditions that are probably more severe than the FNP core ladle would encounter. For basic oxygen steel making vessels, steel and iron scraps weighing as much as 100 tons are dropped into the furnace from heights of 25 to 28 feet. These scraps have been known to contain full steel ingots. Considering the thermal shock aspect, the average pouring temperature of mild steel is about 3000°F; however, the pouring temperature for a high chromium alloy can be as much as 3600°F. Usually, magnesia ramming mixes are used in steel furnaces and ladles as a sacrificial layer over the MgO brick to accommodate the physical impact and thermal shock loads.

Core Ladle Design Aspects

OPS and Harbison-Walker provided additional details on the FNP core ladle design. The core ladle design will take advantage of many years of successful experience that exists with the design and operation of crucibles and hearths used in large furnaces of the steel industry for containing molten materials (iron, steel, nickel) under severe service conditions for long periods of time during refining processes. The ladle will utilize burned high purity magnesite bricks, called HARKLASE, which will be arranged in a cylindrical arch configuration with a staggered bricking pattern in both the transverse and longitudinal directions. As shown in Enclosures 2 and 3, the bricks form rows of inverted arches with double interlocking tongue and groove joints. Each course of bricks will be offset and staggered from joints of the courses immediately above and below it to provide a tortuous flow path for molten core debris penetration. No mortar is used between the bricks. An MgO type of mortar (Oxybond, see description in Enclosure 4) is generally used only for patching up imperfections in the brick courses. A dry magnesite ramming mix (HARMIX FE, See Enclosure 5) may be used as leveling material at the ladle shell - MgO brick interface to fill any voids. In addition, an inert ceramic material, such as that described in Enclosure 6, can be used as filler material to accommodate thermal expansion at the interface between the refractory material and the steel shell. MgO mortar may be used on the sidewalls and floor of the ladle. The mortar on the sidewalls may have some water in it in order to bond the bricks together. This would require using hydration treated bricks on the sidewalls. Some additional research would be required to develop an MgO mortar without water that would still bond bricks together on the sidewalls. Metal encased bricks on the sidewalls

could also be used to bond bricks together without the use of mortar. The bricks in the interior regions of the ladle are laid in place as close together as possible (~1/32 in.). When the bricks heat up, they will expand, soften and fuse together without crushing forming a monolythic structure. Most of the thermal expansion is taken up at the outer extremities of the core ladle and not between bricks. The thermal expansion of the cylindrical arch configuration results in forces and movement in the downward and lateral directions, but not upward; therefore, this configuration along with the double tongue and grooves should be very effective in preventing brick float-up caused by higher density core melt debris. In addition, the cylindrical arch configuration results in the bricks always being in compression on heatup, so that cracking or shearing will not result in bricks coming loose. The creep of MgO is significant at higher temperatures, and this tends to fix thermal expansion considerations to a temperature of about 2600°F. Therefore, the bricks tend to relieve themselves at higher temperatures without crushing. The top-most layer of the ladle will probably be a row of chemically-bonded magnesite bricks called TOPEX S (see Table C-1, OPS Topical Report No. 36A59, April 1979, for property list) which has high resistance to thermal shock and physical impact. Harbison-Walker emphasized an additional safety factor that results from (1) the formation of a lower melting point eutectic between the core melt debris and MgO, and (2) dilution of core melt with MgO will lower the melt temperature. Both of these effects will result in making the underlying layers of MgO more refractory compared to the melt material, since as time passes the MgO melt temperature becomes greater compared to the core melt/MgO system temperature. The staff requested additional design information, as it is developed. for the ladle perimeter (sidewalls and floor) region, including the interface between the MgO bricks and the steel shell.

Water Moisture Effects

The staff questioned the effects of water moisture on the performance of MgO refractories, including what can be done to minimize degradation. The applicant emphasized that the core ladle is entirely enclosed in a steel shell to prevent water from coming into contact with the MgO bricks during normal operation. Harbison and Walker pointed out that precautions should be taken to prevent water from coming in contact with MgO. Magnesia will hydrate quite rapidly under steam conditions. Under ambient conditions and moderate humidity, magnesia brick normally hydrates quite slowly. Hydration of MgO will result in expansion and swelling. Bricks have been stored inside for several years with no significant effects. MgO bricks can be treated to improve hydration resistance. Harbison-Walker will evaluate further the long term effects on MgO of moisture absorption caused by high atmospheric humidity. Accelerated hydration testing may be used to examine this. The staff asked that Harbison and Walker also evaluate what atmospheric conditions are unacceptable for the core ladle performance, so that the necessary environmental control can be established.

Gas Evolution

Considering the potential for gas evolution during heating of TOPEX S bricks, which are chemically bonded, Harbison-Walker estimated a maximum of 2 to 3% by weight of sulphur and carbon which will form SO_2 and CO_2 in an oxydizing atmosphere. For an oxygen poor atmosphere as may exist in the reactor cavity during a core melt event, the quantity of non-condensible gases produced would be much less, perhaps of the order of 1%. For the core ladle application, Harbison and Walker agreed to calculate the maximum amount of gas evolution that can be expected from TOPEX S bricks. For the HARKLASE burned bricks, there should be essentially no gas evolution. Harbison and Walker also stated that they would perform a literature survey and prepare a bibliography of technical references on MgO refractories.

Slag-Line Attack

With respect to the consequences of a three component (UO_2, ZrO_2, Fe_2O_3) oxide mixture attack on MgO and possible lower melting point eutectic formations, there is no direct information and this will be the subject of future research. Considering the potential for slag-line attack which is the preferential attack of MgO by an iron oxide slag layer on top of molten iron, this has been observed in steel making operations. In some furnaces like the blast furnace, the slag layer is about one-third of the total height (or approximately several feet) of the molten material height and increased erosion by about 25% at the slag-line has been observed. Discussions with U. S. Steel have indicated that for a 24 to 27 inch MgO lining in a basic oxygen furnace, it would last about 3000 heats (at 1 hour each) or 3000 hours of operation before slag line erosion would have to be repaired. Harbison and Walker suggested that a special brick composition, such as one composed of chromium, alumina and magnesia, could be used in the ladle design at the slag line to mitigate this particular erosion mechanism. Another possibility would be to increase the lateral thickness of the ladle. The staff requested that Harbison and Walker provide more quantitative information on the extent of slag line attack of MgO that has been observed in steel making operations.

Ladle Cavity Configuration

The staff questioned whether OPS has examined a deeper core ladle cavity that would result in increased MgO sidewall thickness and decreased MgO floor thickness as compared to the present configuration. It is the staff's opinion that increasing the sidewall thickness of MgO should increase the melt-through delay time because it provides greater margin for: (1) lateral melt front erosion which is likely to be equal to the verticle erosion, (2) slag-line attack, (3) barge list or tilt, (4) increased quantities of core melt debris, and (5) heat being generated primarily in the oxide phase and not the metal phase which is likely to be on bottom. OPS responded that

they have some flexibility to change the configuration of the core ladle, and as the design information is developed, they will examine other core ladle configurations to address the above staff concerns.

Core Melt Constituents

Considering the amount of iron that can be included in with the core melt debris, the applicant assumed 25% of the lower core support plate steel and 25% of the reactor vessel steel would melt. They do not expect much of the iron to be oxidized inside the reactor vessel because water is expected to be boiled off before vessel melt-through. However, pumping water onto the core melt debris could oxidize the iron in the core ladle, especially if there is no crusting on top of melt. Any iron oxide formed is expected to be mixed in with UO_2 and ZrO_2. The staff asked for an update and expansion of information, perhaps in a tabular form, of the quantities of the core melt constituents, including a range of possible quantities of iron oxide that can be formed.

Ladle Instrumentation

The staff questioned whether the moisture level in the core ladle would be monitored since hydration could have an adverse effect on the performance of MgO. OPS mentioned that a test was performed at Sandia where molten material was poured on a water saturated MgO brick. The test results indicated that the absorbed water had little effect on the performance of MgO. The staff stated that they were not familiar with these test results and would check further with Sandia. The applicant agreed to evaluate the need for moisture monitors; sufficient justification will be provided if they are found not to be necessary. The need for monitoring moisture will also be factored into Harbison and Walker's evaluation of moisture effects on MgO during humid days and whether special treatment of MgO to improve hydration resistance would be necessary.

With respect to measuring temperatures in the core ladle region, OPS indicated that they plan to install thermocouples on the very outside top and bottom surfaces where they would be accessible for replacement. The staff emphasized that it would be highly desirable to have thermocouples installed inside the core ladle MgO to monitor the approximate extent of the core melt front penetration. The staff indicated that it would be necessary to know about when melt-through was going to occur in order to take appropriate actions. OPS indicated concern as to whether the thermocouples would last for 40 years and whether they could be replaced. The staff stated that it is not clear that the thermocouples either would not last for 40 years, or that it would be very difficult to replace them. The applicant agreed to evaluate the need for installing thermocouples inside the core ladle; if this is found not to be necessary, sufficient justification will be provided. There may be a problem related to reading the thermocouples in the control room because of high radiation levels associated with the core melt debris being in the ladle. This will be investigated further.

Heat Transfer Calculations

The applicant provided further clarification of the transient heat transfer calculations that are illustrated in Figures IV-4,5 and 6 of OPS Topical Report No. 36A59. A transient heat conduction code (TAP-A) was utilized to perform those calculations; a report describing this code was provided to the staff for review. The calculations for the upward and downward heat transfers were made separately and independently from one another. Future analyses will couple the upward and downward heat transfers. Calculations to date are showing that the upper reactor cavity will rapidly heat up and not much heat ($\leq$ 30%) will be lost in the upward direction. For this amount of upward heat transfer, one layer of MgO bricks on the inside of the upper reactor cavity surface is expected to prevent the concrete from melting and, therefore, plugging of the gas and vapor vent passage should not be a problem. In addition, the effect of an air gap between the MgO and concrete, which was not considered in the calculations, should further reduce the concrete temperatures. The staff expressed concern that some of the upper reactor cavity concrete surfaces were not protected with an MgO liner, and this could result in melting of the concrete. As part of the detail design, OPS will evaluate the need to protect the upper reactor cavity concrete with MgO. In addition, as part of more detailed heat transfer analyses, the applicant will examine other core ladle aspect ratios that result in increased sidewall thickness of MgO to determine if melt-through delay time can be extended.

Gap Size

The staff questioned the bases for the three inch gap size between the core ladle steel shell and the lower reactor cavity concrete. This gap will be used to accommodate thermal expansion of the core ladle and also provide a passage for the venting of gas and vapor from heated concrete. With respect to thermal expansion, OPS assumed a 24 foot long, solid slab of MgO which is heated uniformly to 3000°F. Their calculations indicate that the MgO would expand 6 inches total or 3 inches on each side. Information supplied from U. S. Steel Corp. indicates that in practice, the MgO will expand only about one-half this amount because of the joints between bricks and non-uniform heating conditions. OPS emphasized that this is their first cut at determining the gap size, and there are no constraints from increasing the gap size to 4 or 5 inches if necessary. The staff also questioned how the outside surfaces of the lower reactor cavity concrete would vent. OPS felt that cracks would develop in the concrete which will provide a vent passage through the concrete and then up the normal vent passageway surrounding the core ladle steel shell.

Wet Core Melt Scenarios

The staff questioned whether wet core melt scenarios have been considered, such as pumping water on top of the melt surface. OPS responded that they have only considered dry core melt scenarios in order to be conservative. They stated that pumping ECCS water on top of the melt would certainly improve the upward heat dissipation, but that this should probably be avoided because of the increased activity which steam generation would sparge from the melt into the upper containment region where it can be released via cracks in the containment building. A decision regarding the desirability of pumping ECCS water on the core melt debris will be made as part of the final design evaluation when more research information will be available to better judge the advantages and disadvantages of this action. Any ECCS water which is introduced into the core ladle region following a meltdown will come in contact with the core melt debris and not with the MgO.

Seismic Conditions

With respect to the seismic conditions imposed on the core ladle, the applicant stated that the ladle will be designed for all of the operating basis environmental loadings, including remaining functional during an Operating Basis Earthquake (OBE). OPS stated that the core ladle will not be Seismic Category I. The staff stated that it would be desirable to show through a reasonable analysis that the core ladle can adequately withstand the Safe Shutdown Earthquake (SSE) loadings. OPS felt that their present position is adequate and requested the staff to provide in the future additional guidance on seismic requirements for the core ladle.

Concrete Type

For those areas of the reactor cavity that are exposed to high temperatures, the applicant stated that they will use either basaltic type concrete or some other equivalent type in order to minimize gas generation. Limestone type concrete will not be used in the reactor cavity. OPS is also investigating the possible use of a granitic type concrete.

R&D Program

The staff questioned whether a scale-model demonstration test of an integral refractory ladle design would be performed as part of the applicant's R&D program. OPS responded that this is not planned, and that in their opinion the combination of their proposed separate effects tests and the experience that exists with steel making operations are sufficient for their information needs. In addition, OPS recommended that the staff observe first-hand some related steel-making operations in order to gain more confidence. OPS agreed to provide contacts for arranging such a visit for the staff and its consultants.

Loading Conditions

One of the design requirements for the reactor cavity structure and core ladle is that "... the platform structure shall withstand expected loading conditions for the duration of core-melt debris retention." The staff stated that it is not clear from the OPS Topical Report (No. 36A59) what the expected loading conditions are when the core melt material is in the ladle. The applicant agreed to provide more discussion and explanation of the expected loading conditions, including an identification of the loading parameters, for the lower reactor cavity structures and the core ladle.

Plant-Site Interface Criterion

As part of the plant-site interface, OPS has re-written the NRC staff's siting requirement (1.B, p. XV, FES-III) for locating FNP's at estuarine or riverine sites. OPS claimed that the staff's criterion is not definitive enough, and that their proposed criterion is more quantitative and allows for a better yardstick to determine compliance. OPS requested the staff to further review their plant-site interface criterion and provide additional guidance in developing a more definitive site criterion for estuarine or riverine sites.

Summary of Commitments

At the conclusion of the meeting, all of the commitments were identified and are summarized below.

The applicant including its consultant, Harbison-Walker Refractories, committed to providing the staff with information related to the following items prior to the staff's issuance of its SER supplement:

1) The long term effects on MgO of moisture absorption caused by high atmospheric humidity.

2) Estimate the maximum amount of gas evolution (SO_2, CO_2) from TOPEX-S bricks.

3) Bibliography of technical references on MgO refractories.

4) Contact in steel-making industry who is familiar with operations involving nickel melts in contact with MgO.

5) Quantitative information on the extent of slag-line attack of MgO in steel making operations.

6) Tabular list of quantities of core melt constituents, including a range of possible quantities of iron oxide that can be formed.

7) Identify loading conditions and parameters for the lower reactor cavity structures and the core ladle.

MAY 17 [illegible]

Following the staff's issuance of its SER supplement and as OPS further develops the FNP core ladle design, the applicant committed to providing information related to the following items:

1) Evaluation of the need for instrumentation (thermocouples, moisture monitors) in the core ladle region; sufficient justification will be provided if the instrumentation is found not to be necessary.

2) As part of more detailed heat transfer calculations, evaluate the need to: (a) protect the upper reactor cavity concrete with MgO, and (b) increase the ladle sidewall thickness of MgO such that the melt-through delay time is extended.

3) Provide design details of the MgO ladle perimeter (sidewalls and floor) region, including the interface between the MgO and the steel shell.

4) Provide contacts in the steel-making industry regarding visiting and observing related steel-making operations.

The applicant requested that the NRC staff provide additional guidance in the areas of: (1) seismic requirements for the FNP core ladle, and (2) a more definitive site criterion for estuarine or riverine sites.

A. R. Marchese

A. R. Marchese
Advanced Reactors Branch
Division of Project Management

Enclosures:
As stated

cc w/encls:
See next page

ENCLOSURE 1

ATTENDANCE LIST

MEETING WITH OFFSHORE POWER SYSTEMS

MAY 7-8, 1979

NRC - STAFF

A. Marchese
G. Chipman

Offshore Power Systems

D. Walker
H. Stumpf
A. Caudrin
B. Haga
C. Dotson
R. Touchton
R. Thomas
N. Seaborne

Aerospace Corp.

