DATE ISSUED:
January 23, 1980

ACRS-1694

MINUTES OF THE ACRS SUBCOMMITTEE
ON THE FLOATING NUCLEAR PLANT
LOS ANGELES, CA
NOVEMBER 17, 1979

The ACRS Subcommittee on the Floating Nuclear Plant (FNP) met with representatives of the NRC Staff and Offshore Power Systems (OPS) in Los Angeles, California on November 17, 1979, to continue its review of the Offshore Power Systems application for a manufacturing license for the Floating Nuclear Plant. The specific topics of discussion at the meeting were the review of the proposed magnesium oxide core ladle design and the implications of the Three Mile Island, Unit 2 accident on the FNP design. A notice of the meeting appeared in the Federal Register on November 2, 1979 (Attachment A). A copy of the detailed presentation schedule is attached (Attachment B). A list of attendees at the Subcommittee Meeting is attached (Attachment C). A list of documents provided to the Subcommittee for this meeting is attached (Attachment D). There were no public statements either written or oral. The entire meeting was open to members of the public.

MEETING WITH THE NRC STAFF AND OFFSHORE POWER SYSTEMS (OPEN SESSION)

1.0 Subcommittee Chairman's Opening Remarks

Dr. Moeller, Subcommittee Chairman, introduced the members of the Subcommittee and noted the purpose of the meeting. He pointed out that the meeting was being conducted in accordance with the provisions of the Federal Advisory Committee Act and the Government in the Sunshine Act and that Mr. Gary Quittschreiber was the Designated Federal Employee for the meeting. He stated that no requests for oral statements nor written statements from members of the public had been received with regard to this meeting.

2.0 Introductory Remarks

Mr. Etherington discussed some history of the ACRS core ladle review noting that the Subcommittee previously asked a number of questions concerning a previous design. The presently proposed design is radically different and recognizes the Subcommittee's previous concerns. Modifications include the following:

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- o Modifications have been made to protect the superstructure from radiant heat
- o Capacity of ladle has been increased to accommodate the pressure vessel and other debris
- o The side walls have been thickened to provide stability and additional protection. The ladle has essentially become a large enclosed furnace.

Mr. Etherington noted the basic problem remaining with regard to the ladle was to determine how much heat is absorbed in the ladle and how much is radiated upwards, this will take considerable time to determine.

Mr. Birkel, NRC Staff, summarized the significant milestones achieved during the FNP review. Birkel noted that the NRC Staff wished to conclude its review of the ladle promptly and asked that the ACRS review the concept and preliminary design of the ladle at its December 1979 meeting so that they could consider the ACRS comments as part of their review effort.

Mr. Baer, NRC Staff, noted that Mr. Denton has indicated that some sites, such as Indian Point and Zion, may have longer evacuation times than acceptable and may require additional mitigating features. One of the concepts being considered is a core ladle. Baer noted that for that reason Denton wished to proceed with the FNP ladle review and to get the Committee's views on the matter.

Mr. Haga, Offshore Power Systems, reminded the Subcommittee that the hearing boards have completed all outstanding contentions. All that is needed by the board to finish its review for the manufacturing license is the following:

- o Latest plant design amendments
- o Final ACRS letter
- o Final Environmental Statement supplement.

Mr. Haga noted that they are including the ladle only because it is required by the NRC. The design is based on best estimate calculations. He noted there is a lot of flexibility to change the configuration in the space available for the ladle. He added that they have evaluated the TMI accident and the Kemeny Report and concluded that they can readily incorporate all of the identified changes during the detailed design of the FNP.

Mr. Clifford Haupt, NRC Staff, discussed the siting of the FNP on riverine and estuarine sites. He also discussed the NRC Staff's reasoning for requiring the FNP core ladle as an environmental consideration and not a public health and safety issue.

Dr. Okrent discussed the need to look into the development of liquid pathway acceptance criteria for accidents. He questioned how one could determine the acceptability of the consequences of an accident for an FNP located on a river or in an estuary if there is no liquid pathway acceptance criteria for accidents on which to make the judgment. Dr. Moeller expressed an interest in having criteria established for determining the acceptability of the performance of interdiction measures.

3.0 Technical Presentations

3.1 FNP Core Ladle Description, Configuration, and Structural Evaluation

Mr. Clint Dotson, OPS, discussed the design requirements and configuration of the core ladle (Attachments 1-6). The proposed design is adequate to contain all the fuel assemblies plus 90% of the reactor vessel steel attachments and internals. The configuration has been developed within the existing constraints of the steel structure. The bend radius of the in-core instrumentation tubing was modified to accommodate the new design. Any concrete that could be exposed to thermal radiation from the molten pool will be protected by high temperature insulating bricks. Mr. Etherington discussed a concern of his that the concrete would disintegrate due to dehydration long before it would begin to melt. OPS representatives noted that the concrete outside the ladel area is needed for shielding and insulation to the structural steel. It does not serve any structural support purposes. Mr. Etherington asked that OPS/NRC discuss what happens to the concrete and the consequences at such time as this matter is brought before the ACRS. Dotson noted that all the structural steel necessary to maintain the integrity of the cavity will be shielded such that the temperature at its surface will not exceed 1000^{o}F for the two day duration of the core debris retention period.

Mr. Dotson noted that they have assumed gross failure of the bottom reactor vessel head around the periphery with a total weight of approximately 200 tons impacting the ladle as a crushable missile. The ladle support structure would remain within its elastic resistance during such an event.

3.2 Ladle Design Thermal Calculations

Mr. Robert Bruce, OPS, discussed the ladle design thermal calculations (Attachments 7-12). OPS has assumed that 20% of the total heat of the molten mass is radiated to the vessel. OPS has assumed the worst case conditions with an initial pool temperature of 2600°C falling to about 2000°C after one day. The decay heat rate follows the ANS curve less 20%.

OPS does not have a coupled heat transfer model which takes heat sinks into account. Mr. Etherington indicated that the calculations were unrealistic. Mr. Stumpf noted that a large fraction of the early heat being generated would be deposited into the walls. After a day or so the heat would be reflected back into the cavity or diffuse through the walls to the concrete such that the ratio of the fraction of heat deposited to the fraction of heat generated would drop. Mr. Bruce indicated that until they have a coupled calculation they can not say whether their calculations are conservative or not. Mr. Walker noted that the temperature would be limited to 2800°C since iron boils away at that temperature.

Dr. Shewmon stated that he considered the assumption that no water would come into contact with the molten mass as extraordinarily conservative. Mr. Bruce added that they used the most conservative estimate which is that no water would come into contact with the molten mass.

OPS has concluded that the delay time for melt-through of the proposed ladle design is at least two days while maintaining the basaltic concrete wall temperature below 2200°F.

In response to a question from Dr. Catton concerning the Germans' feelings that lateral heat transfer may be higher than vertical, Bruce said there is no knowledge that this would be the case for magnesium oxide, but if it were found to be the case, they could change the dimensions of the ladle in the detailed design. Dr. Dana Powers, Sandia Laboratories, stated that the Germans' comment concerning double lateral heat flux has frequently been misquoted and the

Germans are withdrawing their statement on this more and more frequently.

Dr. Walker noted that Fe and UO_2/UO_3 vaporize at 2600° to 2800°C. The pool will be limited to about 2800°C due to the vaporization. Walker noted that zirconium is about the only material that would be in the condensed phase at 3000°C.

3.3 Effect of Hearth Material on Airborne Release

Dr. Walker discussed several containment failure modes relative to the effect base mat material has on the airborne release pathway to the environment (Attachments 13-15). Dr. Okrent indicated that the OPS submission in this area did not make him feel confident that OPS had sufficiently examined this area. He was concerned that adding the ladle to reduce the liquid pathway environmental effects may increase the airborne release effects on public health and safety. He also noted that the recommendations in the final report of the TMI-2 Lessons Learned Task Force will likely include modifications of the containment which will effect airborne releases in case of a core melt. Mr. Etherington asked that the effects of water getting into the ladle be addressed from the standpoint of a steam explosion.

Mr. Marchese, NRC Staff, felt that OPS had understated the advantages of the magnesium oxide core ladle. Marchese stated that the NRC Staff has not found any disadvantages of the ladle. He felt that it would significantly mitigate the temperature, pressure, and hydrogen transients in the containment. He also indicated it would significantly mitigate the amount of activity sparged from the debris into the atmosphere of the upper containment.

3.4 NRC Staff's Evaluation of the FNP Core Ladle

Mr. Marchese noted that the NRC Staff and consultant groups have been performing core melt evaluations on the FNP, FFTF, and CRBR for at least five years (Attachments 23 & 24).

Dr. Pratt, Brookhaven National Laboratory, discussed the heat transfer feedback effect between the molten pool and the ladle and structures and the radiation feedback from the walls back to the pool (Attachments 16-22). Pratt discussed the MELSAC code which incorporated the feedback effect of the molten pool heat transfer of the heating up of structures around the pool. The code assumes the pool is initially pure UO_2, and as the ladle melts it assumes the UO_2 is diluted with MgO. Pratt felt that the assumption that the pool is initially pure UO_2 instead of UO_2/steel mixture was conservative and simple. The code allows for the formation of a crust at the upper surface which strongly influences the pool heat transfer. The difference between the upward pool heat transfer and that lost from the surfaces dictates whether the pool is growing. The use of the code has allowed them to perform relatively simple scoping studies. They found that the ladle penetration time is relatively insensitive to the pool heat transfer correlations. The code has allowed them to add certain feedback features to determine the effects of additional considerations. For example, adding the vessel steel into the pool increases the ladle penetration time by 23%. Modelling natural convection from the back surface of the ladle increases penetration time by 10%. These studies showed that, typically, ladle penetration was greater than five days and reactor vessel melting was something less than two days. Pratt indicated that additional work on sideward penetration may be important since the code assumes a density driven heat transfer correlation which has no sideward penetration. Stratified layers of steel in the pool would increase the sideward penetration substantially. Pratt expected considerable damage to the upper structures in the cavity between the two and six day hold-up times and felt thicker cavity walls may be needed to protect the structure beyond two days.

