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

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

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STAFF REVIEW AND EVALUATION

OF

OFFSHORE POWER SYSTEMS

RESPONSE TO ACRS LETTER OF JULY 25, 1979

NOVEMBER 1979

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INTRODUCTION

On June 27, 1979 the ACRS Subcommittee responsible for the review 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 20th 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)

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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.

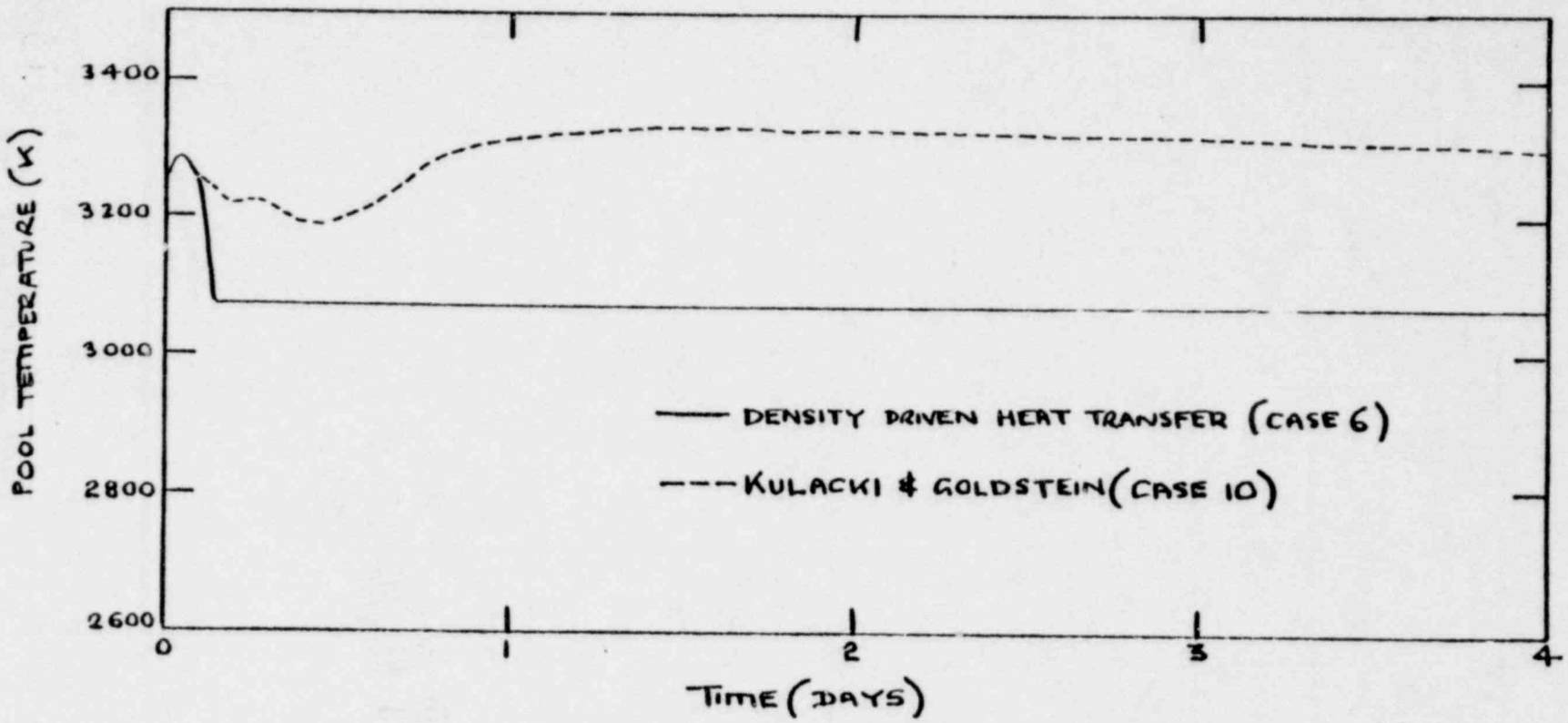
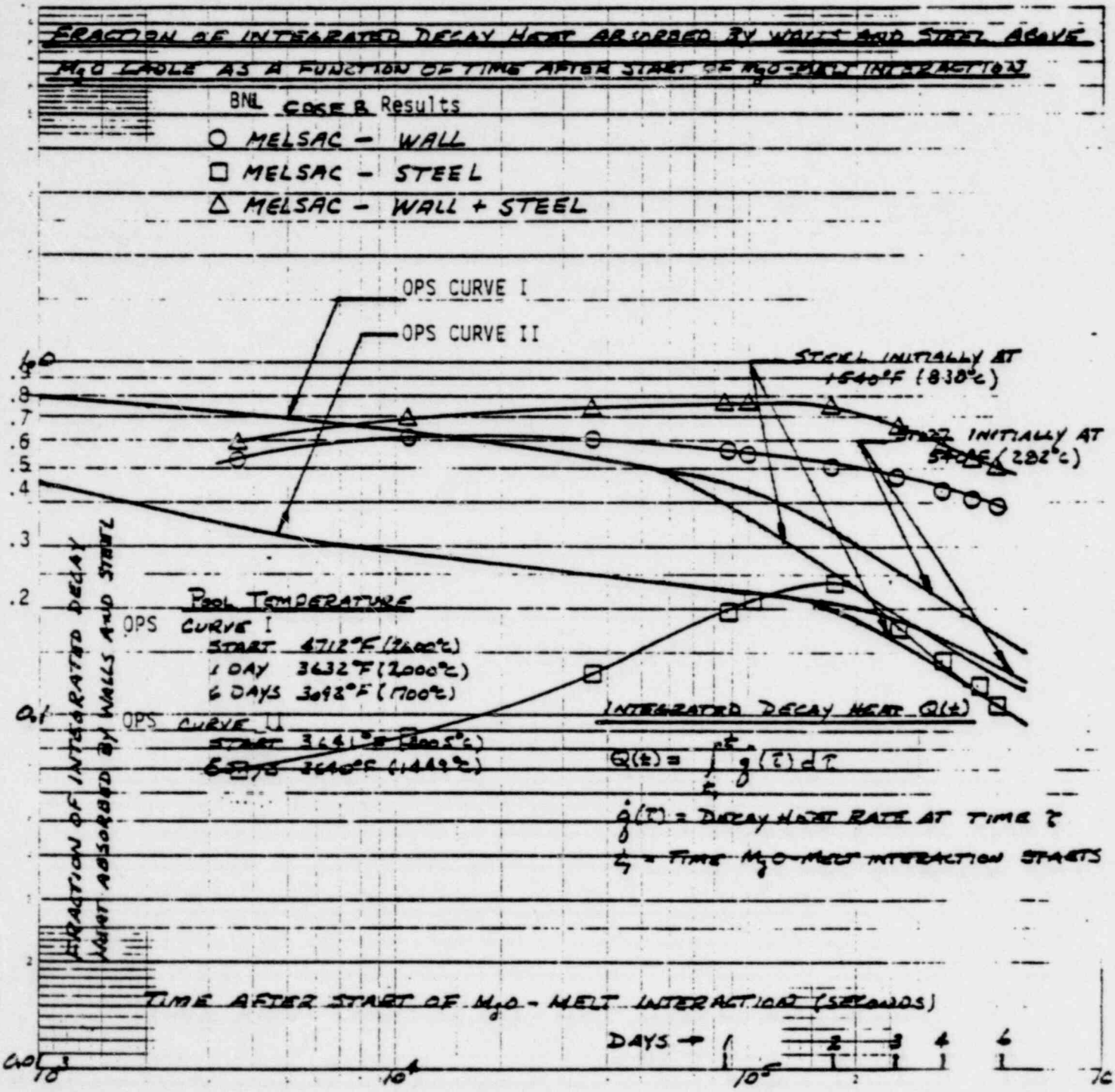