D. Swanson

Brookhaven National Laboratory

W. Pratt

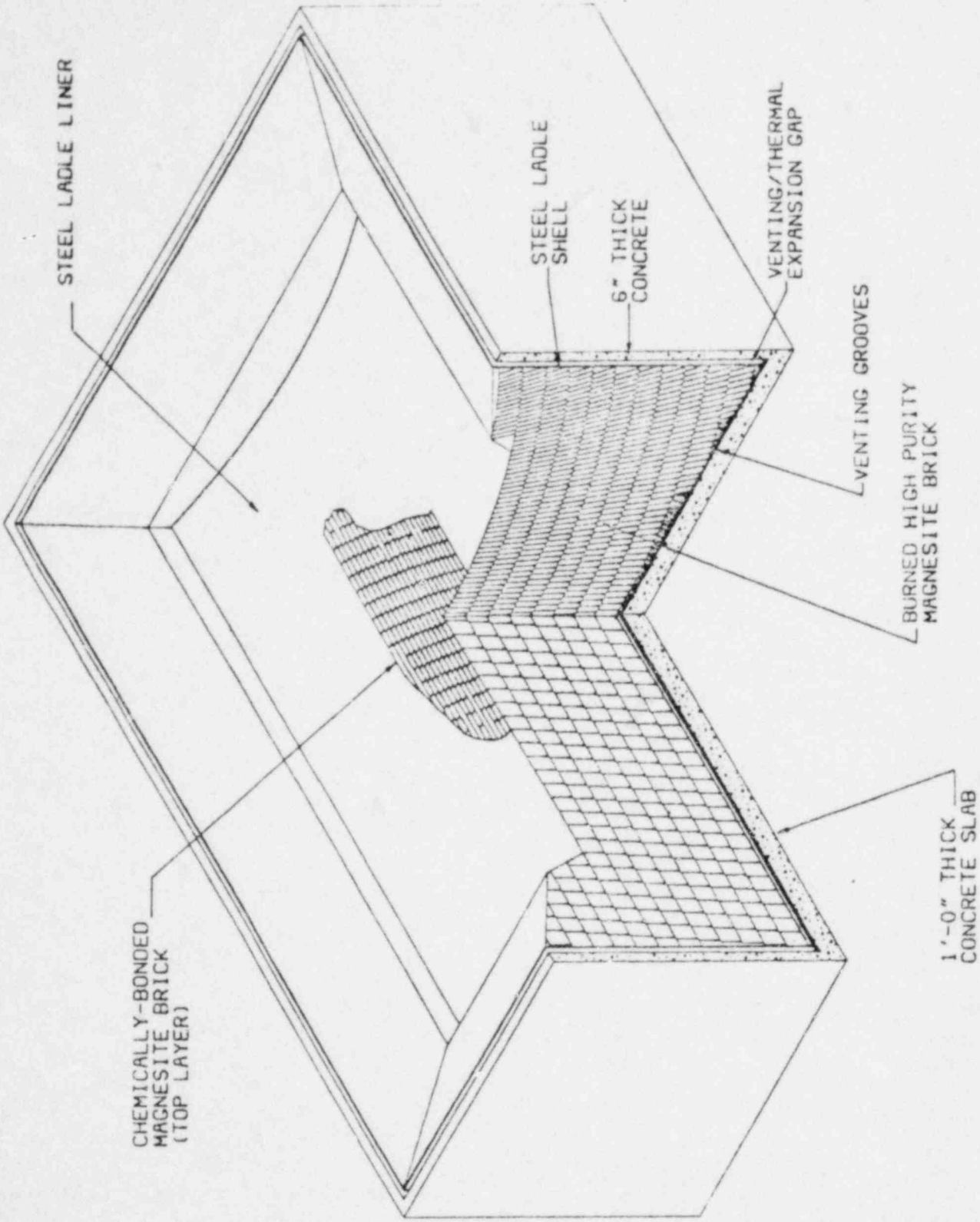
Harbison and Walker Refractories

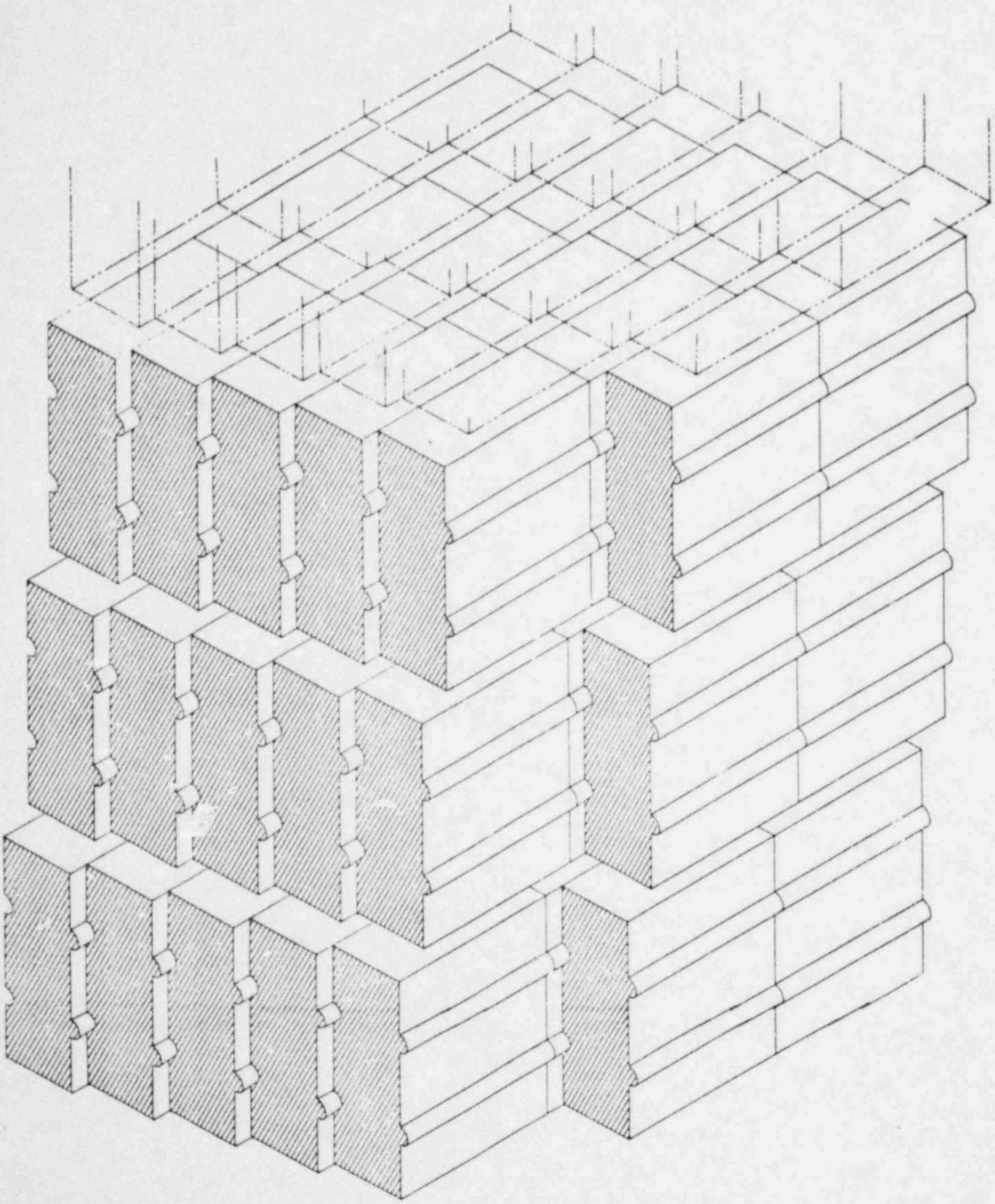
L. Allen
C. Morton
P. Schlett

ACRS STAFF

G. Quittschreiber

ENCLOSURE 2


STEEL LADLE LINER
STEEL LADLE SHELL
6" THICK CONCRETE
VENTING/THERMAL EXPANSION GAP
VENTING GROOVES
BURNED HIGH PURITY MAGNESITE BRICK
1'-0" THICK CONCRETE SLAB
CHEMICALLY-BONDED MAGNESITE BRICK (TOP LAYER)



HARRISON-WALKER REFRACTORIES
Dresser Industries, Inc.
2 Gateway Center, Pittsburgh, Pennsylvania 15222

OXIBOND

Description: Extremely refractory, dry, heat-setting magnesite mortar.

Uses: Used dry for leveling and grouting Basic Oxygen Furnace working linings.

Mixed with water, OXIBOND is ideally suited for laying H-W MAGNESITE, H-W PERIKLASE, and other types of basic brick in applications where a heat-setting basic mortar is required.

Technical Data: Physical Properties (Typical)

Approximate pounds required per 1000 9" Equivalent. Dry for BOF linings -	80 to 85
If used wet, approximate amount of water required for trowelling consistency (per 100# mortar)	3-3/4 to 4-1/4 U.S.Gal.
Approximate Pounds Wet Mortar required per 1000 9" Equivalent:	
Brick Laid Dry then Grouted	300 to 400
Brick Laid using Thinly Trowelled Joints	500 to 600

Refractoriness Test: Mortar does not melt or flow out of joints when heated for 5 hours at 2910°F.

Chemical Analysis: (Typical) (Calcined Basis)

Silica	(SiO_2)	4.2%
Alumina	(Al_2O_3)	0.3
Iron Oxide	(Fe_2O_3)	0.3
Lime	(CaO)	0.9
Magnesia	(MgO)	94.3

NOTE: All data subject to reasonable deviation.

Shipping Data: Shipped dry in 100# moisture-proof sacks.

ASTM Test Methods, where applicable, used for determination of data.

FG-5

HARMIX FE

Technical Data:

Physical Properties: (Typical)

	English Units	SI Units
Maximum Service Temperature	4,000°F	2,204°C
	lb/ft³	kg/m³
Weight Required For Ramming	164	2,620
Bulk Density After Drying at 230°F (110°C)	164	2,620
	lb/in²	kPa
Modulus of Rupture After Heating at 3,000°F (1649°C)	800 to 1,000	5,500 to 6,900
Cold Crushing Strength After Heating at 3,000°F (1649°C)	4,000 to 6,000	27,600 to 41,400
Permanent Linear Change, % After Heating at 3,000°F (1649°C)	-0.4 to -0.6	

Chemical Analysis:
(Approximate)
(Calcined Basis)

Silica	(SiO_2)	0.8%
Alumina	(Al_2O_3)	0.2
Iron Oxide	(Fe_2O_3)	0.2
Lime	(CaO)	0.5
Magnesia	(MgO)	98.3

All data subject to reasonable deviations and therefore should not be used for specification purposes.

ASTM Test Methods, where applicable, used for determination of data.

HARMIX FE (Cont'd)

Description:	A magnesite ramming mixture of exceptional purity and stability.
Features:	Closely controlled grain sizing, optimum for good ramming, and dense, stable structure. Highly resistant to chemical attack by ferrous metals and oxides.
Uses:	Designed primarily as a lining material for coreless-type induction furnaces melting ferrous alloys.
Shipping Data:	Shipped in multi-wall moisture-proof sacks of 100 pounds (45.36 kg.) net weight.

APPENDIX D

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 25, 1979

Harold R. Denton
Director, Office of Nuclear Regulatory Regulations

SUBJECT: ACRS REVIEW OF THE FLOATING NUCLEAR PLANT CORE LADLE DESIGN

At the June 27, 1979 ACRS Subcommittee Meeting on the Floating Nuclear Plant, members of your staff requested that the ACRS meet at an early date to discuss the proposed FNP Core Ladle Design and to write a letter to Mr. Gossick commenting on that preliminary design prior to the NRC Staff's issuance of its safety evaluation. The Acting ACRS Subcommittee Chairman informed your staff and representatives of Offshore Power Systems that the suggestion to hold an early ACRS meeting would be considered at the July 1979 ACRS meeting.

The proposal to hold an early ACRS review of the conceptual design of the FNP core ladle was discussed at the July 1979 ACRS meeting. It was decided that additional information, as indicated below, is necessary before the Committee can proceed with its review of the FNP.

a. Items Related to the Impact that the Core Ladle Will Have on Other Containment Structures

1. Calculate the fraction of decay heat radiated from the pool for the proposed design.

2. Calculate the effects of heat radiation in Item 1 on the rate of:

   (a) disintegration and collapse of exposed concrete

   (b) disintegration and collapse or melting of concrete behind the 6 inch magnesite brick wall

   (c) collapse of steel from the reactor cavity.

3. Discuss the consequences of Item 2 with respect to:

   (a) loss of integrity of superstructures

   (b) loss of hearth capacity

(c) impact resistance of the hearth and its supports

(d) integrity of structural steel members.

4. Discuss the stability of the 6 inch magnesite brick wall above the hearth level with respect to:

   (a) loss of brick by spalling

   (b) differential motion with respect to the hearth, concrete walls, and anchors

   (c) loss of concrete behind the wall by spalling, disintegration, and melting at calculated temperatures, or at temperatures indicated in Fig IV-6 of OPS Topical Report No. 36A59

   (d) slagging reaction between the brick walls and melted concrete.

5. Discuss the fluxing of magnesite brick by siliceous material falling into the hearth.

6. Discuss the properties and merits of basalt as a concrete aggregate.

7. Discuss the possibility of the heat flux being higher on the sides of the molten mass than on the bottom (FRG conclusion for concrete melt) with melting going horizontally faster than vertically.

b. [illegible] Related to Three Mile Island Accident

1. Discuss the possibility of the Upper Head Injection System releasing nitrogen into the primary system and impeding the ability to establish or maintain natural circulation.

2. Discuss the acceptability of the single failure criterion.

3. Discuss the timed sequence of events upon the loss of all AC power before core damage will result.

4. Discuss the reliability of the auxiliary feedwater system.

5. Discuss how H_2 buildup in the ice condenser containment is dealt with following a TMI event and following a core melt.

6. Discuss how the FNP compensates for the difficulty, due to the remote location and the lack of space available, in improvising new systems and techniques in case of an accident.

7. Discuss how one faces lack of flexibility for design changes due to the compactness and lack of available space on the FNP.

c. Items Concerning the Effects of Changing Base Mat Materials

1. Discuss the effects of changing the base mat from concrete to magnesium oxide on the probability of a major air release during a core melt accident. Discuss the comparisons of probabilities and dose levels for air releases associated with concrete and magnesium oxide during a core melt accident.

2. Discuss the consideration given to the use of a vented containment. Discuss the consideration given to the use of sea water for venting and/or cooling a molten core.

3. Discuss the change in position for allowing the FNP to be placed on riverine and estuarine sites. Has the proposed installation of the core ladle changed the NRC Staff's position on this matter, if so, why? What actions and in what time period, are considered practical to isolate the core for a riverine or estuarine site?

4. Discuss the NRC Staff's position that the FNP Core Ladle is considered an environmental issue and not a safety issue.

d. Additional Information Requested From the NRC Staff

1. Provide available information on the Sandia 100 plant liquid pathway study.

2. Provide available information on the WASH-1400 type study of the ice condenser type plant, along with a comparison for non-ice condenser type plants.

Following receipt of Offshore Power System's response to the items listed above and a written evaluation by the NRC Staff, another ACRS Subcommittee meeting will be held. Please advise us of the date by which you believe the above information will be available so we can schedule related ACRS activities.

R. F. Fraley
Executive Director

cc: D. Muller, DSE
E. Case, NRC
D. Vassallo, DPM
F. Schroeder, DSS

APPENDIX E

Offshore Power Systems

8000 Arlington Expressway
Box 8000, Jacksonville, Florida 32211

904-724-7700
Telex 568406

September 14, 1979

Mr. Robert L. Baer, Chief
Light Water Reactors Branch No. 2
Division of Project Management
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20852

B. Haga

Contracts & Licensing

Re: Docket STN 50-437; ACRS Questions on Core Ladle and TMI-2

Dear Mr. Baer:

Transmitted herewith are 20 copies of the Offshore Power Systems responses to the ACRS Subcommittee questions contained in R. F. Fraley's letter to H. R. Denton dated July 25, 1979. Please note that we have not offered responses to part d. of Mr. Fraley's letter as these requests were made specifically to the NRC Staff. By copy of this letter, 20 copies of our responses are being transmitted directly to Mr. Fraley for distribution within ACRS.