Dr. Swanson, Aerospace Corporation, discussed the stability of the six inch magnesite wall with regard to loss of bricks. He discussed specific instances in the steel industry where molten metal comes into contact with cold MgO bricks with no detrimental results. He noted that even though two-to-three inch cracks occur the MgO bricks still remain in place and give protection.

Swanson noted that scrap chunks of metal as large as ten tons are routinely dropped directly onto the refractory liners in modern day furnaces with little or no shock damage. Also, flat bottoms in blast furnaces with diameters as large as 18 ft, have had no problem with flotation of MgO bricks by molten metal. Low carbon bricks, which have a much lower density than the MgO bricks proposed for use on the FNP, have successfully been used without flotation by using a key lock design such as that proposed by OPS.

Mr. Marchese summarized stating that the NRC feels the FNP core ladle concept is feasible in terms of providing significant delay times. The ladle can be engineered to provide retention of the molten debris for at least two days to one week assuming the worst possible core melt scenario. Marchese indicated that they have only been looking at the dry core melt which was the most conservative but that they intended to look at the wet core melt in the future. Haga indicated that it was likely that no water would be in the cavity since the sumps are isolated from the cavity. The only way water could get to the cavity is if it was pumped into the cavity through a hole in the vessel. Mr. Etherington suggested that a position should be taken on whether water gets into the cavity and the position should be discussed and evaluated.

Mr. Marchese stated that the structural criteria that OPS has outlined, to support the position that the ladle will not fail prior to core melt debris melting through the ladle, are acceptable to the NRC. He added that they would ensure these criteria are met during the final design.

Dr. Okrent said that he was interested in the containment as a whole system and whether having the ladle leads to a potential for loss of containment integrity possibly in the upward direction. Marchese stated that the fact that the interaction of refractory brick with molten core debris does not generate gases, does not generate water vapor, does not generate hydrogen; had to improve the situation.

3.5 Auxiliary Feedwater System Reliability

Mr. Bruce noted that the FNP design has four motor driven auxiliary feedwater pumps taking suction from two auxiliary feedwater storage tanks and injecting directly into the four steam generators. Also, one turbine driven pump injects directly into the four steam generators. Any two motor driven pumps will maintain the plant in safe shutdown and after five hours any one pump will be sufficient.

OPS has calculated a system unreliability of about 10^{-5} to 10^{-4}. In the case of the loss of offsite power the failure rate of the turbine driven pump is 10^{-2} per demand.

3.6 Total Loss of AC Power

Mr. Bruce said that for the first six or seven hours following total loss of AC power the core would be cooled adequately by the turbine driven auxiliary feedwater pump. DC power to the vital instruments would also be available. After about six hours to about 20 hours the plant will remain in a safe undamaged condition with a continuous manually regulated supply of auxiliary feedwater. After about 20 hours the heat sink is likely to be lost due to steam generator dryout. Reactivity considerations and loss of reactor coolant are not limiting conditions for core damage during the first 17 hours. The FNP design has four diesels. Any two diesels are sufficient to maintain the plant in a safe shutdown condition. After five hours, any one diesel will maintain the plant in a safe shutdown. Bruce indicated that the loss of cooling water to the reactor coolant pump seals would result in an initial seal flow of five gallons per pump, amounting to a small LOCA at 20 gpm. Mr. Baer, NRC Staff, commented that pump seal leakage would likely be small as long as the pumps were not restarted.

Mr. Bruce noted that the huge heat sink in the ice condenser would maintain the containment temperature below the 280°F qualification temperature of the temperature-sensitive-instruments in the containment for about 60 hours.

3.7 Post Accident Hydrogen Buildup

Mr. Bruce noted the following with regard to hydrogen buildup in the FNP containment:

- o In a TMI type event the calculated containment pressure would be in the range of 35 to 40 psig. Estimated containment failure pressure is 49 psig (Attachment 25)
- o In a core melt accident with 100% Zr-H_2O reaction and with no hydrogen burn the calculated pressure is 40 psig.
- o In a core melt accident with hydrogen burn the pressure would reach about 200 psig which would exceed the estimated 49 psig failure pressure.
- o The electrical penetrations have a 60 psig design pressure and are not limiting in the FNP design.
- o Inerting the FNP ice condenser is feasible but totally impractical from the operators standpoint due to the frequent maintenance and inspection required. Present technical specifications of ice condensers require weekly walk-throughs. If things were properly

planned, monthly maintenance would still be required.

In response to questions from the Subcommittee it appeared that containment failure pressure could be increased from the existing 49 psi to about 65 psi by increasing the containment wall thickness from 5/8" to 7/8". Designing to a failure pressure above 65 psi would require some radical changes.

3.8 Vented Containment Concept

Dr. Walker discussed an FNP vented containment design concept which uses four 30-inch pipes that pass from the interior of the containment through the annulus space and down through the bottom of the platform (Attachment 26). The system is designed to pass 100,000 cu ft of gas per minute. The system would accommodate burning of the hydrogen generated by the zirc-water reaction in a period as short as five to ten minutes without exceeding 49 psi. The FNP design has flexibility and space available to accommodate this system if NRC rulemaking shows that it is required.

Westinghouse tests have indicated an iodine decontamination factor of greater than 500 is realistic. Due to the 150 foot flow path from the vents to the side of the platform it may be much better. Also, there is significant fission product dissolution capability in sea water at depths of 20-40 ft. which would provide absorption of the noble gases before they escape to the surface.

Dr. Walker noted that the advantage of diverting iodine and noble gases to the sea water instead of the atmosphere is that it reduces the transport time by factors of 10 to 100 and allows for decay before being received by any dose receptors.

3.9 NRC Evaluation of TMI Related Items

Mr. Baer stated the NRC Staff has done nothing specifically on the FNP with regard to post TMI type requirements. Baer noted that they would have to address the TMI issues before they could issue a final SER and before they can recommend issuance of a manufacturing license.

Mr. Baer noted that the Lessons Learned Task Force recommended that the Commission issue a notice of intent to conduct rule making to solicit comments relating to the consideration of design features to mitigate accidents resulting in some core melt (not substantial melt) and severe core damage.

4.0 Concluding Remarks

Mr. Birkel made the following remarks:

- o OPS has met the NRC Staff's requirements of FES-III providing a core melt delay time of greater than two days
- o The proposed core ladle design is feasible
- o The proposed core ladle design meets 10 CFR 50 requirements for a preliminary design which in this case is more than generally provided for a construction permit
- o The NRC Staff will require OPS to submit a final design of the ladle before manufacture of any major FNP hull structure or component.

Mr. Haga told the Subcommittee that it was essential that they get a manufacturing license in order to continue to exist. He noted that they have made a proposal to a utility which is strongly interested in purchasing two units. In addition, they are preparing proposals to other utilities. Haga asked that the Subcommittee review the FNP project in December 1979 with the objective of preparing a final letter on the FNP for a manufacturing license.

Mr. Etherington, Dr. Shewmon, and Dr. Catton suggested that the ladle design was developed well enough for final ACRS review.

Dr. Okrent suggested that the ladle be reviewed as part of a systems containment basis and not by itself. Okrent also suggested that the ACRS should talk to the Commissioners to find out what they plan with regard to action on the FNP as a result of "the pause." He did not feel the ACRS could write a letter telling the NRC Staff to incorporate whatever it decides with regard to changes resulting from TMI.

Mr. Mathis questioned the advisability of the concept of venting the containment to the sea.

Mr. Etherington suggested the NRC Staff take a position with regard to the possibility of water getting into the ladle.

Dr. Moeller summarized the Subcommittee member's comments. He suggested that time be set aside at the December 1979 ACRS meeting to discuss how the ACRS should proceed with the ladle design.

The meeting was adjourned at 5:00 pm.

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For additional details, a complete transcript of the meeting is available in the Nuclear Regulatory Commission Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555, or from Ace-Federal Reporters, Inc., 444 North Capital Street, N. W. Washington, D. C.

$2.84 an hour for the first 160 hours and $2.74 an hour for the remaining 160 hours. (Tobacco)

Clamourette Fashion Mills, Inc., Quebradillas, PR; 8-17-79 to 8-16-80; 50 learners for normal labor turnover purposes in the occupations of: (1) knitting, for a learning period of 480 hours at the rate of $2.50 an hour for the first 240 hours and $2.67 an hour for the remaining 240; (2) machine stitchers, for a learning period of 320 hours at the rate of $2.50 an hour for the first 160 hours and $2.67 an hour for the remaining 160 hours; (3) pressers, for a learning period of 320 hours at the rate of $2.50 an hour for the first 160 hours and $2.67 an hour for the remaining 160 hours; and (4) kettle handlers and dyers for a learning period of 240 hours at the rate of $2.50 an hour. (Sweaters and related products)

Each learner certificate has been issued upon the representations of the employer which, among other things were that employment of learners at special minimum rates is necessary in order to prevent curtailment of opportunities for employment, and that experienced workers for the learner occupations are not available.

The certificate may be annulled or withdrawn as indicated therein, in the manner provided in 29 CFR, Part 528. Any person aggrieved by the issuance of any of these certificates may seek a review or reconsideration thereof on or before November 19, 1979.

Signed at Washington, D.C. this 25th day of October 1979.

Arthur H. Korn,

Authorized Representative of the Administrator.

[FR Doc. 79-34021 Filed 11-1-79; 8:45 am]

BILLING CODE 4510-27-M

NUCLEAR REGULATORY COMMISSION

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Advisory Committee on Reactor Safeguards, Subcommittee on the Floating Nuclear Plant; Meeting

The ACRS Subcommittee on the Floating Nuclear Plant will hold a meeting on November 17, 1979, at the Los Angeles Marriott Hotel, 5855 West Century Boulevard, Los Angeles, CA 90045 to review the application of the Offshore Power Systems, et al. for a manufacturing license for the Floating Nuclear Plant. Notice of this meeting was published October 18, 1979 (44 FR 60178).