FIGURE 1 POOL TEMPERATURE HISTORIES PREDICTED BY MELSAC

FIGURE 2 *



* REPRODUCED FROM REFERENCE (5)

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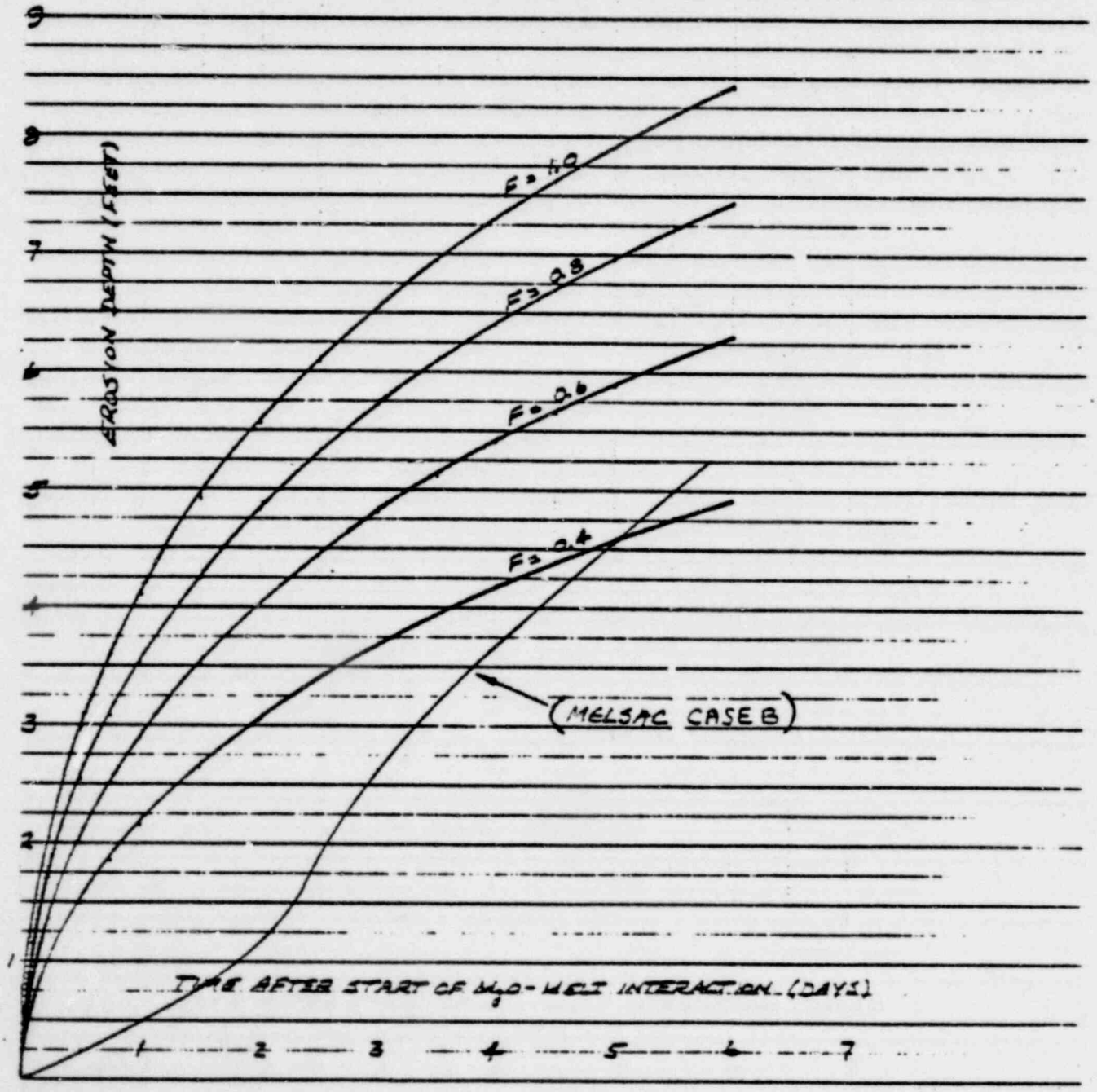
FIGURE 3*

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

NOTE:

1. BASE MAT AREA = 117 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.

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(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

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 Appendi 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

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

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 potential 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 staff 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.

APPENDIX A

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

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(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 CPS 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.

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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

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FNP-PAL-050

APPENDIX B

Offshore Power Systems

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Box 8000, Jacksonville, Florida 32211

904 724-7700
Telex 552406

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

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

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

Very truly yours,

P. B. Haga

P. B. Haga

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Attachments

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

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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

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.