Certain material in the attached responses reflects modification to the design presented in OPS Report 36A59, "FNP Core Ladle Design and Safety Evaluation". The principal changes are increased ladle volume and increased refractory insulation on the walls of the reactor cavity. Both of these changes resulted from our ongoing evaluation of radiant upheating from the pool surface. The analyses of radiant upheating, which are described in the attached responses, are believed to be adequately conservative to show feasibility and therefore to support the issuance of the Manufacturing License. Following NRC Staff review those responses which affect the present content of Report 36A59 will be retransmitted in the form of a revision to that report.

We ask that these responses be reviewed on an expedited basis leading to an ACRS Subcommittee meeting as early as October

1979. To this end, we are prepared to offer any assistance the Staff may require.

Very truly yours,

P. B. Haga

P. B. Haga

/lel

Attachments

CC: R. F. Fraley (ACRS)
V. W. Campbell
A. R. Collier

1979. To this end, we are prepared to offer any assistance the Staff may require.

Very truly yours,

P. B. Haga

P. B Haga

/lel

Attachments

CC: R. F. Fraley (ACRS)
V. W. Campbell
A. R. Collier

APPENDIX F

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C

November 2, 1979

Docket No. STN 50-437

MEMORANDUM FOR: Raymond F. Fraley, Executive Director, Advisory Committee On Reactor Safeguards

FROM: D. B. Vassallo, Acting Director, Division of Project Management, NRR

SUBJECT: ACRS LETTER OF JULY 25, 1979 - FNP CORE LADLE DESIGN

This is in response to your July 25, 1979 memorandum to me indicating that the Committee desired additional information concerning the Floating Nuclear Plant core ladle design and related matters. Offshore Power Systems has addressed the contents of your July 25, 1979 memorandum in their September 14, 1979 letter to us. We have reviewed this information and the Enclosure to this letter provides the Committee with our comments.

I understand that an ACRS Subcommittee meeting on the FNP core ladle design has been scheduled for November 17, 1979. The NRC staff wishes to proceed to conclude our review of this matter as promptly as possible. Therefore we would like to have the concept and preliminary design of the core ladle reviewed by the full Committee at the December 1979 meeting, so that we could consider Committee input and comments as part of our review effort.

D. B. Vassallo, Acting Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosure:
Staff Review and Evaluation
Of Offshore Power Systems
Response To ACRS Letter Of
July 25, 1979

STAFF REVIEW AND EVALUATION

OF

OFFSHORE POWER SYSTEMS

RESPONSE TO ACRS LETTER OF JULY 25, 1979

NOVEMBER 1979

TABLE OF CONTENTS

TABLES

FIGURES

APPENDICES

INTRODUCTION

On June 27, 1979 the ACRS Subcommittee responsible for the rev. · of the Offshore Power Systems application to manufacture eight floating nuclear plants, met with the applicant and the staff to review the FNP core ladle design as submitted by the applicant in Topical Report No. 36A59, "FNP Core Ladle Design and Safety Evaluation." This conceptual design feature was proposed by the applicant in response to the staff's environmental assessment (FES, Part III) and the LPGS Report (NUREG-0440). Subsequent to the ACRS Subcommittee meeting, the ACRS at its July 1979 meeting issued a letter (see Appendix A) requesting additional information from the applicant and an evaluation of the response by the staff. By letter dated September 14, 1979 (see Appendix B) the applicant provided additional information which the staff has reviewed and evaluated. The staff response utilizes the format of the ACRS July 1979 letter.

A. Items Related To The Impact That The Core Ladle Will Have On Other Containment Structures

1. Calculate the fraction of decay heat radiated from the pool for the proposed design.

STAFF RESPONSE

The OPS response is based on calculations which assume the radiation losses from the pool surface to be decoupled from the pool heat transfer processes. It is recognized by OPS that the problem is highly coupled and a computer program is being developed at OPS to solve the coupled problem. The MELSAC Code[1], which is being developed by NRC staff consultants at the Brookhaven National Laboratory (BNL), solves the coupled problem and the results obtained from MELSAC differ from the calculations and assumptions made at OPS.

A major difference between the OPS and the staff results relates to the pool surface temperature histories. OPS assumes the two surface temperature histories shown in Table 1. These temperature histories differ appreciably from the predictions of the MELSAC code. Typical results from MELSAC are shown in Figure 1 for two different pool heat transfer correlations. Case 10* uses the downward heat transfer coefficient given by the Kulacki, Goldstein[3] correlation for an internal heated molten pool. Case 6* models the density driven heat transfer coefficient[4], which arises because the sacrificial material being melted is less dense than the pool material, and a buoyancy-driven motion is induced.

*Case numbers refer to cases reported in Reference 2. Cases A, B and C most closely represent the ladle and cavity configuration described in Reference 5. The assumptions used in Cases A, B and C are included in Table 2.

The temperatures shown in Figure 1 are the bulk pool temperature; the upper pool surface temperature is typically 80 to 100 K below this value. The effect of the higher surface temperature history predicted by MELSAC is to transfer more heat to the upper structures. A typical run (Case B) using MELSAC is compared with the results provided by OPS in Figure 2. After one day the fraction of heat stored in the walls and vessel ranges from 0.24 to 0.46 using OPS assumptions, whereas MELSAC predicts the heat stored to be 0.76. Similarly, after 5.79 days (melt-through for Case B) MELSAC predicts a fraction of 0.5 in the walls and vessel compared with 0.11 and 0.17 at 6 days using OPS assumptions. The result of this additional heat transfer to the walls is that MELSAC predicts (Case A) that the concrete in the cavity wall (24" MgO, 3" gap, 21" concrete) will reach its decomposition temperature (1473 K) after only 1 day. In order to determine a wall configuration that protects the concrete (<1473 K) and steel bulkheads (<810 K) beyond the two-day period, a scoping study was carried out and the results are presented in Table 2. MELSAC, predicts (Case B) that a MgO wall of 36" thick would be required to protect the concrete and steel for 2 days, given the input assumptions listed in Table 2.

The times to melt the whole of the reactor vessel by thermal radiation from the molten pool surface varied between 0.5 to 4 days under the assumptions made by OPS. In Table 2, we have included the MELSAC estimate of reactor vessel melting, which is about 1.8 days. The model used in MELSAC is different from that assumed by OPS and is also exposed to the higher pool surface temperature history. In MELSAC the reactor vessel is modeled as a series of connected masses. Heat transfer between each of

the vessel masses is by conduction. However, the MELSAC prediction is not inconsistent with the range of reactor vessel melting suggested by OPS.

The erosion rates presented by OPS are decoupled from the upward heat transfer and were simply obtained by using a constant fraction of the volumetric heat capacity. The erosion rate predicted by MELSAC (for the wall configuration which protects the concrete and steel, namely, Case B) is compared with the OPS erosion rates in Figure 3. The MELSAC erosion rate clearly shows the coupled effect. At early times the heat transfer to the upper structures is high (because of the large temperature differences) allowing only a small fraction (F) of the decay heat to be directed into the MgO. At later times, the upper structures are at higher temperatures and the upward heat transfer is reduced allowing half (F = 0.5) of the decay heat to be directed into the MgO at the point of melt-through (5.79 days). This is consistent with Figure 2 in which the fraction of decay heat stored in the walls is predicted to be 0.5 at 5.79 days.

It should be noted from Table 2 that MELSAC Case B predicts that the 26" of MgO will protect the concrete and steel for 2 days, whereas the core ladle will hold-up the molten pool for 5.79 days. Under the assumptions of the MELSAC code, and using the above wall configuration, considerable damage would be expected to the upper structures in the reactor cavity after the 2 day period and before the molten pool is released from the ladle at 5.79 days.

Finally, our comments must be qualified as they are based on a first version of the MELSAC code. At present MELSAC assumes the molten pool to be initially pure UO_2. As the meltfront moves into the sacrificial bed, the code computes

the dilution of the UO_2 with molten MgO. However, the dilution of the pool by steel and zircalloy cladding (both of which will certainly be present) is not presently modeled. The omission of molten steel addition to the pool from the reactor vessel is significant as the mass of steel in the reactor vessel is 1.5×10^6 lb compared with 0.22×10^6 lb of UO_2.

It is not clear at the present time what the total effect of introducing such a large quantity of molten steel will have on the pool conditions. One effect may be to decrease the pool temperature. Whether the resulting mixture will remain molten and in what configuration (layered or mixed) is not currently known. A simple calculation, which brings the pool into thermal equilibrium with the molten steel just at the time vessel melting is complete, yields an equilibrium temperature of about $2000^{o}K$. This is considerably below the melting temperature of the UO_2-MgO binary system, but both far above the steel melting temperature and far below the steel boiling temperature. Introducing molten steel into the pool at the rate it is being melted may have the effect of lowering the pool temperature, thus decreasing the radiative heat transfer to the vessel and its resulting melting rate. On the other hand, quenching the pool with molten steel may drop the pool temperature below the MgO melting temperature. If penetration of the ladle stops, then dilution of the pool with molten MgO would also stop. The question then arises as to whether or not the quenching effect of the vessel steel would compensate for the loss of the dilution previously provided by the molten MgO. The 2000 K equilibrium temperature is clearly a lower limit. The pool temperature history, allowing for the effect of steel dilution, cannot yet be predicted but would be between the present MELSAC predictions and 2000 K.

Future work on MELSAC will be directed to addressing the effect of dilution the molten pool with steel and zircalloy. We do, however, consider that the current predictions by MELSAC represent early times with respect to ladle penetration, heating of the structures in the cavity and melting the reactor vessel. The effect of diluting the pool with steel and zircalloy would tend to increase the time scale of the above events. However, quantifying the changes in time scale at this stage is extremely difficult. The ladle penetration times predicted by MELSAC are also dependent on the assumed equal lateral and downward heat transfer correlations (see comments on Question a.7). If evidence becomes available to suggest that penetration should be faster in either direction, then the MELSAC code could easily be modified.

The conclusions of the staff, regarding this question and the remainder of the questions in item a of the Committee's July 25th letter, is that the ladle concept is feasible and can be engineered to provide retention of a molten core for a period of time in the range of two days to one week. As noted above, the applicant is in the process of developing a coupled calculation model. Any significant differences between that model and the staff's model can be resolved during the early phases of the final design.

Once a calculational model is agreed upon, the ladle configuration can be optimized for the available space to provide the largest possible core retention time considering all the factors raised in items a.2 through a.7 of the Committee's letter of July 25, 1979.

TABLE 1*

POOL SURFACE TEMPERATURE HISTORIES

TIME (DAYS)	TEMPERATURE	
I. Sandia Estimate		
0	4712°F	(2600°C)
1	3632°F	(2000°C)
2	3524°F	(1940°C)
4	3308°F	(1820°C)
6	3092°F	(1700°C)

TIME (DAYS)	TEMPERATURE	
II. OPS (Black Body) Estimate		
0	3641°F	(2005°C)
1	2394°F	(1312°C)
2	2232°F	(1222°C)
4	2092°F	(1144°C)
6	1952°F	(1067°C)

* REPRODUCED FROM REFERENCE (5)

TABLE 2 SCOPING STUDY USING MELSAC*

CASE	THICKNESS OF MgO IN WALL (m)	THICKNESS OF CONCRETE IN WALL (m)	TIME TO PENETRATE CORE LADLE (DAYS)	TIME TO START MELTING VESSEL (HRS)	TIME TO MELT VESSEL (DAYS)	TIME TO MELT CONCRETE (DAYS)	STEEL TEMP AT START OF CONC. MELT (K)
A	0.6096	0.6096	5.25	2.46	1.78	1.04	360
B	0.9144	0.3048	5.79	2.53	1.83	2.01	870
C	1.0668	0.1524	6.00	2.53	1.83	2.48	1145

* ASSUMPTIONS :-

MASS OF UO_2 101032 Kg

MASS OF VESSEL 670328 Kg

LADLE THICKNESS 1.6 m

MELTING TEMP MgO 3073 K

MELTING TEMP CONCRETE 1473 K

POOL H/T BASED ON DENSITY DRIVEN CORRELATION[6]

POOL SURFACE - STRUCTURES EFFECTIVE EMISSIVITY 0.5

HEAT TRANSFER BEHIND VESSEL, WALL & LADLE BY NATURAL CONVECTION.

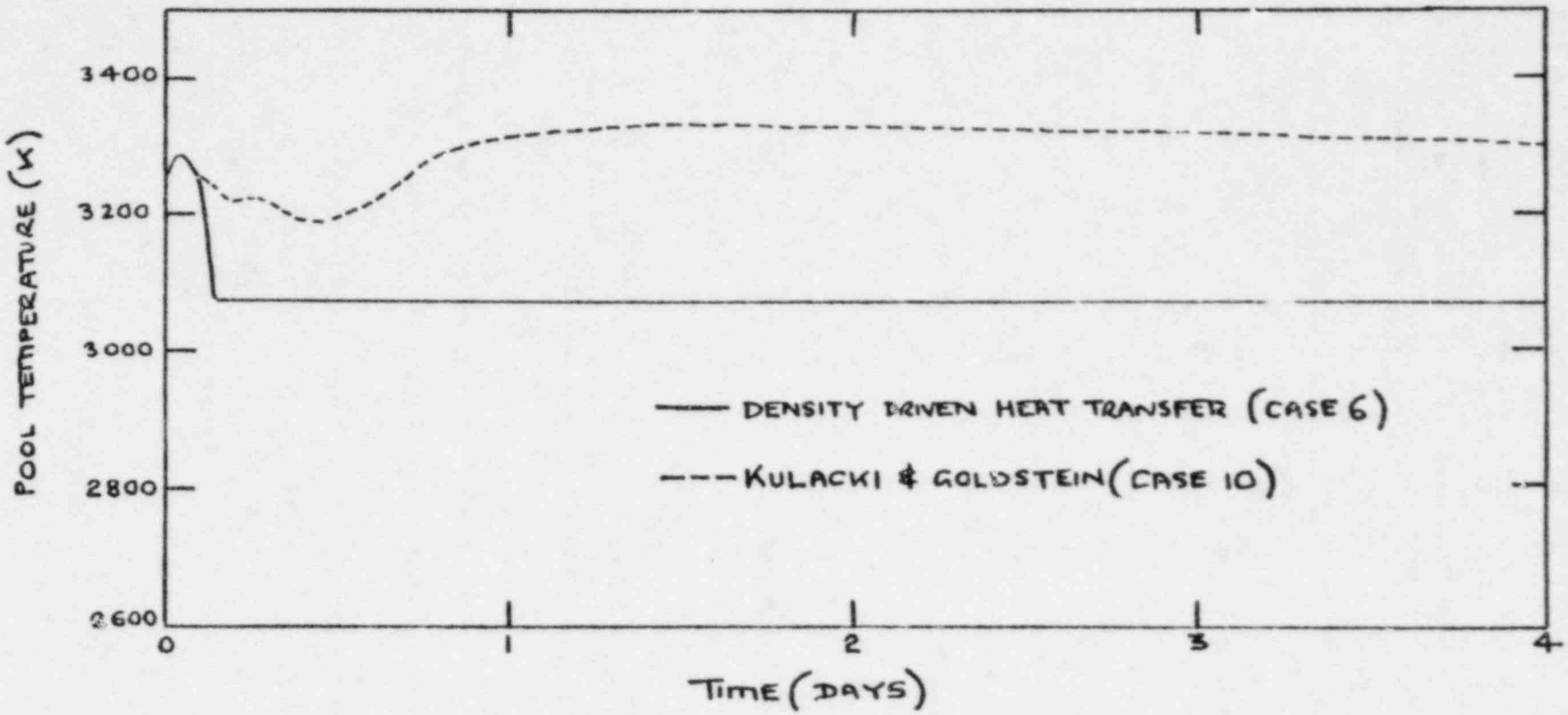
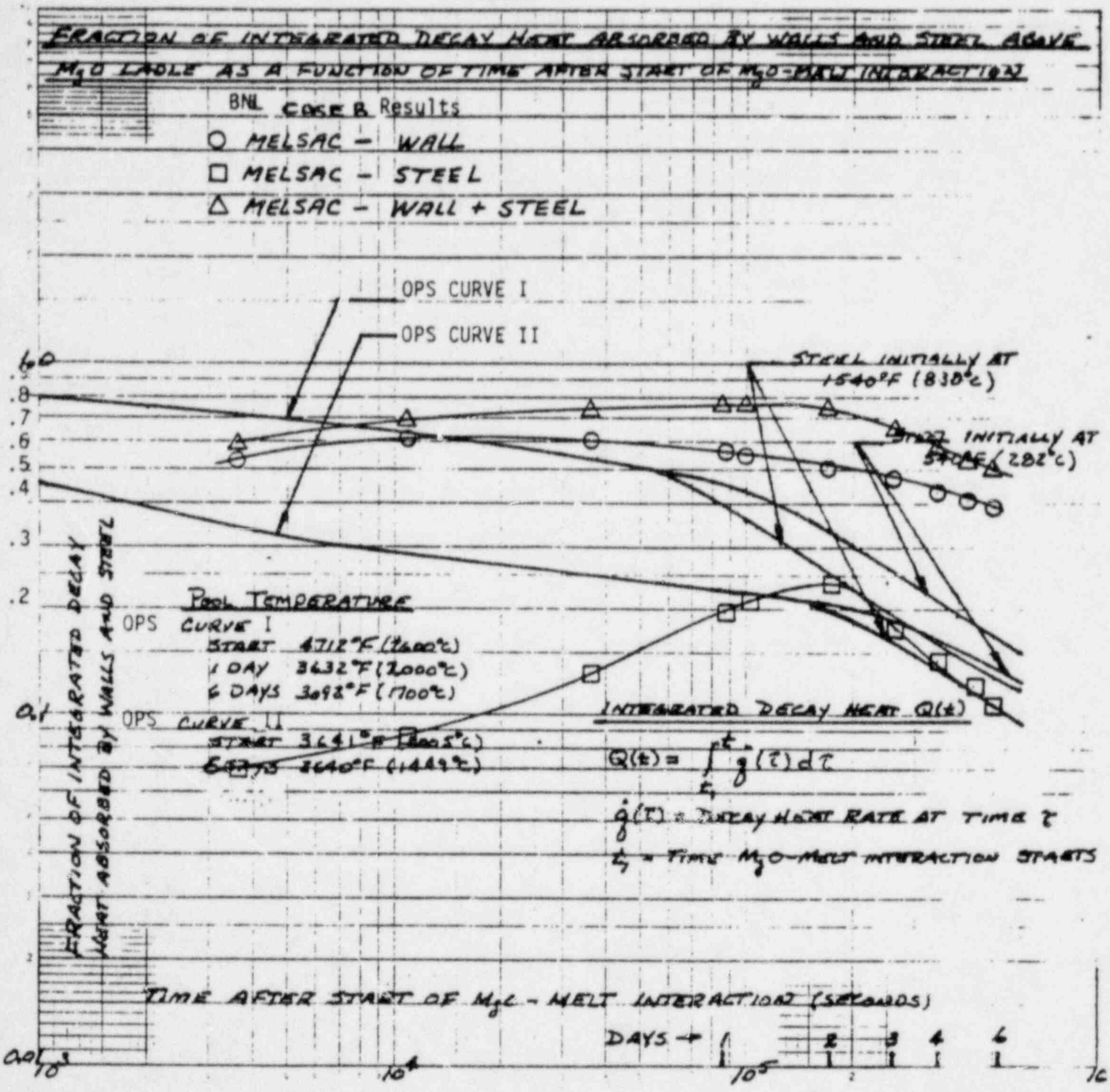