In accordance with the procedures outlined in the Federal Register on October 1, 1979 (44 FR 56408), oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The agenda for subject meeting shall be as follows: *Saturday, November 17, 1979, 8:30 a.m. until the conclusion of business.*

The Subcommittee may meet in Executive Session, with any of its consultants who may be present, to explore and exchange their preliminary opinions regarding matters which should be considered during the meeting and to formulate a report and recommendations to the full Committee.

At the conclusion of the Executive Session, the Subcommittee will hear presentations by and hold discussions with representatives of the NRC Staff, Offshore Power Systems, et al, and their consultants, pertinent to this review. Specific topics to be discussed include the proposed design of the core ladle and implications of the Three Mile Island, Unit-2 Accident on the Floating Nuclear Plant design.

In addition, it may be necessary for the Subcommittee to hold one or more closed sessions for the purpose of exploring matters involving proprietary information. I have determined, in accordance with Subsection 10(d) of Public Law 92-463, that, should such sessions be required, it is necessary to close these sessions to protect proprietary information (5 U.S.C. 552b(c)(4)).

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefore can be obtained by a prepaid telephone call to the Designated Federal Employee for this meeting, Mr. Gary R. Quittschrieber, (telephone 202/634-3267) between 8:15 a.m. and 5:00 p.m., EST.

Background information concerning items to be considered at this meeting can be found in documents on file and available for public inspection at the NRC Public Document Room, 1717 H Street, N. W., Washington, DC 20555 and at the Jacksonville Public Library, 122 North Ocean St., Jacksonville, FL 32204, the Business and Science Division, New Orleans Public Library, 219 Loyola Ave., New Orleans, LA 70140, and the Stockton State College Library, Pomona, NJ 08240 and (regarding TMI-2 Accident Implications) at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Street, Harrisburg, PA 17126.

Dated: October 29, 1979.

John C. Hoyle,

Advisory Committee, Management Officer.

[FR Doc. 79-33823 Filed 11-1-79; 8:45 am]

BILLING CODE 7590-01-M

SECURITIES AND EXCHANGE COMMISSION

[Rel. No. 10916; 811-2693]

Bowen Investment Co.; Application Pursuant to Section 8(f) of the Act for an Order Declaring That Company Has Ceased To Be an Investment Company

October 26, 1979.

Notice is hereby given that on September 18, 1979, Bowen Investment Company ("Bowen") (formerly called Automatic Service Company), 2175 Parklake Drive, N.E., Atlanta, Georgia 30345, filed an application for an order pursuant to Section 8(f) of the Investment Company Act of 1940 ("Act") declaring that it has ceased to be an investment company as defined in the Act. All interested persons are referred to the application on file with the Commission for a statement of the representations contained therein which are summarized below.

Bowen was incorporated under the laws of North Carolina and registered under the Act on October 31, 1976, as a diversified, closed-end management investment company.

Pursuant to approval by vote of shareholders on June 6, 1979, Bowen on June 8, 1979, transferred substantially all of its assets to Fidelity Municipal Bond Fund, Inc. ("Fidelity"), in exchange solely for the number of shares of Fidelity stock having an aggregate net asset value equal to the value of Bowen's net assets transferred to Fidelity. Immediately thereafter, Bowen commenced liquidation, distributing the Fidelity stock pro-rata to its shareholders of record entitled thereto by means of the establishment of open accounts on the stock records of Fidelity in the names of such stockholders representing the respective pro-rata number of shares of fidelity stock due such shareholders. At the time of the application, all but 136 of Bowen's shareholders had tendered their shares of Bowen stock and received their respective amounts of Fidelity stock in return. Pursuant to the laws of the State of North Carolina, Bowen will convert all unclaimed shares of Fidelity stock to cash and deposit same with the appropriate state officials to be held

Attachment A

PRESENTATION SCHEDULE
Floating Nuclear Plant Subcommittee Meeting
November 17, 1979
Los Angeles, CA

	PRESENTATION TIME	APPROXIMATE TIME
MEETING WITH OFFSHORE POWER SYSTEMS AND THE NRC STAFF (OPEN SESSION)		
1.0 SUBCOMMITTEE CHAIRMAN'S OPENING REMARKS		8:30 am
2.0 INTRODUCTORY REMARKS		
2.1 NRC Staff	5 min	8:35 am
2.2 Offshore Power Systems	5 min	8:45 am
3.0 TECHNICAL PRESENTATIONS		
3.1 Staff Positions - NRC Staff Presentation w/OPS Response o Consideration Given to Use Vented Containment o Riverine and Estuarine Evaluations o Safety Versus Environmental Issues	15 min	8:55 am
3.2 Core Ladle Design - OPS Presentation o Description of Configuration and Structural Evaluations o Ladle Design Thermal Calculations o High Temperature Materials Interactions o Impact of MgO Ladle on Core Melt Airborne Releases	1 hr. 10 min	9:25 am
Break for Lunch		12:00 noon - 1:00 pm
3.3 Evaluation of Core Ladle - NRC Staff Presentation	30 min	1:00 pm
3.4 Space and Layout Considerations Relative to Post-Accident Flexibility - OPS Presentation w/NRC Staff Response	10 min	2:00 pm
3.5 TMI Related Systems Considerations - OPS Presentation o Total AC Power Loss Evaluation o AFW system Reliability o Post Accident Hydrogen Buildup o Vented Containment Concept	30 min	2:20 pm

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3.6	Evaluation of TMI Related System Considerations - NRC Staff Presentation	30 min	3:45 pm
4.0	CAUCUS		4:45 pm

- Conclusions/Remarks
- Discuss Future Meeting Schedule

ADJOURNMENT 5:00 pm

Note: (1) A maximum of 30 minutes will be allowed for receiving oral statements from members of the public if requested.

(2) The speakers should limit their prepared presentations to the time allowed. An allowance, amounting to 100% or more of presentation time, has been made for questioning by the Subcommittee.

ATTENDEES LIST

ACRS Members

D. Moeller, Chairman
D. Okrent
P. Shewmon
W. Mathis
H. Etherington

ACRS Consultants

I. Catton

ACRS Staff

G. Quittschreiber, Designated Federal Employee

OPS

D. C. Aabye
R. A. Bruce
R. A. Thomas
R. J. Conney
N. Seaborne
R. S. Orr
H. J. Stumpf
Dee Walker
P. B. Haga
Clinton Dotson

Westinghouse

T. M. Daugherty

Aerospace Corp

D. G. Swanson

BNL

T. Pratt

Sandia Labs

D. A. Powers

NRC

W. C. Milstead, DSS
R. Codell, NRR
J. Read, DSE
A. Marchese, NRR
R. Baer, NRR
R. Birkel, NRR

ATTACHMENT C

DOCUMENTS PROVIDED TO THE
SUBCOMMITTEE
FOR THIS MEETING

1. Topical Report No. 36A59, dated April 1979, and Revisions 1 and 2, Offshore Power Systems FNP Core Ladle Design and Safety Evaluation.

2. Letter, P. Haga, Offshore Power systems to R. Baer, NRC, dated September 14, 1979, forwarding OPS response to ACRS questions.

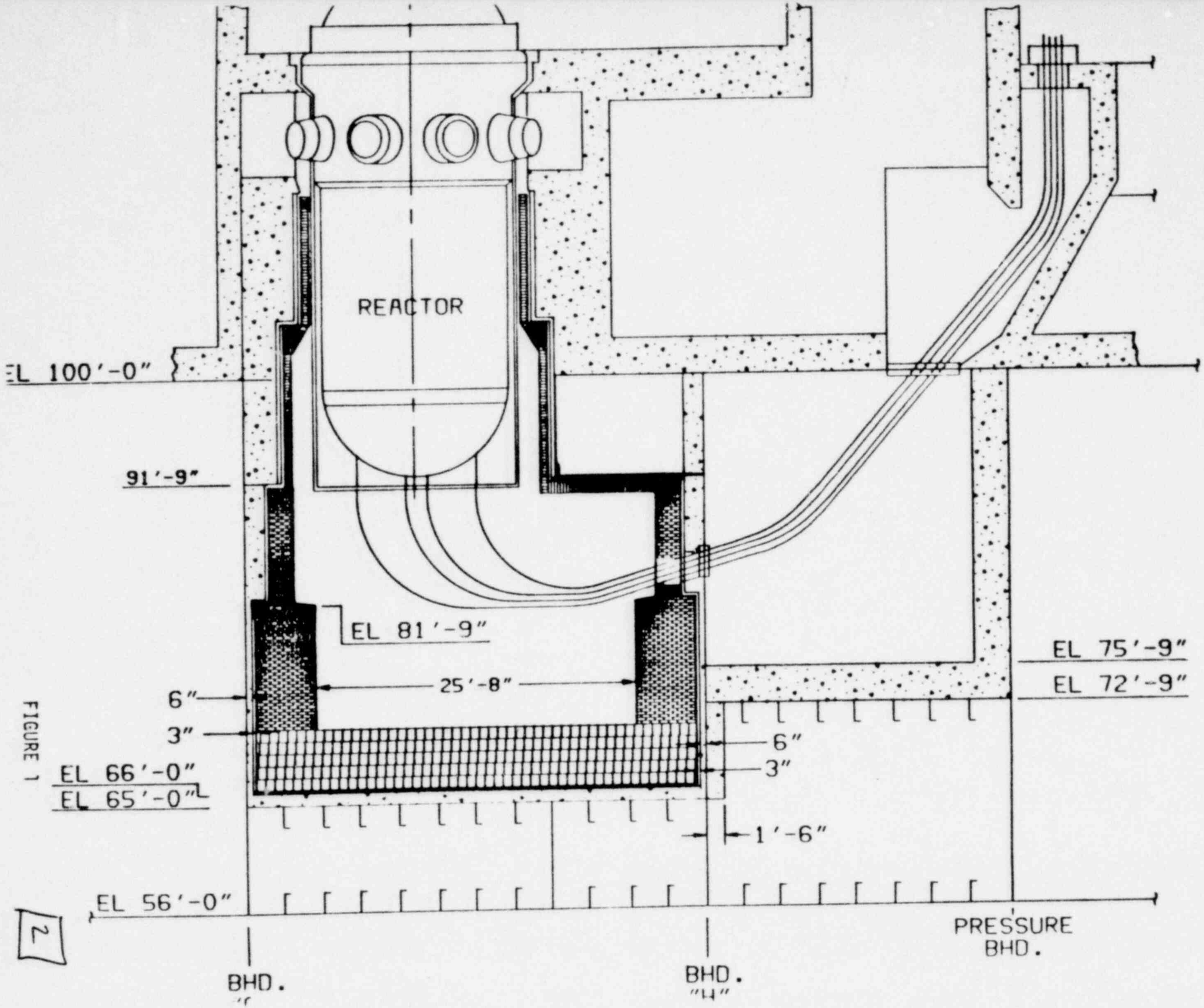
3. Staff Review and Evaluation of Offshore Power Systems response to ACRS letter of July 25, 1979, dated November 1979.

4. View-graphs shown at the meeting are provided as attachments 1 thru 26. A complete set of all handouts are provided in the meeting transcript and in the ACRS Office file for this meeting.

ATTACHMENT D

DESIGN REQUIREMENTS

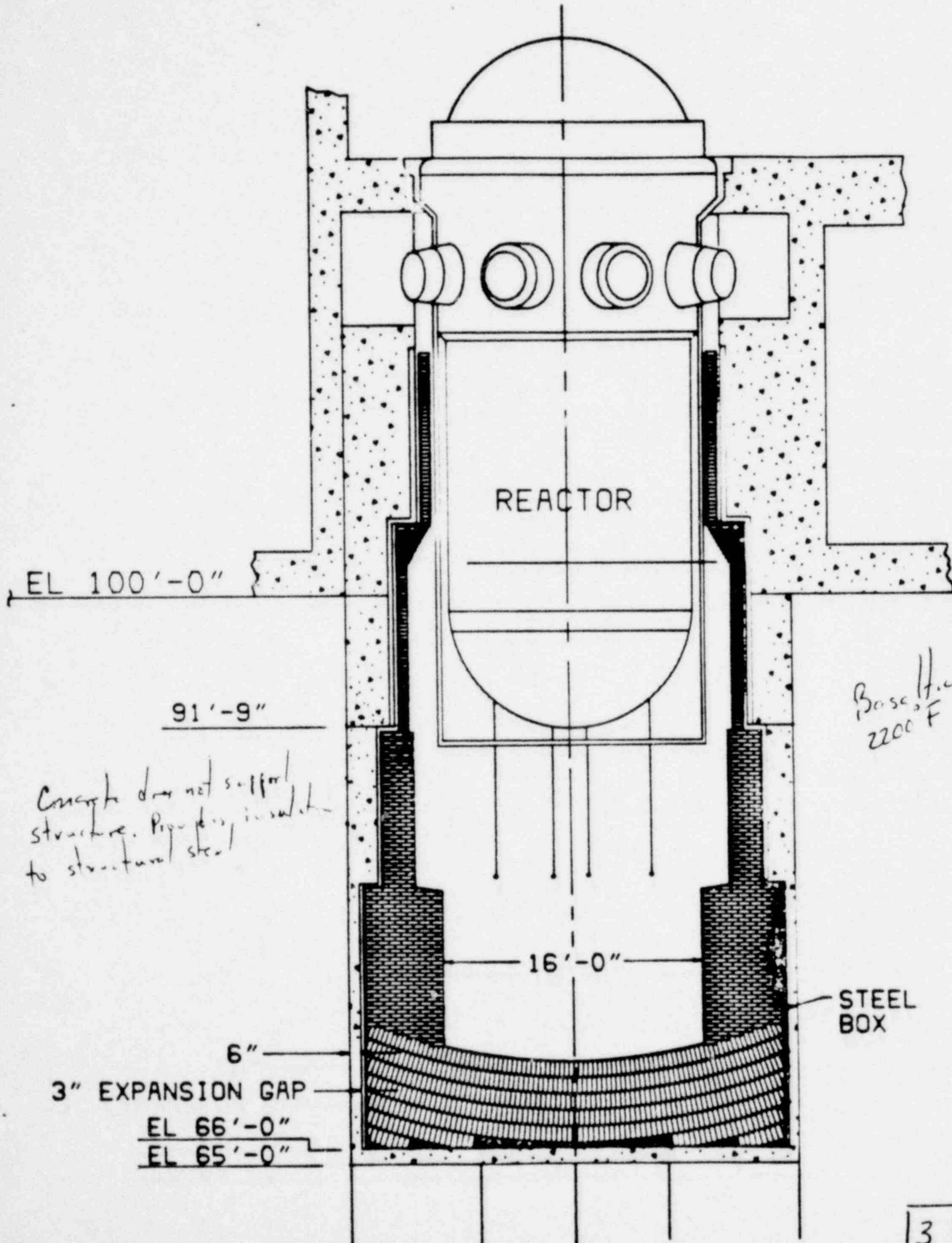
1. INCORPORATE A LADLE INTO THE EXISTING FNP DESIGN WITH MINIMUM ALTERATIONS.
2. ALTERATIONS TO THE REACTOR CAVITY SHALL NOT COMPROMISE SAFETY REQUIREMENTS INCLUDING:
   A. STRUCTURAL INTEGRITY OF THE PLATFORM SHALL BE MAINTAINED FOR ALL OPERATING AND DESIGN BASIS CONDITIONS PRIOR TO A POSTULATED CORE MELT ACCIDENT.
   B. WATER-TIGHT REDUNDANCY SHALL BE MAINTAINED BETWEEN THE BASIN AND REACTOR CAVITY.
   C. RADIATION SHIEDING REQUIREMENTS SHALL BE MAINTAINED.
3. THE PLATFORM STRUCTURE SHALL WITHSTAND LOADING CONDITIONS FOR THE DURATION OF CORE-MELT DEBRIS RETENTION.
4. THE REACTOR CAVITY STRUCTURE SHALL NOT BECOME THE WEAKEST LINK OF THE CONTAINMENT PRESSURE BOUNDARY AS A RESULT OF THE ADDITION OF THE LADLE.
5. THE LADLE CONFIGURATION AND MATERIAL SHALL NOT COMPROMISE OTHER SAFETY REQUIREMENTS.
6. THE LADLE SHALL BE AS THICK AS PRACTICABLE WITHIN THE VARIOUS DESIGN CONSTRAINTS BUT SHALL NOT BE LESS THAN FOUR FEET IN ANY DIRECTION.
7. THE LADLE POOL VOLUME SHALL BE SUFFICIENT TO CONTAIN THE MOLTEN CORE DEBRIS DURING CONTINUOUS BASIS MOTIONS ($1/2^{\circ}$).
8. THE LADLE SHALL BE DESIGNED AND ANALYZED TO REMAIN FUNCTIONAL FOR OPERATING BASIS ENVIRONMENTAL CONDITIONS. FOR MORE SEVERE CONDITIONS, THE PLANT CAN BE SHUT DOWN FOR INSPECTION OF THE LADLE.



REACTOR
EL 100'-0"
91'-9"
EL 81'-9"
EL 75'-9"
EL 72'-9"
25'-8"
6"
3"
6"
3"
EL 66'-0"
EL 65'-0"
1'-6"
EL 56'-0"
PRESSURE BHD.
BHD.
BHD. "H"

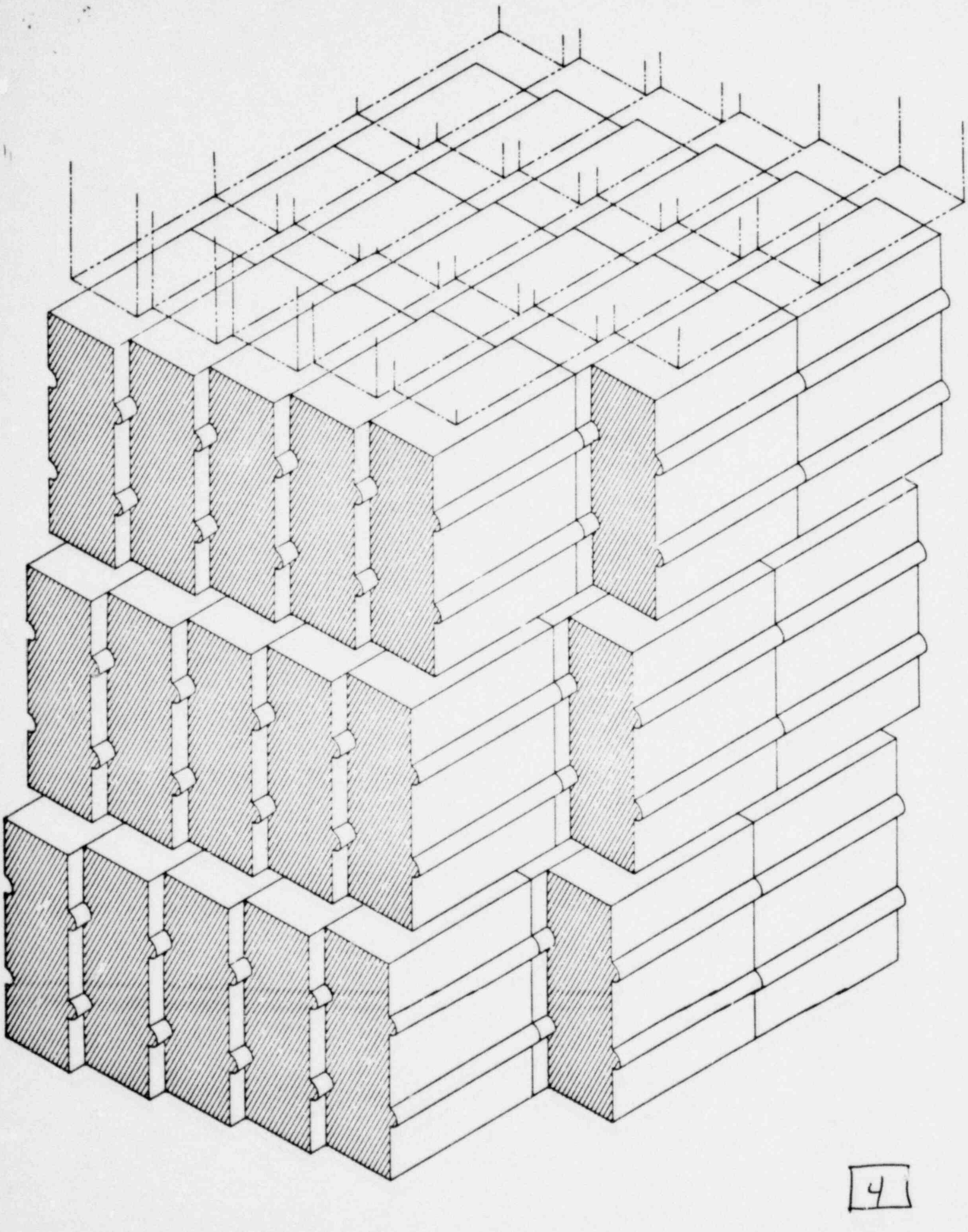
FIGURE 1

2


REACTOR
EL 100'-0"
91'-9"
Concrete does not support structure. Piping insulation to structural steel
Basaltic Concrete 2200 F melt
16'-0"
STEEL BOX
6"
3" EXPANSION GAP
EL 66'-0"
EL 65'-0"