FIGURE 1 POOL TEMPERATURE HISTORIES PREDICTED BY MELSAC

FIGURE 2 *



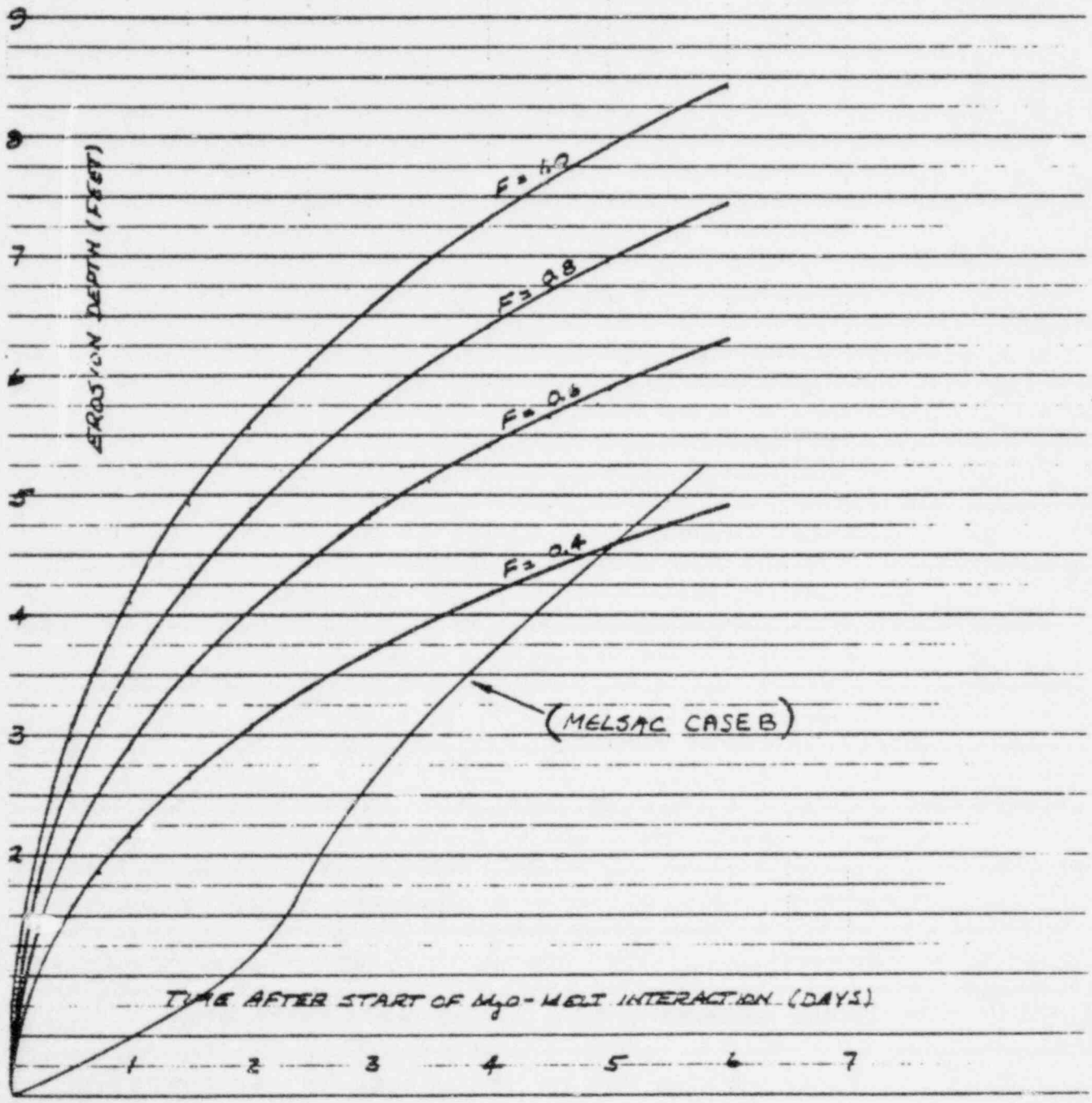
* REPRODUCED FROM REFERENCE (5)

Figure 3*

EROSION DEPTH AS A FUNCTION OF TIME AFTER START OF MgO - MELT INTERACTION FOR VERTICAL AND HORIZONTAL EROSION

NOTE:

1. BASE MAT AREA = 417 FT^2 AT START OF INTERACTION
2. F = FRACTION OF DECAY HEAT DIRECTED INTO MgO
3. VOLUMETRIC HEAT CAPACITY OF MgO IS A FUNCTION OF POOL COMPOSITION. INITIAL VALUE = 3.9×10^5 BTU/FT^3
4. UNIFORM HEAT FLUX INTO MgO



* REPRODUCED FROM REFERENCE (5)

2. Calculate the effects of heat radiation in Item 1 on the rate of:

(a) disintegration and collapse of exposed concrete

STAFF RESPONSE

We agree with the response but with qualifications discussed by us in Question a.1, namely, that we do not accept the pool surface histories and recommended a wall configuration with at least 36" of MgO protection.

(b) disintergration and collapse or melting of concrete behind the six-inch magnesite brick wall

STAFF RESPONSE

We agree with the response regarding the fact that the walls can be protected for two days using suitable high temperature insulating brick. The melting point of basalt aggregate is exactly the value we have used in MELSAC. We do not, however, believe that the configuration of 24" MgO and 21" Basaltic concrete will be sufficient to protect the concrete (refer to Question a.1). We recommend at least 36" of MgO.

(c) collapse of steel from the reactor cavity

We agree with the response and note that the MELSAC prediction for vessel melting of 1.8 days is within the range of 0.5 to 4 days suggested by OPS. The addition of this steel to the pool cannot be modeled in the present version of MELSAC and the effect of dilution of the molten pool by this amount of molten steel was discussed as part of the comments to Question a.1. We again emphasize that the current predictions of MELSAC, therefore, represent early times with respect to penetration, heating up of the structures in the cavity, and melting of the reactor vessel.

3. Discuss the consequences of Item 2 with respect to:

(a) loss of integrity of superstructures

STAFF RESPONSE

The applicant has stated that proper protective barriers can be utilized to protect structures above the reactor vessel which may be affected by upheating following the vessel melt-out.

The protective barriers include high temperature insulating brick (MgO, ZO_2 or Al_2O_3, or possibly ceramic fibres (Fibrefax). These barriers will prevent the concrete from disintegrating, collapsing or melting when properly designed. The applicant proposes a shield that would limit the maximum temperature of the concrete to less than 2200°F, which is below the melting temperature of basaltic aggregate. However, the staff requires that the applicant prove that the melting temperature of basaltic aggregate is the controlling factor and not those of other components of the in-place concrete. With regard to the steel structures, the applicant plans to limit the surface temperature of the primary steel components to 1000°F. Also, for steel components subject to high thermally induced stresses the temperature limit will be reduced. This position meets the requirements of the American Institute of Steel Construction. However, the applicant has not identified any specific structure that will require the above specified protection. The applicant plans to identify the specific protection that will assure the integrity of the superstructures from radiation heat during the final design phase of the FNP.

Based on the evaluation of the above material, we conclude that the applicant has providedan adequate preliminary design. This design provides enough information to give reasonable assurance that the final design will satisfy all of our requirements. However, our final approval is subject to our review of the final design.

(b) loss of hearth capacity

STAFF RESPONSE

The applicant has increased the ladle capacity to over four times the previous capacity and has identified additional space where a properly designed supplementary ladle can be placed to take care of any overflow from the basic core ladle. These actions should dismiss any concerns on the adequacy of the ladle capacity. We consider the core ladle support structures to be acceptable since these structures have been redesigned to comply with the structural acceptance criteria as outlined in the Standard Review Plan. However, the applicant should demonstrate in the final design, the adequacy of the structural systems supporting the ladle in order to determine the time dependent structural capacity to support the actual ladle configuration. Basically, the applicant should demonstrate that the structural members supporting the core ladle will not fail prior to any failure of the core ladle by melt-through of its contents.

(c) impact resistance of the hearth and its supports

STAFF RESPONSE

The applicant has performed analyses with the following postulated new loading conditions: (1) Reactor Vessel Bottom Head Impact, and (2) Upper Reactor Vessel Impact, in order to consider the appropriate impactive loads in the evaluation of the core ladle. The analyses indicated that case (1) noted above, controls the design. The analyses for this case showed that the ladle can resist the impact load without failure. We agree with the applicant's conclusion. However, the applicant should document in the final design the adequacy of the structures supporting the ladle for their capacity to resist these impactive loads. We find the approach used by the applicant acceptable, subject to our review of the final design details.

(d) integrity of structural steel members

STAFF RESPONSE

The applicant plans to shield all of the primary structural steel members within the reactor cavity (bulkheads, floor and deck) from energy radiated from the core ladle.

The protective barriers are designed to reduce the temperature to a level that will avoid rapid deterioration of the steel members or excessive thermally induced stresses. The applicant plans to follow the requirements of the American Institute of Steel Construction. This design code considers steel components fire resistive if the average temperature of the steel members does not exceed 1000°F. The applicant plans to limit this temperature to the surface temperature of the steel members and plans to lower this limit for areas of possible high thermally induced stresses.

4. Discuss the stability of the six-inch magnesite brick wall above the hearth level with respect to:

(a) loss of brick by spalling

STAFF RESPONSE

The response to the question compares magnesia and zirconia brick and can be construed as implying that ZrO_2 is more resistant to spallation than MgO. Although it is true that magnesia brick does have a relatively high rate of expansion among refractories, it should be noted that zirconium dioxide undergoes a phase change from monoclinic to tetragonal at approximately 1150°C. A polycrystaline sample of ZrO_2 brick expands approximately 0.8% from room temperature to 1150°C. In going through the transition the brick shrinks 0.9% in the 100°C interval above 1150°C so that the volume at 1250°C is 0.1% less than its volume at room temperature. This phase change has caused mechanical problems in zirconia bricks that have been used above 1250°C. Consequently, although magnesia has a greater, but montonically varying expansion, structures constructed from magnesia brick should suffer less mechanical damage than structures constructed from ZrO_2.

(b) differential motion with respect to the hearth, concrete walls, and anchors

STAFF RESPONSE

There is considerable information available from past experience with furnance design for steel-making operations that should enable the applicant to design a satisfactory structure from the standpoint of differential motion.

As part of the final design effort of the FNP core ladle, we will request design details and drawings of the proposed anchoring system. We find this criteria acceptable, since their reference is a standard acceptable to the staff.

(c) loss of concrete behind the wall by spalling, disintegration and melting at calculated temperatures, or at temperatures indicated in Figure IV-6 of OPS Topical Report No. 36A59

STAFF RESPONSE

We agree with the OPS response subject to the qualifications discussed in our response to Questions a.1, a.2(a) and a.2(b). As long as the brick wall remains substantially intact the concrete will heat up slowly so that spallation is not expected to be a problem.

(d) slagging reaction between the brick walls and melted concrete

STAFF RESPONSE

We agree with the OPS response subject to the qualifications discussed in our response to Questions a.1, a.2(a) and a.2(b). Slagging reactions should not be a problem as long as the wall remains intact and the concrete remains below its melting point.

5. Discuss the fluxing of magnesite brick by silicious material falling into the hearth

STAFF RESPONSE

We agree with the OPS response, however, please refer to our response to Questions a.1, a.2(a) and a.2(b) with respect to preventing the concrete side walls from melting prior to two (2) days.

We would also like to point out that the response mentions an experiment conducted at Sandia Laboratories in which basalt concrete was melted in an MgO crucible. It should be noted that the MgO brick in the experiment was also heated along with the concrete to 1400°C and this presumably facilitated the formation of the glass-like matrix observed. In the reactor system under discussion, the MgO brick would initially be relatively cold. Therefore, it is questionable whether the experiment is prototypic.

6. Discuss the properties and merits of basalt as a concrete aggregate

STAFF RESPONSE

We agree with the OPS response. Of the relatively common materials that can be used as concrete aggregate, basalt is a good choice. It does not generate gas when it is heated. However, it does contain silica which tends to form low melting mixtures with other materials.

7. Discuss the possibility of the heat flux being higher on the sides of the molten mass than on the bottom (FRG conclusion for concrete melt) with melting going horizontally faster than vertically

STAFF RESPONSE

We agree with the response and it is also our understanding that the high lateral erosion rates observed during penetration of concrete is due to gas generation. The concrete results are obviously not applicable to erosion of MgO.

A number of simulant experiments have been carried out to address the above concern and a number of heat transfer correlations have been proposed. In particular, the experiments carried out under L. Baker at ANL and I. Catton at UCLA have indicated that the correlations used previously[3] are inappropriate to molten core penetrating MgO because the pool is less dense than the molten MgO. The simulant experiments indicate that there is considerable bouyancy-driven motion under these circumstances.

The earlier correlations[3] developed for pool heat transfer were based on a Rayleigh number formulation, which uses the difference between the bulk pool temperature and the melting interface temperature as the driving force. The experiments at UCLA and ANL indicate that the Rayleigh number formulation should be based on the density difference between the pool and the melting material. The density-driven correlations result in much higher heat transfer correlations.

Applying these correlations to prototypic conditions was done through the GROWS code.[6] The original version of the GROWS code used the temperature-driven formulations and the heat transfer correlations tended to favor lateral penetration in preference to downward penetration. A later version of GROWS (GROWS-2) was issued at a recent meeting[7] held at ANL. GROWS-2 uses the density-driven formulations, and while there are large differences in density between the pool and the molten MgO, the code predicts downward penetration rates much faster than the lateral penetrations rates. However, as the pool is diluted with molten MgO, the density difference between the pool and the molten MgO is obviously reduced and the density-driven heat transfer correlations are reduced to the original temperature-driven formulations. The net result is that for core penetration of MgO over a period of several days, the lateral and downward penetration are approximately the same.

For the above reasons we have made the lateral heat transfer coefficient equal to the downward heat transfer coefficient in MELSAC. We only distinguish between the effects of density-driven heat transfer and the original convective formulations using a temperature difference driven Rayleigh number. There is clearly not any experimental evidence yet available that would suggest changing the heat transfer formulations in MELSAC. If such evidence becomes available, it could easily be incorporated into MELSAC.

B. Items Related To Three Mile Island Accident

1.

Discuss the possibility of the Upper Head Injection (UHI) System releasing nitrogen into the primary system and impeding the ability to establish or maintain natural circulation.

STAFF RESPONSE

Upper head injection systems have been reviewed on a generic basis[1], and on specific plants (e.g., Sequoyah). The methods of analysis for UHI systems are generally considered to be conservative modifications of the methods used in analyzing other PWRs, and, subject to the reservations stated in reference 1, are considered acceptable. Questions similar to the ACRS have been raised previously by the staff in particular UHI reviews (e.g., Sequoyah) about the mechanical action of the valve system that shuts off the flow of water from the UHI accumulator tank after injection. These valves are opened when the primary system is brought up to pressure and remain open during reactor operation. Their only action is to close after injection to prevent nitrogen from following the injected water into the upper head. Two lines in parallel from the UHI accumulator provide the reliability needed in the injection path, and two valves in series in each of these lines provide the reliability needed in shutting off the water flow. The system has been reviewed in several PWR plants, and it has been determined that, since the valves have separate and independent power supplies and water-level sensing devices, the valve system meets the single-failure criteria.

(1) S. L. Israel, et al., "Safety Evaluation Report on Westinghouse Electric Company ECCS Evaluation Model for Plants Equipped with Upper Head Injection," NUREG-0297, April, 1978.

It has further been required that the actual installation in each plant be tested prior to plant operation. The tests are to determine the actual amounts of dissolved plus entrained nitrogen carried over to the upper head with the UHI water by a direct sampling technique. These amounts have generally been of the order of <2% compared to an allowable specification of about 4.38% by volume.

The tests are conducted into a system at atmospheric pressure, which gives maximum flow velocities and turbulence, with a maximum possibility for entrainment. The tests are considered conservative in this sense.

The tests indicate that the amount of nitrogen injected into the upper head would be conservatively estimated to be of the order of 20-40 cubic feet at atmospheric pressure. These small amounts would not be significant in the ECCS operation, nor would they interfere with natural convection if that were required.

The applicant's response to the ACRS question has developed essentially these same arguments. The staff has concluded, therefore, that because of the unlikelihood of more than a single valve failure, and in view of the small quantities of nitrogen injected in preoperational tests, the problem of accidental nitrogen injection has been satisfactory resolved.

The effects of the injection of non-condensable gases are the subject of an experimental program that is currently underway. The results of these experiments will be factored into the staff's evaluation as they become available.

2. Discuss the acceptability of the single failure criterion

STAFF RESPONSE

The OPS response acceptably describes the single failure criterion as currently applied by the staff. This criterion however, does exclude some passive failures and some operator errors which have been identified by the TMI-2 Lessons Learned Task Force. Recommendations by the Task Force may modify this criterion.

3. Discuss the timed sequences of events upon the loss of all AC power before core damage will result

STAFF RESPONSE

The OPS response to this question reviews a scenario in which failures are limited to the loss of on-site and off-site AC power sources, without recovery. (DC power supplies were assumed to be available) The turbine driven auxiliary feedwater pump is assumed to function properly, even though operator action could eventually be required to control this system. In the OPS analysis, the principal loss of primary coolant inventory was through pump seals, estimated at 5 gpm per pump. Under the set of assumptions used in the OPS discussion, the consequences are probably satisfactorily discussed. OPS has concluded that the core will remain covered for about 17 hours in this sequence. Although some core boiling will occur, the core will be relatively undamaged during this time. Heat removal through the auxiliary feedwater system is expected to remain viable for about the same length of time (20 hours) based on the use of the turbine driven auxiliary feedwater pump. Instrument systems in the containment building are qualified for a temperature environment that would not be exceeded during this period.

The staff notes that its own studies, using more restrictive assumptions, have predicted core damage in a shorter period. For example, with loss of both AC and DC power supplies including loss of the turbine driven feedwater supply, core uncovery would take place in one to three hours. For this scenario, rapid loss of inventory from the primary system would take place

through the pressurizer valve, since no heat sink would be available to keep the pressure down. Battelle Columbus has studied accidents sequences for the Sequoyah plant, which has an ice-condenser containment similar to OPS. Battelle has used the MARCH code to estimate independently that the core will become uncovered in three hours. The staff estimates that the assumption of failure of the turbine driven feedwater supply introduces a factor of 10^{-1} to 10^{-2} into the overall probability of the sequence, depending on the accessibility of the turbine controls, which varies widely from plant-to-plant.

We also note that similar sequences have been calculated in WASH-1400 (App. V, page 39) for the Surry Plant. Here failure of the turbine driven auxiliary pump was assumed. In the unlikely event that the sequence was completed, core melt could begin in two to three hours.

The staff believes that, considering the less restrictive conditions proposed to the applicant, his estimates are probably consistent with the others.

4. Discuss the reliability of the auxiliary feedwater system

STAFF RESPONSE

The estimates on unreliability by OPS for the three scenarios investigated are believed to be an appropriate characterization for the proposed AFWS design. Relative to those generic reliability perspectives derived recently for AFWS designs in 33 operating pressurized water reactor plants, the proposed FNP-AFWS would be as characterized being of high reliability.

With exceptions of the scenario involving total loss of AC (where only the steam turbine driven train would be available for automatic actuation), the expected dominant contributions to AFWS unreliability would be undetected human errors (pre-existing) that result in incorrectly positioned manual valves in the system. Such errors could arise from either incorrectly positioned valves in the suction portion of the system or from failure to correctly position valves following surveillance testing for pump operability. The staff believes that such human interaction potentials can be minimized through appropriate procedural and administrative controls being put into place prior to FNP operation. To this end, the staff suggests that those generic recommendations derived from the recent 33 plants AFWS evaluations be considered in establishing the appropriate procedural/administrative controls for operation of the FNP-AFWS.

5. Discuss how H_2 buildup in the ice condenser containment is dealt with following a TMI event and following a core melt

STAFF RESPONSE

The accident at Three Mile Island, Unit 2 (TMI-2) resulted in a substantial release of hydrogen gas due to an extensive zirconium-water reaction in the reactor core. A deflagration inside containment followed which produced a pressure buildup in the containment on the order of 28 psig. Since the TMI-2 containment structure was built to an internal design pressure of 60 psig, there was no loss of containment integrity. However, the immediate question arose as to the consequences of a TMI-2 type of event were it to occur in an ice condenser containment, which has both a smaller volume and a lower internal design pressure. Internal containment design pressures of an ice condenser vary between 12 and 15 psig.

We will first discuss the staff's position which includes the direction of our current efforts and the proposed basis for continued operation and licensing of nuclear power plants utilizing ice condenser containments. We will follow this with our critique of the OPS response to the ACRS concern. The OPS response, which considered 100% metal-water reaction, did not, however, address the consequences of a core melt. It is the OPS view that the prior work on core melt as reported in Topical Report No. 36A59, "FNP Core Ladle Design and Safety Evaluation" satisfies the present information requirements.

The OPS plant has a core ladle which is designed to hold the postulated molten core for two days. The applicant only addressed 100 percent metal water reaction and not a core melt accident since it was concluded in NUREG-0440, "Liquid Pathway Generic Study," that a core melt would violate containment integrity.

Staff Position

The accident at TMI-2 was one that exceeded the design basis for a nuclear power plant. The failure of the PORV was accompanied by operator error (turning off the safety injection pumps), procedural error (misalignment of the auxiliary feedwater control valves), design error (pressurizer water level indicator) and a host of deficiencies in the accident analyses. One of these deficiencies was the assumed five percent metal-water reaction in the reactor core. Nuclear powered plants are not designed to withstand multiple-failure events similar to that which occurred at TMI-2.

Although the TMI-2 accident exceeds the design basis, the staff has considered the consequences of a TMI-2 event, including 100% metal-water reaction, in an ice condenser containment. We are unable at this time to give a quantitative response to the question of hydrogen control. There exists a substantial number of uncertainties in such an analysis. Many of these are brought up in critique of the OPS response.

Qualitatively speaking, the assumption of 100% metal-water reaction seriously challenges containment integrity under almost any scenario. The introduction of high temperature, non-condensible hydrogen gas resulting from a complete metal-water reaction following a LOCA could possibly overpressurize and cause containment failure. The assumptions of deflagration and/or core melt with the introduction of additional non-condensible gases would almost certainly cause containment failure in an ice condenser.

In Chapter 3, particularly Section 3.3, of NUREG-0585 "TMI-2 Lessons Learned Task Force Final Report" there is a discussion of the need for and feasibility of hydrogen control features in all LWR's that would go beyond the current design bases specified in the NRC regulations. In recommendation 10 of that report, the Task Force recommended to the Director of Nuclear Reactor Regulation that the Commission give notice of intent to conduct rule making relating to the consideration of design features to mitigate degraded core and core melt accidents; in particular, systems for preventing the uncontrolled combustion of hydrogen that could be produced in such accidents. The Director of NRR has asked the ACRS to review and comment on the recommendations in NUREG-0585, and the Office is currently reviewing the recommendations in context with those of the President's Commission on the accident at Three Mile Island and others. The results of that review will be presented to the Commission for decision. A decision on whether and how to proceed with the proposed rule making is not expected to be made by the Commission for several months.

The staff proposes to defer imposition of requirements on hydrogen control beyond present requirements pending a decision by the Commission on the proposed rule making. Of primary importance will be the new metal-water reaction rate to be assumed. In the course of rule making, we would expect to consider the feasibility of various alternatives, such as inerting, filtered venting, and controlled burning of hydrogen.

We believe deferral of further action at this time relative to hydrogen control is justified, particularly for a manufacturing license. The design of the FNP will be required to accomodate the accident prevention measures currently being introduced by the Lessons Learned and Bulletins and Orders Task Forces, those recommended by the President's Commission, and others. These measures include changes in safety equipment design, operating training, accident response, and diagnostic instrumentation to reduce the probability of future accidents which, like TMI-2, might exceed the current design basis and produce large amounts of hydrogen. On this basis, we proposed that for an interim period, until rule making can be conducted, the licensing process for FNP can continue to be conducted in accordance with the current regulations and guides for the design and installation of post-accident containment combustible gas control system, i.e., paragraph 50.44 to 10 CFR Part 50 and Regulatory Guide 1.7. During this interim period the staff does not foresee that viable alternatives will be foreclosed.

Critique of Offshore Power Systems Response

Offshore Power Systems has concluded that core damage similar to that at TMI-2 with burning of hydrogen or 100 percent metal-water reaction in the core without burning of hydrogen would result in a containment pressure of 40 psig. They also concluded that the ice condenser containment would remain intact at 40 psig. The pressure resulting from 100 percent metal-water reaction and burning of hydrogen would cause containment failure.

The applicant concluded that the containment would not experience gross failure at 40 psig. However, we believe that leakage around the penetrations may reach unacceptable levels at pressures as low as 22 psig due to failure in a gradual, ductile manner. Moreover, the increased leakage of containment penetrations due to pressurization above the design limits will have to be evaluated in the analysis of hydrogen buildup inside containment.

The applicant's conclusion that the containment would remain intact with 100 percent metal-water reaction and no burning of hydrogen implies that preventing the burning of hydrogen by inerting the containment would enable the containment to survive the 100% metal-water reaction. This conclusion is based on the applicant's calculation that the containment pressure will reach a maximum of 40 psig. The containment pressure attained for 100 percent metal-water reaction without hydrogen burning depends on the following information:

a. Amount of water available in the vessel for metal-water reaction and steam generation;

b. The portion of chemical energy transferred to hydrogen and to steam;

c. The chemical energy release as a function of time;

d. The amount of chemical energy absorbed by heat sinks (vessel, piping, etc.) before it enters the containment;

e. The amount of ice available for pressure suppression; and

f. The number of trains of the containment spray system available for pressure suppression.

In calculating the containment pressure of 40 psig for 100 percent metal-water reaction without hydrogen burning, the applicant made the following assumptions in order to account for the above information:

a. The applicant assumed 1.4×10^7 Btu's was absorbed by the hydrogen. In addition, the applicant assumed that the total chemical reaction energy was absorbed by steam and transported into the containment. (This appears to involve a double accounting for ten percent of the chemical energy.) With this assumption, the applicant accounted for the chemical energy absorbed by the hydrogen and steam and the amount of water available in the vessel;

b. The chemical energy was released at a constant rate over a period of one hour;

c. All of the chemical energy was released to the containment;

d. Approximately twenty-five percent of the ice was available for pressure suppression;

e. Three out of four containment spray trains were available for pressure suppression.

The applicant's assumption on the amount of energy absorbed by the hydrogen and the steam was based on the hydrogen leaving the reactor vessel at 1800°F. Hydrogen at 1800°F accounts for 1.4×10^7 Btu's of the chemical energy. This assumption may be non-conservative for that accident wherein only that amount of water needed for the metal-water reaction is available in the core.

If there was not sufficient water available for steam generation in the reactor, the hydrogen produced would absorb a larger portion of the energy produced by the metal-water reaction. The temperature of the hydrogen entering the containment would then be higher than 1800°F which results in a containment pressure higher than 40 psig.

The applicant's assumption that the hydrogen is generated over a period of one hour is dependent on the type of accident that causes the core damage. If the hydrogen were generated over a shorter period of time, a containment pressure greater than 40 psig would be expected.

The applicant assumed none of the chemical energy was absorbed by heat sinks on the path out of the reactor vessel. This is the most conservative assumption they could make in this area.

The applicant's assumption that twenty-five percent of the ice was available for pressure suppression may be non-conservative for some accident scenarios. Less ice may actually be available resulting in higher containment pressures.

After a core meltdown, more than one train of the containment spray system may be lost due to debris from the accident being pumped through the system and damaging the pumps. If less than three out of four containment spray system trains were available, the resulting containment pressure would be above the predicted value of 40 psig.

The impact of the assumptions on the containment pressure depends on the actual accident scenario. Numerous accident scenarios will have to be studied to determine which one results in the maximum containment pressure for the case of 100 percent metal-water reaction without hydrogen burning.

We do not presently have sufficient information to verify the applicant's conclusions on the effects of a postulated accident involving 100 percent metal-water reaction without hydrogen burning. Moreover, we do not have sufficient staff resources in the short term to analyze the pressurization of ice condenser containments due to 100 percent metal-water reaction without hydrogen burning for various accident scenarios.

Based on a cursory review of the applicant's analyses, we conclude that there are too many uncertainties in the applicant's assumptions to place much credence on the associated conclusions.

6. Discuss how the FNP compensates for the difficulty, due to the remote location and the lack of space available in improvising new systems and techniques in case of an accident

STAFF RESPONSE

We agree with the OPS response. It should be noted, however, that over the past several months following the Three Mile accident, the staff has been conducting an intensive review of the design and operational aspects of power plants and the emergency procedures for coping with potential accidents. The purpose of these efforts was to identify measures that should be taken in the short-term to reduce the likelihood of such accidents and to improve the emergency preparedness in responding to such events. To carry out this review, efforts were established in four areas: (a) licensee emergency preparedness, (b) operator licensing, (c) bulletins and orders followup (primarily in the areas of auxiliary feedwater systems reliability; loss of feedwater and small break loss-of-coolant accident analysis; emergency operating guidelines and procedures) and (d) Short-Term Lessons Learned. The results of these efforts are a set of requirements that the staff has recommended for implementation. The Commission may add to or modify these staff positions after reviewing them. Additional staff requirements may be developed as the Lessons Learned Task Force completes its long-term recommendations.