3

FIGURE 2



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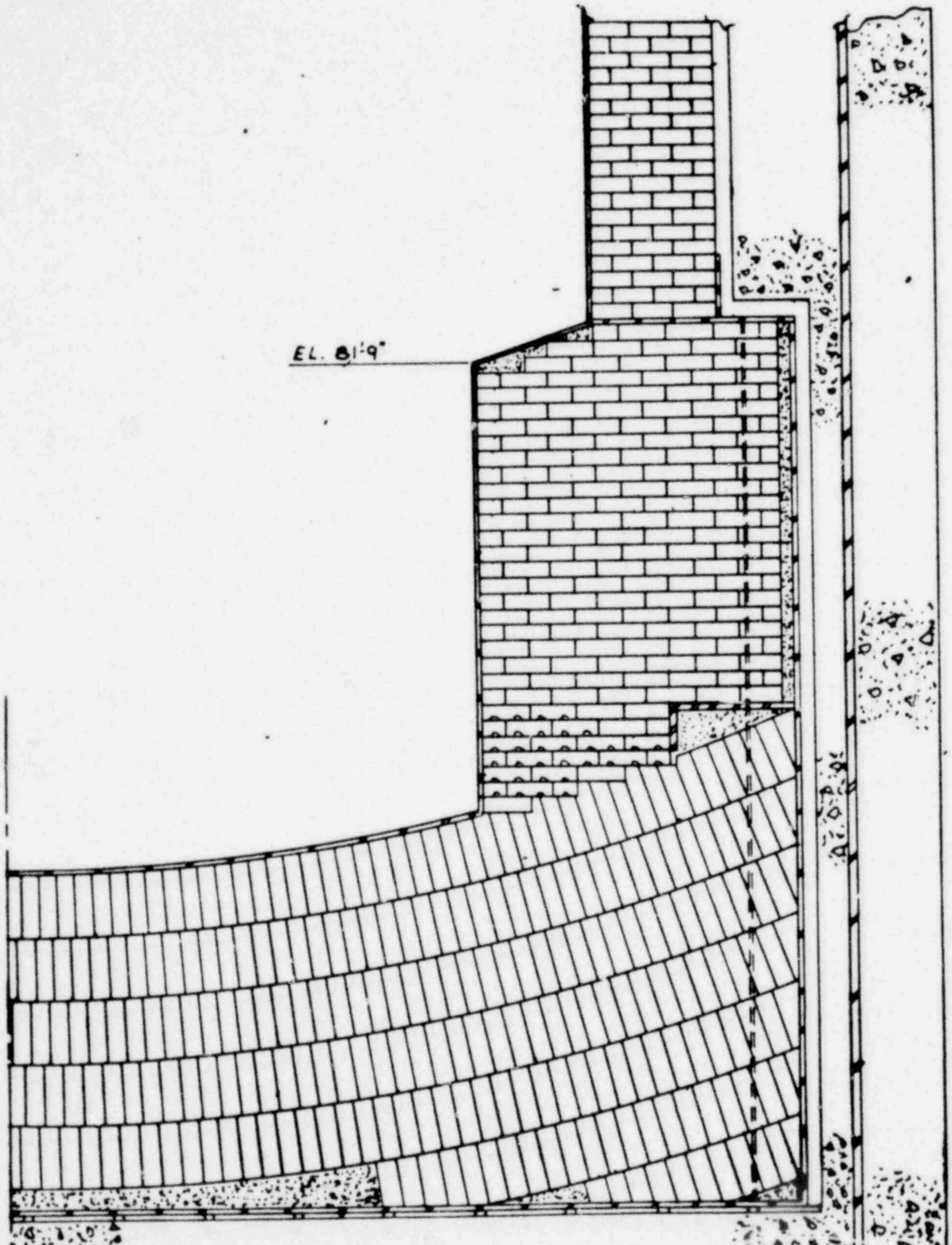
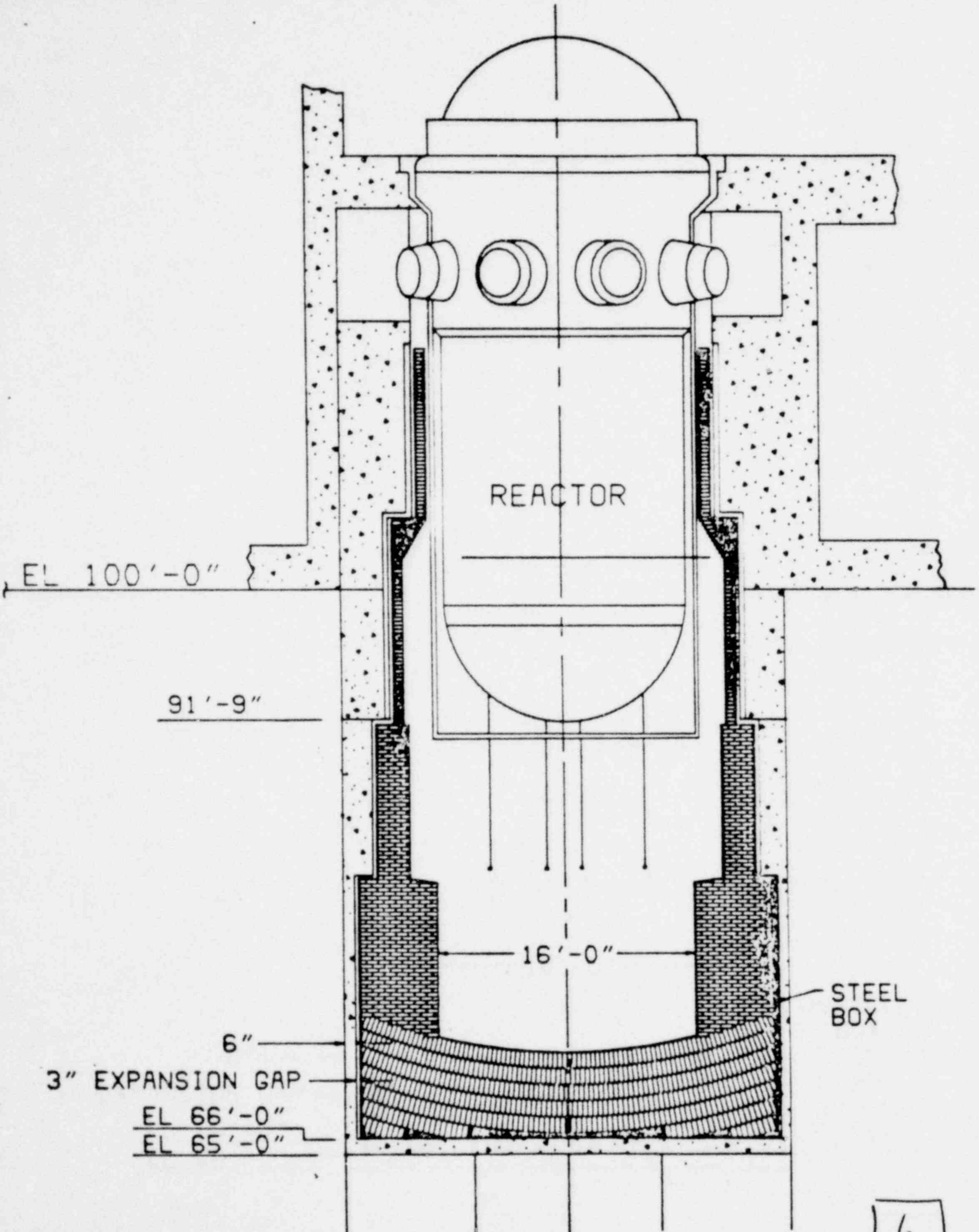

EL. 81'-9"


REACTOR
EL 100'-0"
91'-9"
16'-0"
STEEL BOX
6"
3" EXPANSION GAP
EL 66'-0"
EL 65'-0"