Efforts are underway within the NRC to review all aspects of emergency planning, including the adequacy of present planning and the need for coordination with and participation of other agencies in developing emergency planning. Appendix C outlines the requirements developed to date resulting

fission products would be swept out of the reactor compartment as a step towards their possible ultimate release. Counteracting this is the reduced generation of gases due to the altered chemical composition of the melt, such that the net effect of the ladle is to reduce mass flow from the reactor compartment.

Exceptions to this generalization are those accidents in which water at late times in the accident is introduced into the compartment, and for these cases the adverse volume effect is less favorable by only a very small fraction.

(1 vs 2) The higher temperatures of melts in the ladle design increase the molar entropy of fission product vapors in thermal equilibrium with the melt. Counteracting this is the increased molar entropy of fission products within the melt due to dilution by MgO and the much larger amount of molten steel in the ladle as opposed to the smaller concrete-basalt melt. For those species having pure phase boiling points above about 2500° K the net effect would be expected to be a reduction in equilibrium vapor pressures due to dilution in the ladle, while for the more volatile species, such as cesium, the higher temperatures in the ladle would lead to higher equilibrium vapor pressures. Since no non-condensible gas generation is expected in the ladle design, there is little driving force available to remove fission product vapors from the reactor compartment for the formation of aerosols, hence little, if any, adverse affect upon airborne source terms.

The core ladle design allows a much higher energy density to occur, since it results in the storage of latent and internal energies (heat of fussion and heat capacity integrated over temperature) which, in the

concrete mat design, are discharge by pyrolysis and volatilization of the concrete and comparatively rapid ultimate discharge to the water beneath the hull should water come into contact with the melt to procduce a steam explosion. the thermodynamic efficiency (conversion of heat energy to mechanical work) could be higher with the higher energy density in the ladle. The probability of a steam explosion, however as well as the conversion efficiency is a complex function of a large number of functions which were previously documented in NUREG-0440, "Liquid Pathway Generic Study," and also discussed with the ACRS. It is the staff's conclusion that neither the probability nor the thermal to mechanical work compression efficiency will be changed in any appreciable way by the presence of the core ladle.

Temperatures in the reactor compartment when it is largely lined with magnesia would be of the order of hundreds of degrees hotter than if unlined. Significant vapor pressures of silver control rod material and steel components could be maintained. The generation and condensation elsewhere of these vapors could constitute a significant heat transport mechanism from the melt. There would be larger thermal and concentration gradients within the volumes and openings connecting the reactor compartment to the remainder of the containment. Diffusive and Soret effect transport in these connecting volumes could lead to aerosol formation by condensation and reaction of metal gases with the containment atmosphere. In addition, the molar volumes of gases within the reactor compartment would be several times larger than those in the upper containment, while the mean molecular weights would be only two to three times larger, making possible connective transport. These effects are also present in the concrete mat design, but to a lesser degree.

Each of the effects outlined above have differences between ladle and concrete mat designs which, to some unknown extent, counteract one another in their roles in determining the expected air release source

from the staff's Emergency Preparedness Studies. Further, the Commission has initiated a rule making procedure, now scheduled for completion in January 1980 in the area of Emergency Planning and Preparedness. Additional requirements are to be expected when rule making is completed and some modifications to the emergency preparedness requirements contained in the Appendix may be necessary. Moreover, an NRC-EPA Task Force Report, NUREG-0396 dated December 1978 recommended 10- and 50-mile emergency planning zones and the Commission has endorsed this recommendation.

The results of our ongoing studies and rule making hearings will be applied to the FNP design as well as to the utility-owner for site dependent matters.

7. Discuss how one faces lack of flexibility for design changes due to the compactness and lack of available space on the FNP

STAFF RESPONSE

We agree with the OPS response. The consideration of compactness and lack of space has been raised as early as 1971 during our preapplication review. We have recognized this aspect and have considered it in our subsequent review and evaluation and we have not found lack of space to be a design constraint at this time.

C. Items Concerning The Effects Of Changing Base Mat Material

1. Discuss the effects of changing the base mat from concrete to magnesium oxide on the probability of a major air release during a core melt accident. Discuss the comparisons of probabilities and dose levels for air releases associated with concrete and magnesium oxide during a core melt accident.

STAFF RESPONSE

Large amounts of airborne radio-isotopes can be dispersed outside containment by the release of either gas or liquid to the environment. Gaseous radioisotopes or aerosol particles can be released in a gas phase, or, as at TMI-2, radio-isotopes dissolved in liquid phase can be released to subsequently emit gaseous radio-isotopes. To the extent that the presence of the core ladle delays or inhibits one or more steps in such releases, it may be considered to reduce the probability of release by the affected mechanisms, as noted in the applicant's response. Of greater interest is the possible existence of release mechanisms unique to the ladle design.

The presence of a core ladle alters the possible release mechanisms during a core melt accident by the following means:

1. The radio-isotopic inventory is maintained for a longer time and at a higher temperature within the reactor compartment.
2. The total amount of material melted is increased, and its chemical composition altered.
3. The free volume of the reactor compartment is reduced by the excess volume of the ladle over that of the concrete.

To some extent, these differences between core ladle and concrete mat design counteract one another.

(2 vs 3) The lower free volume means that for a given accident- induced flow through or from the reactor compartment a larger fraction of volatile

term. The effects differ in the two designs only by degree, such that they offer no airborne release mechanism unique to the ladle design. On balance, the choice between a concrete base mat or an MgO core ladle does not substantially affect the airborne release probability or the inventory of radio-isotopes susceptible to release, except insofar as the ladle design inhibits or delays eventual breach of the FNP hull. Since this exception is significant, the applicant's unquantified assessment of overall risk reduction appears warranted.

2. Discuss the consideration given to the use of a vented containment. Discuss the consideration given to the use of sea water for venting and/or cooling a molten core.

STAFF RESPONSE

A vented containment design accepts small, controlled leakage in exchange for reduction of the likelihood of massive uncontrolled release from containment failure. Venting in principle can protect containment from failure due to steam or hydrogen flame burning overpressurization, but not from failure due to detonation. Detonations are, however, inherently less likely than less violent, though rapid, pressurizations, since the formation of strong shocks is possible only under restricted circumstances.

Hydrogen gas is much more easily dissociated, either thermally or by ionizing radiation, than oxygen, nitrogen, or water vapor. It is therefore, susceptible to spontaneous ignition when in locally high concentration (diffusion flame conditons) and correspondingly less likely to form detonable mixtures throughout the containment. Slow hydrogen generation, on the other hand, is controllable by the hydrogen recombiners.

Steam explosions due to the rapid mixing of water and molten material are less likely to contribute containment failure than more slower overpressurization events (See NUREG-0440, "Liquid Pathway Generic Study," pages A-16 to A-24).

The accident sequences leading to containment over pressurization are, therefore, dominated by comparatively slow pressure increases, which are susceptible to mitigation by containment vent systems.

Seawater is generally slightly alkaline (pH8), and thus could serve as a reducing agent for iodine. In addition, it contains a mean concentration of 50 mg per tonne of natural iodide and could function well as an isotopic exchange reservoir for vented radio-iodines. If gas were vented into seawater at depth, i.e., under pressure, the solubility of xenon in seawater would also be significant. Venting to seawater could, therefore, reduce the potantial for airborne release, although at a cost of unnecessary seawater contamination during accidents which did not challenge containment integrity.

The chloride ion concentration of seawater renders it unsuitable for use within containment as a coolant due to corrosive effects on steel and concrete. Sufficient feedwater is present on the barge to cool the molten core if provisions were made for this purpose.

The simplest and most direct method of venting to the sea would be a well or standpipe communicating the containment to the underside of the hull. A skirt or inverted wall surrounding the bottom of the barge could then capture vented gases and guarantee that containment pressure would never rise above the hydrostatic pressure on the hull bottom.

3. Discuss the change in position for allowing the FNP to be placed on riverine and estuarine sites. Has the proposed installation of the core ladle changed the NRC Staff's position on this matter, if so, why? What actions and in what time period, are considered practical to isolate the core for a riverine or estuarine site?

STAFF POSITION

The NRC staff position related to the generalized siting of FNPs in estuarine and riverine areas has remained unchanged throughout the course of the staff's environmental review.[1,2,3] This position is that " ... finding acceptable FNP sites in estuaries, rivers, or near barrier islands, will most likely be extremely difficult, but [the staff] cannot conclude that there are no acceptable estuarine, riverine or barrier island locations for FNP emplacement when appropriate mitigative actions are taken."[4] Both the staff and the U.S. EPA concluded that siting FNPs in such areas could produce a significant potential for adverse environmental impact, particularly with actions associated with construction and maintenance dredging. Furthermore, in its assessment of the FNP core-melt accident at an estuarine or riverine site, the staff concluded that a direct release of radioactive material to such areas would result in unacceptable consequences to the environment[4]. As such, the staff, in consultation with the U.S. EPA, has concluded that applicants who wish to site FNPs at specific locations (including sites in estuaries and rivers) must comply with certain environmental siting requirements including specific mitigative actions to limit the environmental consequences of a core-melt accident at an estuarine/riverine sited FNP.

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[1] FES, Part II

[2] FES, Part II Addendum

[3] FES, Part III (NUREG-0502)

[4] FES, Part III (NUREG-0502) p. xiv

The proposed installation of the core ladle in the FNP did not change the NRC staff's position regarding the acceptability of FNP siting in estuarines, rivers or near barrier islands.

Environmental siting requirement 1.B reproduced below from the FES, Part III must be complied with by an applicant who wishes to locate an FNP at a specific site in an estuary, river or near a barrier island and since it relates to specific site conditions it was not imposed as a condition of the manufacturing license application.

Environmental Siting Requirement 1.B

"Proposed FNP sites in estuaries, river or near barrier islands must be appropriately modified in an environmentally acceptable manner such that in the event of a core-melt accident, the release of radioactive material into the surrounding water body shall be limited to levels that will not result in undue impact to man or the ecosystem."

With respect to actions and time periods considered practical to isolate the core for river and estuary sites, the staff concluded that total isolation of radioactive core-debris from open estuarine/riverine waters, following a core-melt accident would be very difficult to achieve. Furthermore, the staff concluded that total isolation would not be necessary, provided the combination of site characteristics, FNP design features and interdiction methods could provide adequate assurance that a core-melt type accident would not produce risks any worse than a typical land-based plant at a river or estuary site. Thus the staff required (siting requirement 1.B) that an FNP site in such areas must be modified to restrict the potentially widespread and chronic release of radioactivity in the event of a core-melt accident. Siting requirement 1.B is stipulated independently of manufacturing license condition No. 4 which requires

that the FNP be redesigned to incorporated a core ladle.[5] The core-ladle design would provide additional delay before potential melt-through beneath the reactor vessel in order to provide additional time to incorporate interdictive measures, but in the event of an actual melt-through, radioactive debris would undoubtedly be released to the ambient estuarine/riverine environment. This would, in the staff's view, produce unacceptable environmental impacts.

Environmental siting requirements 1.B is intended to prevent waterborne contaminants resulting from core-melt type accidents from spreading offsite in an uncontrolled manner. The bases for the requirement included consideration of mitigation and interdiction techniques that could be employed at both land-based and FNP sites to limit the offsite migration of activity into the estuary or river and reduce the long-term environmental consequences of such releases. The environmental consequences in most estuary and river siting situation were judged likely to produce both acute and chronic effects on biota due to the generally very slow natural pollutant flushing capability of such water bodies. Classes of aquatic biota might be destroyed, therefore impacting the ecosystem for years. A direct result of such chronic conditions upon biota would be an indirect effect upon man due to relatively long-term public restriction of water resource related activities on a large scale.

So as to implement this environmental siting requirement, the applicant has proposed an additional plant-site interface criterion in their FNP Core Ladle Topical Report[6]. The criterion requires that site modifications be made at proposed

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[5]FES, Part III, p. xv

[6]FNP Core Ladle Design and Safety Evaluation, Offshore Power Systems, Topical Report No. 36A59, April 1979, p. VI-2.

specific FNP sites in estuaries and rivers to ensure that the environmental consequences of an FNP core-melt accident in these areas would be no worse than those for estuary sited typical land based plants considered in the LPGS Report. The staff has accepted this criterion, noting that the consequences of core-melt type accidents should be assessed for any proposed estuary or river FNP site. The assessment will consider specific FNP site and plant design information for comparison with typical land based reactor sites in estuaries and rivers. Thus, at this time no specific sections in a given time period have been specified.

4. Discuss the NRC Staff's position that the FNP Core Ladle is considered an environmental issue and not a safety issue.

STAFF POSITION

The following background is needed in order to place the response to this ACRS request in perspective. Under its present mandate the NRC assesses the implications of licensing nuclear power plants under two Acts: the Atomic Energy Act of 1954 as amended (i.e., protection of the public health and safety) and the National Environmental Policy Act of 1969 (NEPA) (i.e., protection of the environment and overall cost-benefit balancing).

Pursuant to the Atomic Energy Act, 10 CFR Part 50 of the Commission's regulations was issued in the mid 1950's and formed the basis for the staff's analysis of the safety of proposed nuclear power plants. Subsequently, in August 1974, the Commission issued an interim statement of policy in the Federal Register concerning the treatment of postulated accidents in the staff's safety reviews. The Commission stated:

> "In the approach to safety reflected in the Commission's regulations, postulated accidents, for purposes of analysis, are divided into two categories -- "credible" and "incredible." The former ("credible") are considered to be within the category of design basis accidents. Protective measures are required and provided for all those postulated accidents falling within that category, and proposed sites are evaluated by taking into account the conservatively calculated consequences of a spectrum of severe postulated accidents. Those accidents falling within the "incredible" category are considered to be so improbable that no such protective measures are required."

Using this statement as a basis, the staff judged that a core-melt accident fell into the "incredible" category and therefore would not be considered in its safety evaluations prepared for the licensing of nuclear plants, including the FNP. This reasoning is based upon the staff's perception that the FNP nuclear system design is similar to that of land-based plants, and thus the probability of occurrence of a core-melt accident was viewed as equivalent (i.e., incredible)

for both types of siting options. The scope of the staff's safety review relative to the core ladle centers only upon the question of whether incorporation of the core ladle in the FNP would alter previous staff conclusions regarding the overall safety of the FNP design.

The Commission's implementing regulations for NEPA are contained in 10 CFR Part 51 but there is no specifically approved Commission regulation for the consideration of accidents under NEPA. The NRC has historically considered the potential environmental consequences of plant accidents in the manner prescribed in the proposed Annex A to 10 CFR Part 50, Appendix D.

The proposed Annex is a part of an AEC proposed regulation to implement NEPA. The Annex was issued for public comment in December 1971, but no final Annex has been prepared. Subsequently, the Commission replaced Appendix D to 10 CFR Part 50 with 10 CFR Part 51 which specifically addresses the NRC consideration of NEPA issues. Technically, the rulemaking proceeding for the Annex is still pending before the NRC, and while the Commission has never formally adopted the Annex, it authorized its use as guidance.

The proposed Annex divided radiological accidents into nine classes for NEPA evaluation purposes. With respect to the ninth class (Class 9 accidents), the Annex concluded that applicants would not be required to discuss such accidents in their Environmental Reports since the probability of occurrence was so low as to make the risk negligible.

With regard to the OPS application for FNP's, the staff found that the FNP design offered a departure from land-based siting, and the potential environmental

consequences from a core-melt type accident could differ in type and magnitude from those ascribed to land-based plants. Therefore the environmental review of the core-melt accident at the FNP need not be guided by the proposed Annex.

With this information, one can specifically respond to the ACRS request. Since the Commission Policy Statement precluded the review of the FNP-core-melt accident pursuant to the Atomic Energy Act, the staff decided to evaluate such an accident under i.e., within the bounds of the FNP environmental review, pursuant to 10 CFR Part 51 of the Commission's regulations which implements NEPA. Various factors weighed heavily in the staff's decision to consider this accident under NEPA. These included: (a) The novel siting option of the FNP led to FNP core-melt consequences different from LBP core-melt consequences; (b) the staff position that the proposed Annex A to 10 CFR Part 50, Appendix D, did not apply to FNPs; and (c) NEPA's mandate to disclose to the fullest extent possible the consequences of major federal actions.

(a) <u>FNP core-melt consequences different from LBP core-melt consequences</u>. Early in its review of the FNP concept the staff perceived that the core-melt accident at an FNP could result in environmental consequences different from those for a similar accident at an LBP. From a core melt accident viewpoint, the FNP did not offer the same degree of natural isolation as a LBP, i.e., a core-melt accident at the FNP could probably result in a prompt release of radioactive debris into the water which then could be diffused by currents and tides. In the LBP, such an accident would probably result in the retention of core-debris in the earth with significantly different liquid pathway impacts. This staff perception together with the ACRS concerns expressed in their letter of

November 1972, to the AEC, prompted the NRC to initiate the LPGS in order to compare the design basis and core-melt liquid pathway risks for FNPs and LBPs.

(b) Proposed Annex A to 10 CFR Part 50, Appendix D did not apply to FNPs*

The staff reasoned that the FNP could produce core-melt environmental consequences different in kind from LBPs (i.e., liquid pathway consequences) and undoubtedly different from those considered when the proposed Annex was being developed. Further, the FNP concept was not specifically considered by the Commission when it issued the proposed Annex in 1971 for public comment. The staff concluded, therefore, that the policies set forth in the proposed Annex were not applicable to FNPs and that an evaluation of the environmental impacts from core-melt accidents was a proper topic for staff consideration in the generic environmental impact statement for FNPs.

(c) NEPA Mandate of Full Disclosure

The staff's position is that NEPA requires a federal agency to fully disclose all pertinent environmental information including controversial or opposing views, within an environmental impact statement such that the decision-makers and public are fully informed. The staff, having concluded in the LPGS Report that the liquid pathway consequences at an FNP differed significantly from the LBP counterpart, was obligated under the intent of NEPA to fully assess the environmental implications of such a finding in the staff's FES, Part III.

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*This position is supported by the Commission's Memorandum and Order of September 14, 1979 concerning the certified question: "Are Class 9 accidents a proper subject for consideration in the staff's environmental statement on the floating nuclear power plant manufacturing license application?"

In summary, the proposed core ladle requirement is a direct result of the staff's decision to consider "class 9" or core-melt type accidents as part of the environmental review for the FNP manufacturing license application. The aff views the proposed FNP core ladle design requirement set forth in the FES. Part III (NUREG 0502) as both an environmental and safety issue. The genesis and imposition of the core ladle requirement, however, is based solely on the staff's environmental assessment (FES, Part III) and the LPGS Report (NUREG-0440). The overall safety implications of incorporating such a feature into the FNP is currently under evaluation by the staff.

D. Additional Information Requests From The NRC Staff

1. Provide available information on the Sandia 100 plant liquid pathway study

STAFF RESPONSE

The Liquid Pathway Study at Sandia has not yet produced useful results. If and when the study produces results, we will make them available. We anticipate some early results in the next few months; however, the numerical uncertainties are expected to be so large as to make the results useful only for directing areas of further study.

2. Provide available information on the WASH-1400 type study of the ice condenser type plant, along with a comparison for non-ice condenser type plants

STAFF RESPONSE

A draft Sandia report on the ice condenser exists at this time and was provided to D. Okrent of the ACRS in July 1979. A comparison study with other non-ice condenser plants in underway but is currently stopped because of higher priority work. That report (comparison study) is not expected to be in draft form before about June 1980.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 25, 1979

Harold R. Denton
Director, Office of Nuclear Regulatory Regulations

SUBJECT: ACRS REVIEW OF THE FLOATING NUCLEAR PLANT CORE LADLE DESIGN

At the June 27, 1979 ACRS Subcommittee Meeting on the Floating Nuclear Plant, members of your staff requested that the ACRS meet at an early date to discuss the proposed FNP Core Ladle Design and to write a letter to Mr. Gossick commenting on that preliminary design prior to the NRC Staff's issuance of its safety evaluation. The Acting ACRS Subcommittee Chairman informed your staff and representatives of Offshore Power Systems that the suggestion to hold an early ACRS meeting would be considered at the July 1979 ACRS meeting.

The proposal to hold an early ACRS review of the conceptual design of the FNP core ladle was discussed at the July 1979 ACRS meeting. It was decided that additional information, as indicated below, is necessary before the Committee can proceed with its review of the FNP.

a. Items Related to the Impact that the Core Ladle Will Have on Other Containment Structures

1. Calculate the fraction of decay heat radiated from the pool for the proposed design.

2. Calculate the effects of heat radiation in Item 1 on the rate of:

   (a) disintegration and collapse of exposed concrete

   (b) disintegration and collapse or melting of concrete behind the 6 inch magnesite brick wall

   (c) collapse of steel from the reactor cavity.

3. Discuss the consequences of Item 2 with respect to:

   (a) loss of integrity of superstructures

   (b) loss of hearth capacity

(c) impact resistance of the hearth and its supports

(d) integrity of structural steel members.

4. Discuss the stability of the 6 inch magnesite brick wall above the hearth level with respect to:

   (a) loss of brick by spalling

   (b) differential motion with respect to the hearth, concrete walls, and anchors

   (c) loss of concrete behind the wall by spalling, disintegration, and melting at calculated temperatures, or at temperatures indicated in Fig IV-6 of OPS Topical Report No. 36A59

   (d) slagging reaction between the brick walls and melted concrete.

5. Discuss the fluxing of magnesite brick by siliceous material falling into the hearth.

6. Discuss the properties and merits of basalt as a concrete aggregate.

7. Discuss the possibility of the heat flux being higher on the sides of the molten mass than on the bottom (FRG conclusion for concrete melt) with melting going horizontally faster than vertically.

b. <u>Items Related to Three Mile Island Accident</u>

1. Discuss the possibility of the Upper Head Injection System releasing nitrogen into the primary system and impeding the ability to establish or maintain natural circulation.

2. Discuss the acceptability of the single failure criterion.

3. Discuss the timed sequence of events upon the loss of all AC power before core damage will result.

4. Discuss the reliability of the auxiliary feedwater system.

5. Discuss how H_2 buildup in the ice condenser containment is dealt with following a TMI event and following a core melt.

6. Discuss how the FNP compensates for the difficulty, due to the remote location and the lack of space available, in improvising new systems and techniques in case of an accident.

7. Discuss how one faces lack of flexibility for design changes due to the compactness and lack of available space on the FNP.

c. Items Concerning the Effects of Changing Base Mat Materials

1. Discuss the effects of changing the base mat from concrete to magnesium oxide on the probability of a major air release during a core melt accident. Discuss the comparisons of probabilities and dose levels for air releases associated with concrete and magnesium oxide during a core melt accident.

2. Discuss the consideration given to the use of a vented containment. Discuss the consideration given to the use of sea water for venting and/or cooling a molten core.

3. Discuss the change in position for allowing the FNP to be placed on riverine and estuarine sites. Has the proposed installation of the core ladle changed the NRC Staff's position on this matter, if so, why? What actions and in what time period, are considered practical to isolate the core for a riverine or estuarine site?

4. Discuss the NRC Staff's position that the FNP Core Ladle is considered an environmental issue and not a safety issue.

d. Additional Information Requested From the NRC Staff

1. Provide available information on the Sandia 100 plant liquid pathway study.

2. Provide available information on the WASH-1400 type study of the ice condenser type plant, along with a comparison for non-ice condenser type plants.

Following receipt of Offshore Power System's response to the items listed above and a written evaluation by the NRC Staff, another ACRS Subcommittee meeting will be held. Please advise us of the date by which you believe the above information will be available so we can schedule related ACRS activities.

R. F. Fraley
Executive Director

cc: D. Muller, DSE
E. Case, NRC
D. Vassallo, DPM
F. Schroeder, DSS

APPENDIX B OF APPENDIX F

Offshore Power Systems

8000 Arlington Expressway
Box 8000, Jacksonville, Florida 32211

904 724-7700
Telex 568406

September 14, 1979

Mr. Robert L. Baer, Chief
Light Water Reactors Branch No. 2
Division of Project Management
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20852

S. Haga

Licensing & Licensing

Re: Docket STN 50-437; ACRS Questions on Core Ladle and TMI-2

Dear Mr. Baer:

Transmitted herewith are 20 copies of the Offshore Power Systems responses to the ACRS Subcommittee questions contained in R. F. Fraley's letter to H. R. Denton dated July 25, 1979. Please note that we have not offered responses to part d. of Mr. Fraley's letter as these requests were made specifically to the NRC Staff. By copy of this letter, 20 copies of our responses are being transmitted directly to Mr. Fraley for distribution within ACRS.

Certain material in the attached responses reflects modification to the design presented in OPS Report 36A59, "FNP Core Ladle Design and Safety Evaluation". The principal changes are increased ladle volume and increased refractory insulation on the walls of the reactor cavity. Both of these changes resulted from our ongoing evaluation of radiant upheating from the pool surface. The analyses of radiant upheating, which are described in the attached responses, are believed to be adequately conservative to show feasibility and therefore to support the issuance of the Manufacturing License. Following NRC Staff review those responses which affect the present content of Report 36A59 will be retransmitted in the form of a revision to that report.

We ask that these responses be reviewed on an expedited basis leading to an ACRS Subcommittee meeting as early as October

1979. To this end, we are prepared to offer any assistance the Staff may require.

Very truly yours,

P. B. Haga

P. B. Haga

/lel

Attachments

CC: R. F. Fraley (ACRS)
V. W. Campbell
A. R. Collier

NEAR TERM REQUIREMENTS FOR IMPROVING EMERGENCY PREPAREDNESS

While the emergency plans of all power reactor licensees have been reviewed in the past for conformance to the general provisions of Appendix E to 10 CFR Part 50, the most recent guidance on emergency planning, primarily that given in Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants", has not yet been fully implemented by most reactor licensees. Further, there are some additional areas where improvements in emergency planning have been highlighted as particularly significant by the TMI-2 accident.

We plan to undertake an intensive effort over about the next year to improve licensee preparedness at all operating power reactors and those reactors scheduled for an operating license decision within the next year. This effort will be closely coordinated with a similar effort by the Office of State Programs to improve State and local response plans through the concurrence process and the efforts of the Office of Inspection and Enforcement to verify proper implementation of licensee emergency preparedness activities. Further, the Commission has initiated a rulemaking procedure, now scheduled for completion in January 1980, in the area of Emergency Planning and Preparedness. Additional requirements are to be expected when this rulemaking is completed and some modifications to the emergency preparedness requirements contained in this letter may be necessary.

Our near term requirements in this effort are as follows:

(1) Upgrade licensee emergency plans to satisfy Regulatory Guide 1.101, with special attention to the development of uniform action level criteria based on plant parameters.

(2) Assure the implementation of the related recommendations of the Lessons Learned Task Force involving instrumentation to follow the course of an accident and relate the information provided by this instrumentation to the emergency plan action levels. This will include instrumentation for post-accident sampling, high range radioactivity monitors, and improved in-plant radioiodine instrumentation. The implementation of the Lessons Learned Task Force's recommendations on instrumentation for detection of inadequate core cooling will also be factored into the emergency plan action level criteria.

(3) Determine that an emergency operations center for Federal, State and local personnel has been established with suitable communications to the plant, and that upgrading of the facility in accordance with the Lessons Learned Task Force's recommendation for an in-plant technical support center is underway.

(4) Assure that improved licensee offsite monitoring capabilities (including additional thermoluminescent dosimeters or the equivalent) have been provided for all sites.

(5) Assess the relationship of State/local plans to the licensees' and Federal plans so as to assure the capability to take appropriate emergency actions. Assure that this capability will be extended to a distance of ten miles. This item will be performed in conjunction with the Office of State Programs and the Office of Inspection and Enforcement.

(6) Require test exercises of approved emergency plans (Federal, State, local and licensees), review plans for such exercises, and participate in a limited number of joint exercises. Tests of licensee plans will be required to be conducted as soon as practical for all facilities and before reactor startup for new licensees. Exercises of State plans will be performed in conjunction with the concurrence reviews of the Office of State Programs. As a preliminary planning bases, assume that joint test exercises involving Federal, State, local and licensees will be conducted at the rate of about ten per year, which would result in all sites being exercised once each five years. Revised planning guidance may result from the ongoing rulemaking.

APPENDIX D OF APPENDIX F

REFERENCES

1. R. D. Gasser and W. T. Pratt, "MELSAC - A Computer Code to Determine the Thermal Response of a Sacrificial Bed and Surrounding Structures to a Core Melt Event," to be published as a BNL Report.

2. W. T. Pratt and R. D. Gasser, "Thermal Analysis of an FNP Sacrificial Bed," to be published as a BNL Report.

3. F. A. Kulacki and R. J. Goldstein, "Thermal Convection in a Horizontal Fluid Layer with Uniform Volumetric Energy Sources," J. Fluid Mech., Vol. 55, Pt. 2, pp. 271-287, (1972).

4. R. Faradieh and L. Baker, Jr., "Heat Transfer Phenomenology of a Hydrodynamically Unstable Melting System," J. Heat Transfer, Vol. 100, Pt. 2, pp. 305-310, (1978).

5. Letter from P. B. Haga (OPS) to R. L. Baer (NRC), "Docket STN 50-437; ACRS Questions on Core Ladle and TMI-2," September 14, 1979.

6. For a description of the original GROWS code, see Chapter V of ANL/RAS 74-29, October 1974.

7. W. T. Pratt, "Trip Report on GROWS Code Meeting held at ANL on July 10, 1979," a memorandum to R. A. Bari, Brookhaven National Laboratory, August 9, 1979.

APPENDIX G

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 1 1979

MEMORANDUM FOR: Themis P. Speis, Chief
Liquid Metal Fast Breeder
Reactors Branch

FROM: Mel Silberberg, Chief
Experimental Fast Reactor
Safety Research Branch

SUBJECT: TRANSMITTAL OF SANDIA COMMENTS ON OPS TOPICAL REPORT 36A59 (Docket No.: STN 50-437)

Enclosed for your information and use are the comments from Sandia Laboratories on their technical review of the Offshore Power Systems Topical Report No. 36A59, "FNP Core Ladle Design and Safety Evaluation" (April 1979).

We have reviewed the Sandia comments and have noted several aspects of the Sandia review which are of interest:

- identification of several areas of technical uncertainty which were not considered by OPS

- potentially applicable experience from the glass-making industry (oxide melts) not cited by OPS, merits consideration as an information source

- the potential importance of chemical interactions of oxide melts, other than iron oxide, with MgO merits further evaluation

We believe that the comments provided by Sandia will be of considerable value to the formulation, planning and implementation of research programs directed towards an FNP core retention device. In this regard, in Section V the applicant refers to test programs planned by NRC, DOE, EPRI and foreign countries, as sources for providing information needed for final confirmation of ladle design. In Section V, however, only the current NRC program is briefly outlined. It would be most beneficial to

Themis P. Speis JUN 1 1979

have the applicant provide descriptions of similar test programs from the other sources noted. Such information will allow us to selectively plan the ARSR test program in the most timely, prudent and cost effective manner to meet NRC needs for FNP.

M. Silberberg

M. Silberberg, Chief
Experimental Fast Reactor
Safety Research Branch

Enclosure: As stated

cc: T. Murley, RSR w/o encl.
C. Kelber, RSR w/o encl.
L. Rib, RSR w/o encl.
T. Walker, RSR w/o encl.
W. Johnston, RSR w/o encl.
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G. Quittschreiber, ACRS w encl.
S. Sohinki, ELD w/encl.

REVIEW OF INFORMATION NEEDS FOR DESIGN

OF A MgO CORE RETENTION DEVICE

SANDIA LABORATORIES

MAY 23, 1979

Information Needs for Core Retention Device Design and Analysis

Section V of Offshore Power Systems Topical Report Number 36A59 does a creditable job in identifying areas of uncertainty concerning melt/core retention material interactions. The reviewers were able to identify some additional concerns and had comments on items cited in these documents.