6

FIGURE 2

Bruce

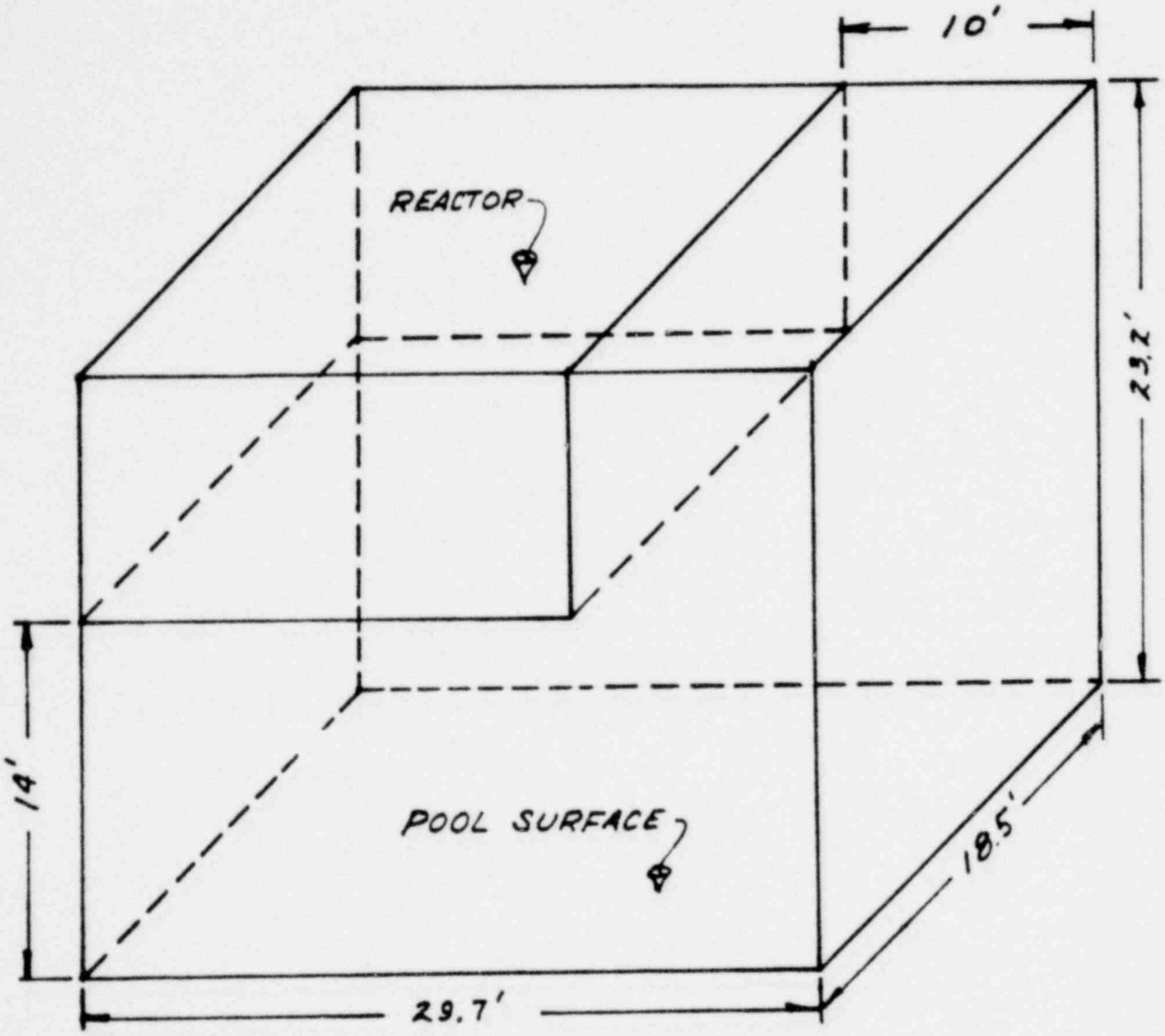
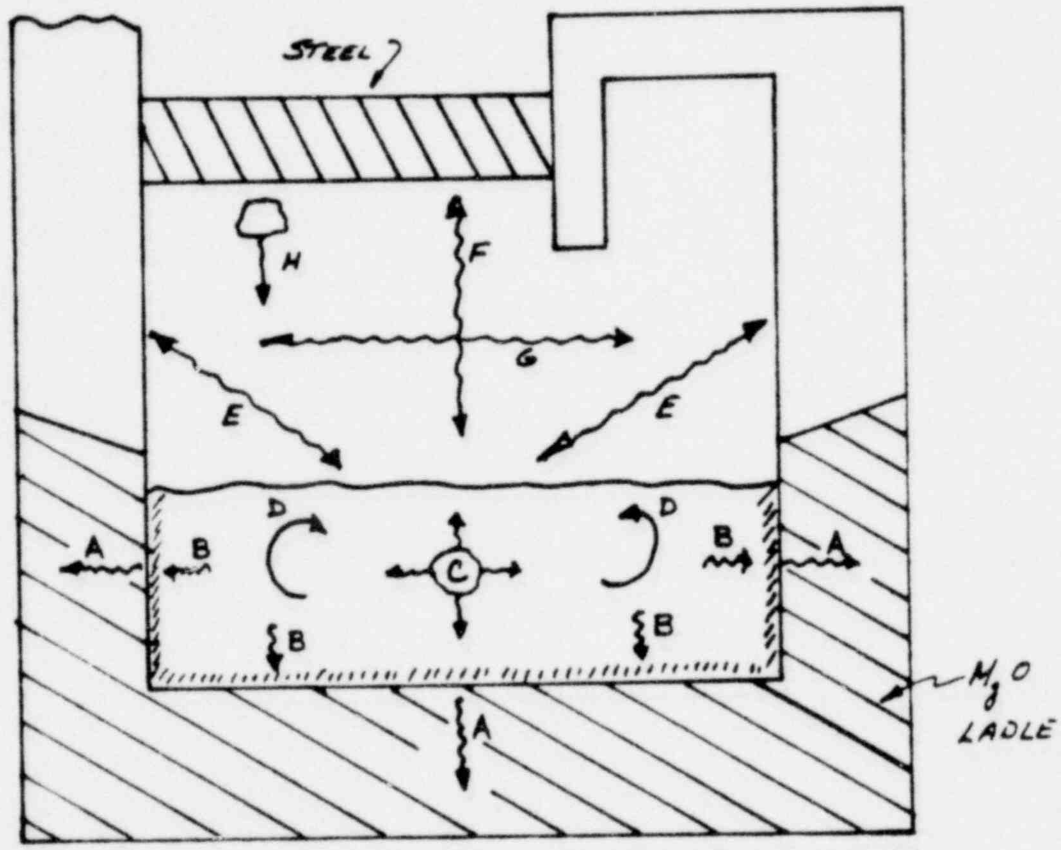


FIGURE 3

GEOMETRY OF WALLS ABOVE THE POOL SURFACE

7



A : CONDUCTION INTO MgO

B : EROSION OF MgO

C : DECAY HEAT

D : CONVECTION CURRENTS

E : RADIATION BETWEEN POOL AND WALLS

F : RADIATION BETWEEN POOL AND STEEL

G : RADIATION BETWEEN WALLS OR BETWEEN WALLS AND STEEL

H : MOLTEN STEEL FALLING INTO POOL

HEAT AND MASS TRANSFER PROCESSES AMONG MOLTEN POOL CORE LADLE AND SURROUNDING STRUCTURES

8

FIGURE 4

TABLE 1.

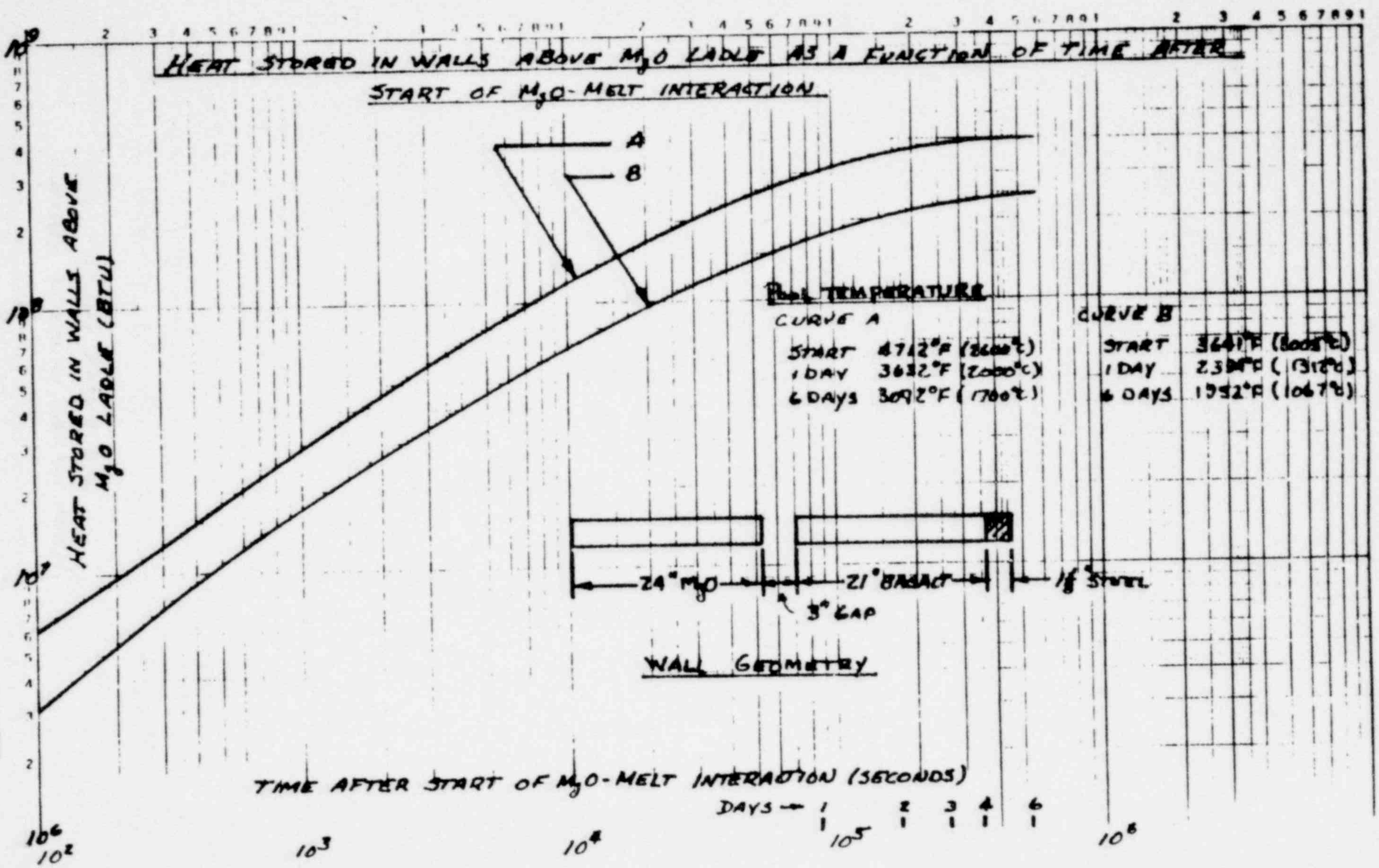
POOL SURFACE TEMPERATURE HISTORIES

1.