Prior to discussing information needed for design of a core retention device, it is important to realize that inclusion of such a device in a power plant would have an impact on the entire course of a hypothetical meltdown accident. For instance, a core

Major conclusions of the review are as follows:

1) The documents submitted for review together do identify most areas of uncertainty. The most important of these, and the additional areas of uncertainty identified by the reviewers, are felt to be:
   a) crust formation and upward heat flux from the melt,
   b) exfoliation of brick layers in the retention device,
   c) thermalhydraulics of the molten core materials,
   d) mechanism of melt attack on the refractory material, and
   e) influence of retention device geometry on local refractory erosion.
2) The design and design analysis of the core retention device submitted by Offshore Power Systems places an unjustifiable reliance on the low temperature experience of the steel industry. Other relevant industrial experience does not appear to have been considered.

retention device would greatly reduce the rate of aerosol generation by gas sparging of material from the melt. Substantial reductions of aerosol generation from non-fuel sources would occur. At the same time a variety of heat removal mechanisms available to the melt while in contact with concrete would not be available to a melt within the core retention device. Aerosol generation by vaporization of fission products from this hotter melt would increase. The net result would be a decrease in aerosol generation and a decrease in the rate of aerosol sedimentation within containment. The aerosols within containment would come primarily from fuel sources rather than non-fuel sources as in the case of melt/concrete interactions.

The reviewers did not attempt to identify collateral impacts on meltdown accidents caused by the inclusion of a core retention device. Such determinations can best be done in conjunction with with accident modeling such as that being performed at Battelle Memorial Institute. Attentions were directed, instead, toward identification of design information necessary to meet other goals of a core retention device, namely retardation of ex-vessel melt movement and gas generation.

A) General Comments from the Reviewers

1) The heavy reliance on steel industry experience is not warranted since the temperature ranges involved in steel manufacture (1350 - 1700°C) are on the low end of the melt temperature range expected during a light-water reactor core meltdown accident (1350 - 2600°C). Temperatures cited in the Offshore

Power System report surprised the reviewers since they seemed quite low. During the first day of a melt/concrete interaction melt temperatures fall rapidly from about 2600°C to about 2000°C. For the next five days melt temperatures smoothly decline over the range from 2000°C to 1700°C.

Many of the heat removal mechanisms available during melt/concrete interactions--such as convective heat transport by gas generation and endothermic decomposition reactions of concrete--are not available during melt interactions with core retention materials. The melt temperatures ought then be at least as great during melt/core retention material interactions as those encountered during melt/concrete interactions.

Melt temperature becomes important because at least a portion of the refractory erosion expected during melt/core retention material interaction is due to chemical reaction. Since the rates of chemical reactions are sensitive and non-linear functions of temperature, it is most hazardous to extrapolate encouraging experience at low temperatures to core meltdown situations.

2) Heat generation in the melt and environs was restricted to fission product decay heat. No consideration was given to heat produced by oxidation of metallic phases

of the core melt. Baukal et al.,* have shown that heat due to oxidation of zirconium and chromium in a core melt can be significant in comparison to fission product decay heat. Generation of this heat might be slower--and consequently more prolonged--during melt/core-retention-material interactions than during melt/concrete interactions since gas transport through the melt would be more limited. However, the oxidizing environment of a light-water reactor accident does assure that this chemical heat source will be available.

3) A significant source of industrial experience was neglected in the Offshore Power System (OPS) report--the glass-making industry. Though again the temperature regimes this industry employs are much lower than the core meltdown temperature regimes, the industry has had to deal extensively with oxide melt/refractory interactions.

4) The reviewers did not feel competent to address questions concerning mechanical damage to a retention device as a result of debris impacting the device. The reviewers felt that industrial experience cited in the OPS report was particularly pertinent and

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*W. Baukal, J. Nixdorf, R. Skoutajan, and H. Winter, "Investigation of the Relevancy and the Feasibility of Measurement of Chemical Reactions During Core Meltdown on the Integral Heat Content of Molten Cores," BMFT-RS-197, Battelle Institut., e.v., Frankfurt am Main, F.R. Germany, June 1977, English translation NUREG/TR-0047, October 1978.

realistic to these questions of mechanical damage. At some point more definitive data than the anecdotal accounts in the OPS report should be made available.

5) The reviewers felt that chemical interactions of oxidic melts with refractory bricks were not adequately treated. Little data are available on this point. Data that are known indicate that assumptions of uniform attack and neglect of melt convection may be seriously in error. The reviewers agreed with the OPS report that the nature of the chemical interactions was a major area of uncertainty.

6) Little data are available concerning melt behavior under conditions of interest. Crust formation over the surface of the melt or other phenomena that would impede upward heat flux from the melt were neglected in the OPS report. The reviewers do not share the OPS confidence that more complete understanding of the surface behavior of the melt could only lead to a greater margin of safety for the retention device. Any reduction in upward heat transfer rate from the melt translates into greater erosion rates of the retention device.

7) The OPS retention device has a minimum vertical thickness of 8 feet 3 inches. It has a minimum lateral thickness of 3 feet 8 inches. Since, to a first approximation, erosion rates in the vertical and lateral directions due to thermal attack are

considered equal, it appears that the sidewalls are the weak link in the design. Lateral penetration, rather than vertical penetration, would be the expected failure mode. Prediction of the failure time must then include the effects of both chemical and thermal erosion by the melt and the effects of material strengths at elevated temperatures.

Data provided by OPS indicate that the core retention device would fail laterally in no more than 60 hours even if only 50% of the decay heat were to go into refractory erosion.

B) Areas of Concern Not Considered by OPS

1) Reflooding of the melt by water was not considered by OPS. OPS did cite unspecified emergency actions should a meltdown accident occur. Deliberate reflooding of the melt could be among these. Reflooding has been indicated by accident analyses (P. Cybulskis, Battelle Memorial Institute, Columbus, Ohio) to be a hazardous undertaking. Reflooding, whether deliberate or accidental, has not been experimentally studied and appears to be an important area of uncertainty.

2) As a corollary to 1) OPS did not consider pressure generation produced when melts contact a water-saturated retention device. Since the MgO bricks described in the OPS report are about 17% porous, they could retain as much as 2030 ft^3 of water. The only design feature

to assure that water does not become entrained in the bricks is a 1/4" steel liner of uncertain description. Vaporization of entrained water could be a significant source of containment pressurization. The reviewers are aware of only a single, scoping, transient experiment in which a high temperature melt was streamed onto a water-saturated brick. (D. A. Powers, Meeting with Experts on the Technology of Sacrificial Materials for Delaying core Melt-Through, August 29-30, 1978, Bethesda, MD) This transient test indicated only relatively smooth vaporization of entrained water.

3) The OPS design of the retention device includes tongue-and-groove bonding of refractory bricks to prevent brick floatation. This design will function satisfactorily only if an entire course of bricks remains intact. Should localized attack penetrate a few bricks, the entire course might exfoliate and float to the top of the pool. This uncertainty adds special emphasis to the uncertainty of localized rather than uniform attack on the refractory.

4) Creep of stressed refractory at high temperatures was not addressed in the OPS study though they provided data for refractory creep at low temperatures (~ 1600°C). Creep of high purity MgO is significant at 1800°C and at lower temperatures for materials of lower purity. Creep rates are exponential functions

of temperature and sensitive to composition.* The core melt places surface bricks in the retention device under loads of about 4 psi. Bricks on the bulkhead walls above the retention device are under loads of about 18 psi. At 2000°C these loads are sufficient to produce significant deformation of refractory structures. Should the bricks be contaminated by solid state diffusion of melt materials into the bricks, even greater creep rates may develop.*

5) The OPS report neglects thermal-hydraulics of the melt. The analysis of MgO erosion is conducted by a thermal ablation model assuming uniform attack on the refractory. The reviewers could find no basis for this assumption. Quite the contrary, available data and industrial experience suggest that localized attack is a major mode of refractory erosion. Some photographs of refractories exposed to glass melts are shown in Figure 1.

The most important variables in determining the rate of localized attack appear to be geometry, melt composition, temperature, and fluid phase convection. Another uncertainty related to melt hydraulics is whether small perturbations in the refractory

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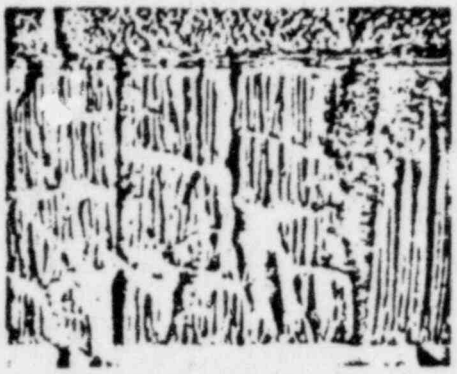
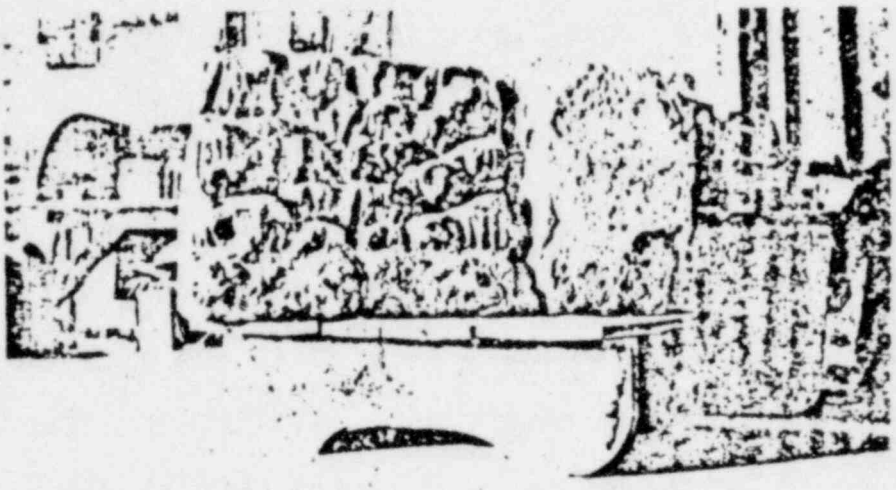
*E. Yasuda and S. Kimuia, "High Temperature Creep of Magnesia with Minor Additives," Proc. Oxydes Refractaires pour Filières Energetiques de Haute Température, Odeillo, France, June 28--July 1, 1977.



a) Note stepped erosion and enhanced erosion at joints between bricks.

c)

Upward-facing surfaces from which the clay-bearing glass was apparently swept as fast as formed, with the result that active dissolving of the surfaces took place.



b) An Example of Local Attack

These two blocks side by side in service, the face of the right hand block did not crack and showed practically no eating. The face of the left hand block shrunk and cracked badly in use; upward eating started in these defects and practically half the block was dissolved away; side wall blocks, bottle glass tank.

Figure 1a from F. C. Flint and A. R. Payne
J. Amer. Ceram. Soc. 9 613 (1926)

Figure 1b, c from D. W. Ross
J. Amer. Ceram. Soc. 9 641 (1926)

Figure 1. Examples of Chemical Attack on Refractories by Glass Melts

surface grow preferentially or are healed during further attack. Industrial experience suggests that both results are possible.

6) Volumetric expansion of the molten pool is not considered by OPS. Crude calculations by the reviewers indicate that the expansion is not likely to be significant if no additional material falls from the reactor pressure vessel or the bulkhead walls into the molten pool. The effective volume change associated with heating a 17% porous brick from 25°C to melting and assuming a volume change on melting of +5% (exact value is not known) is only +0.6%. However, if steel from the lower head of the pressure vessel is added to the melt, the melt volume would increase by at least 24%. Oxidation of metal phases in the pool or collapse of bulkhead walls would further expand the pool to the point that little safety margin would exist in the core retention device.

C) <u>Areas of Concern OPS Treated as Adequately Understood</u>

1) The model for MgO erosion used by OPS was a simple thermal energy balance using the classical steady state ablation formulation. Heat flux applied to the refractory surface was treated as simple fractions of the fission product decay heat (see I-A-1 above) and were independent of the thermalhydraulics of the melt. The model neglected any chemical component attack on the refractory. The model

consequently assumed that the refractory must be heated to a critical melt temperature before it was eroded. The reviewers could not ascertain the basis for the OPS confidence that this simplified model of refractory erosion was verified.

2) The tongue-and-groove construction used for assembling the core retention materials does appear adequate to prevent brick floatation provided:
   -- all bricks remain in place and exfoliation of the brick layers cannot occur (see I-B-3 above).
   -- the tongues do not shear due to thermal or mechanical shock.

3) Thermal shock of refractory bricks is most definitely an area of uncertainty. All tests to date involving prototypic melts deposited on MgO bricks have been of a transient nature. In every case the bricks suffered catastrophic fracture after the melt solidified. Because of the transient nature of the tests, it is not known whether the fracturing was due to surface cooling of the bricks or delayed heating of the brick interior.

4) Upward heat flux from the melt is an area of uncertainty. Crust formation, or spalled refractory floating on the melt surface will depress the melt surface temperatures and consequently the upward heat flux. Heatup of walls and reactor internals above the melt will also depress upward heat flux. Any reduction in

upward heat flux results in more heat being available for ablation of the core retention material.

5) MgO-basaltic concrete interactions were not addressed in the OPS report. These interactions may occur at the bulkhead wall coated with 4" MgO bricks. Once basaltic concrete under this coating reaches 1100°C it will begin to melt and the MgO coating will lose its structural integrity. (See also Section I-B-4.) Molten basaltic concrete will be free to flow into the core melt pool and to attack the core retention material.

D) Areas of Uncertainty Considered by OPS

The reviewers agreed with the listings of uncertainties presented in the OPS report. These areas need not be discussed further here. The reviewers felt, however, that there might be some misunderstanding of oxide chemical attack by liquid oxides--sometimes termed "slag-line" attack or "flux-line" attack. Chemical attack is mass transport dominated erosion of the refractory--as opposed to the heat transport dominated erosion considered in the OPS report. Dissolution as opposed to ablation of the refractory occurs because of favorable free-energy relationships among constituents of the melt/refractory system. The rate of dissolution is given by expressions of the form:

$$\text{rate} = k_o(C,T) \exp(-E(c)/RT)\ [C-C_S(T)]$$

where T = temperature

C = fluid phase composition

C_S = saturation composition of the fluid

K_o and E = kinetic parameters dependent on temperature and fluid composition

R = universal gas constant

Chemical attack is important because it can occur at low temperatures (even in the solid state) and it can be responsible for non-uniform attack.

The rate of attack is very sensitive to temperature. Because of the non-linear nature of the rate expression, it is difficult to extrapolate data from low temperature experiments to predict high temperature behavior. Further, the above rate expression refers only to the net dissolution of refractory. Erosion of solid refractory can occur even when the net rate is zero, provided precipitation of refractory-containing species from the fluid phase occurs. When these precipitated species are of low density--like MgO--and can be swept out of the system--say by floating to the top of an immiscible phase overlying the attacking fluid--this zero net rate dissolution is quite likely.

Iron oxides frequently arise in discussions of refractory attack since they form low melting species with most refractory oxides. However, chemical attack on refractories is not restricted to iron oxides.

Iron oxides are especially important in discussions of light water reactor accidents since these accidents involve high temperature molten steel in very oxidizing environments. Rates of steel oxidation in these conditions can be quite high unless the steel is covered by a reasonably thick, viscous slag layer. (See also Section I-A-2.)

Photographs of refractories subjected to chemical attack shown in Figure 1 illustrate the non-uniform nature of chemical attack. Certainly one of the uncertainties that must be addressed in core retention device design is whether vertical or horizontal surfaces are more grievously affected by chemical attack.

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

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7. AUTHOR(S)

5. DATE REPORT COMPLETED: MONTH February, YEAR 1980

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13. TYPE OF REPORT: Safety Evaluation Report (Supplement No. 3)

PERIOD COVERED (Inclusive dates)

15. SUPPLEMENTARY NOTES

14. (Leave blank)

16. ABSTRACT (200 words or less)

Supplement No. 3 to the Safety Evaluation Report for the Offshore Power System's application for a license to manufacture eight standardized floating nuclear plants in a shipyard-like facility in Jacksonville, Florida has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. This Supplement provides an evaluation of a refractory sacrifical bed, called a core ladle, proposed by Offshore Power Systems to be included in the design of the Floating Nuclear Plant to delay the melt-through penetration of molten core debris in the unlikely event of a core meltdown accident. This design feature is in response to a specific requirement in the Floating Nuclear Plant Final Environmental Statement, Part III, NUREG-0502, that the concrete pad beneath the reactor vessel be replaced with a pad constructed of magnesium oxide or other equivalent refractory material, that will provide increased resistance to melt-through by a molten reactor core, that will not react with core debris to form a large volume of gases and that will not have any deleterious effects on safety.

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