TIME (DAYS)	TEMPERATURE (°F), (°C)
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)

2.

TIME (DAYS)	TEMPERATURE (°F), (°C)
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)



POOR ORIGINAL

10

FIGURE 6

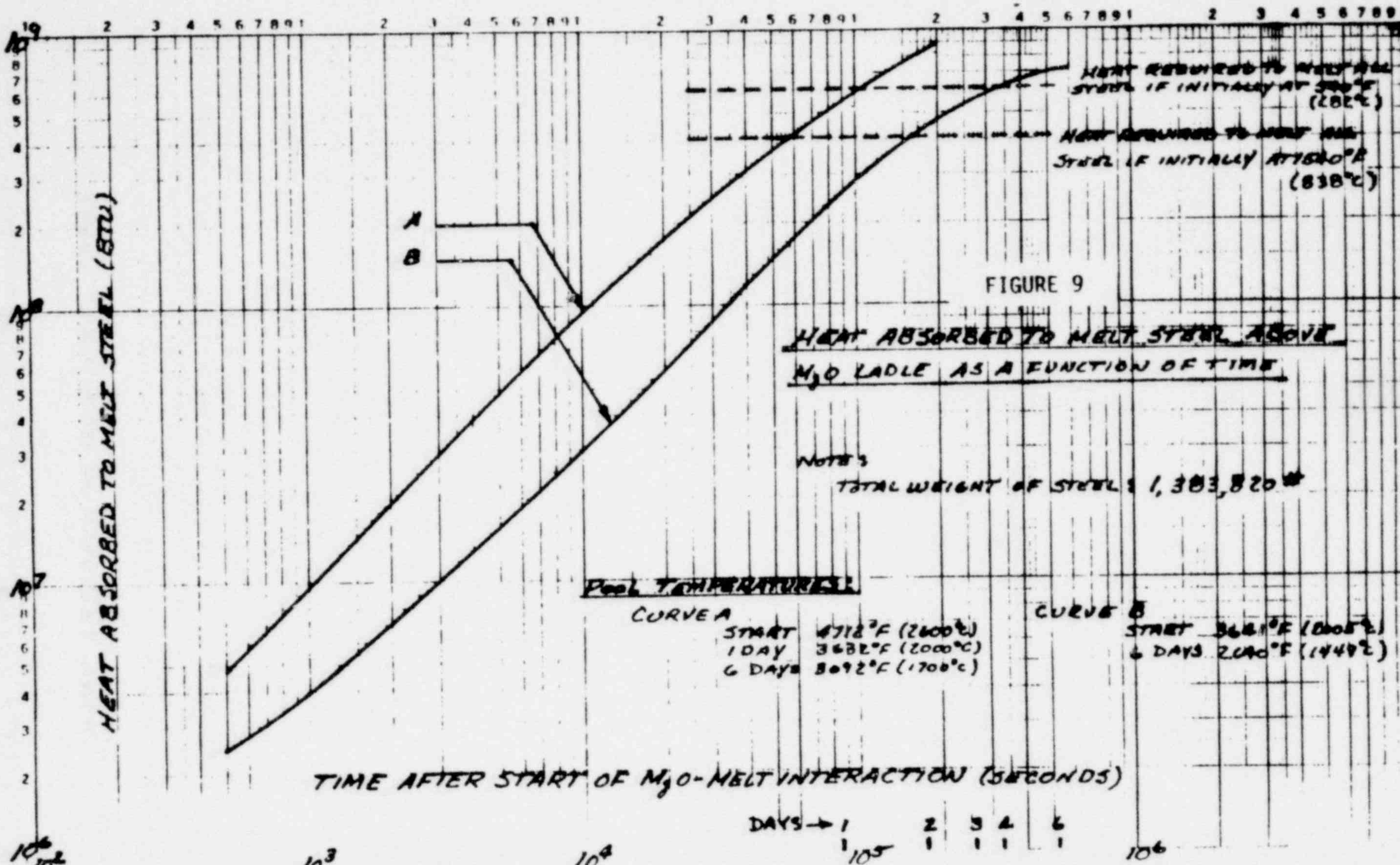


FIGURE 9
HEAT ABSORBED TO MELT STEEL ABOVE
MgO LADLE AS A FUNCTION OF TIME
NOTES
TOTAL WEIGHT OF STEEL 1,383,820 #
POOL TEMPERATURES:
CURVE A
START 4712°F (2600°C)
1 DAY 3632°F (2000°C)
6 DAYS 3092°F (1700°C)
CURVE B
6 DAYS 2640°F (1449°C)
HEAT ABSORBED TO MELT STEEL (BTU)
TIME AFTER START OF MgO-MELT INTERACTION (SECONDS)
DAYS→ 1
2
3
4
6
A
B
(838°C)
10³
10⁴
10⁵
10⁶
10⁷

FIGURE 9

K·E 10 X 10 TO THE CENTIMETER
KEUFFEL & ESSER CO.

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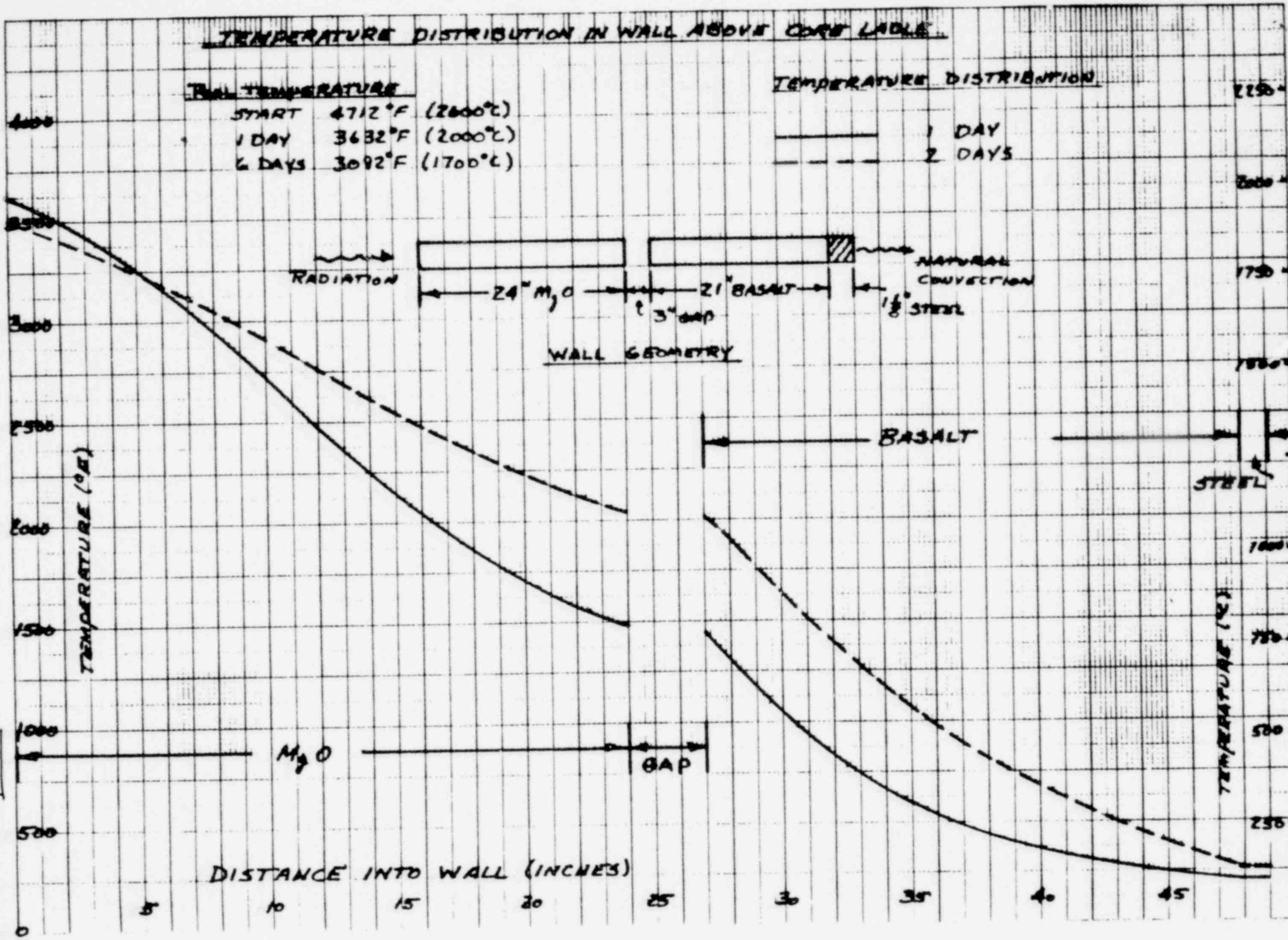
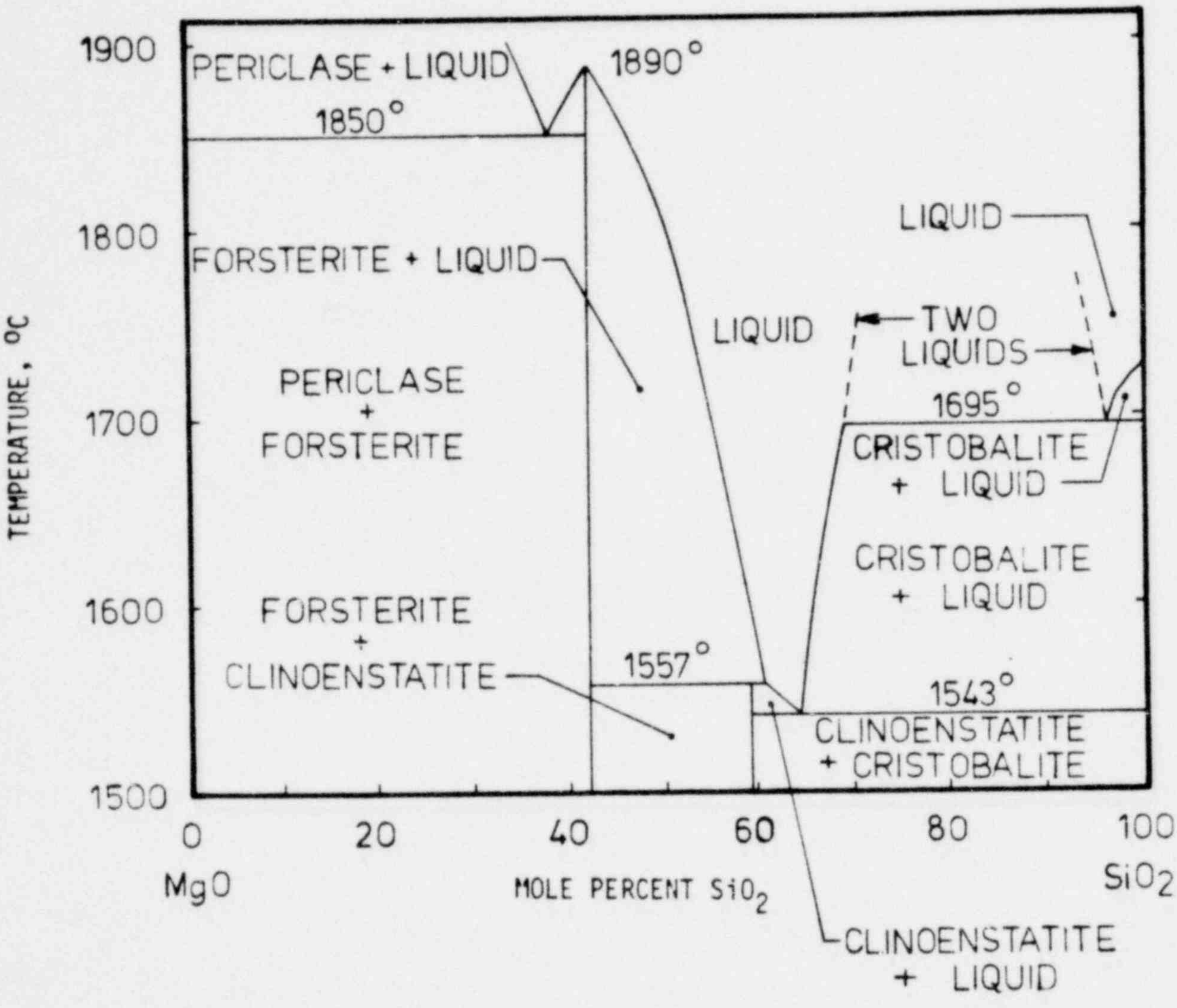


12

FIGURE 14

Walner



Ref: E. M. Levin, C. R. Robbins, H. F. McMurie, "Phase Diagrams for Ceramists," The American Ceramic Society, Columbus, Ohio, 1964

Figure 16
$MgO-SiO_2$ Phase Diagram

DISTRIBUTION AND DOSE CONTRIBUTION FOR PRINCIPAL RADIO NUCLIDES

	NOBLE GASES	IODINE	CESIUM	STRONTIUM
POST ACCIDENT DISTRIBUTION				
CONTAINMENT ATMOSPHERE	100%	*	SMALL, ~1%	SMALL, <1%
SUMP SOLUTION	-	*	90%	~10%
CORE DEBRIS	-	5%	~10%	~90%
AIR PATHWAYS DOSE CONTRIBUTION	LARGE	GREATEST	MINOR	MINOR
LIQUID PATHWAY DOSE CONTRIBUTION	NONE	MINOR	GREATEST	LARGE

* 95% IS DISTRIBUTED BETWEEN THE CONTAINMENT ATMOSPHERE AND THE SUMP WATER. THE DISTRIBUTION DEPENDS ON ACCIDENT SCENARIO.

F

COMPARISON OF AIR PATHWAYS RELEASES WITH MAGNESIA CORE LADLE AND CONCRETE

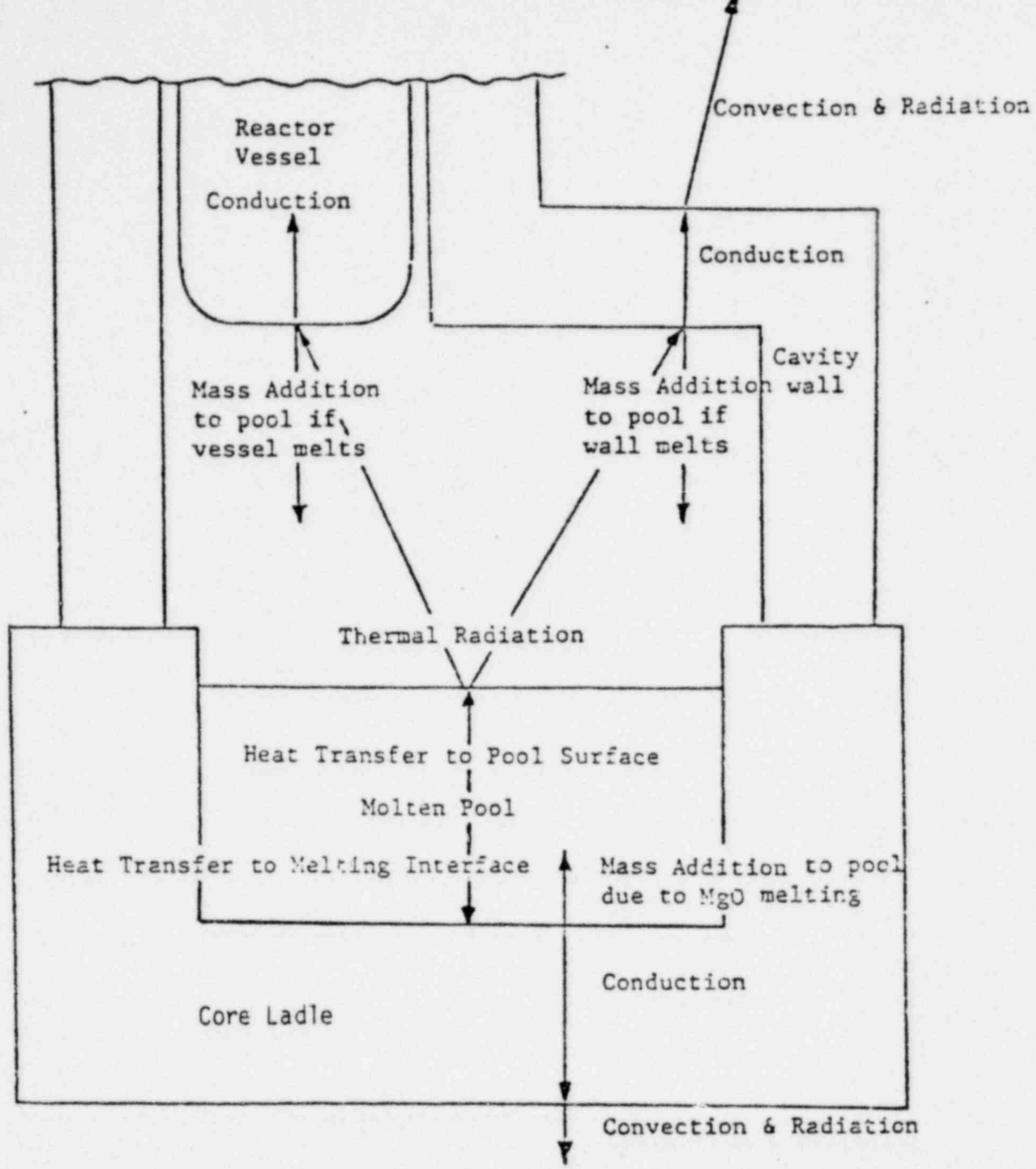
BASE MAT FOR CORE MELT ACCIDENT

FAILURE MODE	COMPARATIVE EFFECT ON RELEASE TO CONTAINMENT ATMOSPHERE			COMPARATIVE EFFECT ON FAILURE TIME OR CONTAINMENT FAILURE MODE	COMPARATIVE EFFECT ON CONSEQUENCES
	NOBLE GASES	IODINE	Cs,Sr		
STEAM EXPLOSION	NONE	NONE	NONE	NONE	NONE
CONTAINMENT MELT-THROUGH	NONE	MAY BE SLIGHTLY LARGER W/ CONCRETE	MAY BE SLIGHTLY LARGER W/ CONCRETE	MgO WILL PROLONG FAILURE TIME AND REDUCE QUANTITY RELEASE	SMALL
OVERPRESSURE FAILURE, NON-CONDENSIBLES	NONE	MAY BE SLIGHTLY LARGER W/ CONCRETE	MAY BE SLIGHTLY LARGER W/ CONCRETE	NON-CONDENSIBLES FROM CONCRETE-DEBRIS INTERACTION ENHANCES FAILURE PROBABILITY VIA THIS MODE	SMALL
FAILURE TO ISOLATE	NONE	NONE	NONE	NONE	NONE
OVERPRESSURE FAILURE, HYDROGEN BURNING	NONE	MAY BE SLIGHTLY LARGER W/ CONCRETE	MAY BE SLIGHTLY LARGER W/ CONCRETE	CONCRETE-DEBRIS INTERACTION CAN PROVIDE H_2 IGNITION	SMALL

15

PREVIOUS CALCULATIONS OF THERMAL PERFORMANCE

- FES PART III APPENDIX E:
  - SCOPING CALCULATIONS TO DETERMINE HOLD-UP TIME.
- OPS TOPICAL REPORT NO. 36 A 59:
  - THREE SEPARATE SCOPING CALCULATIONS:
    - HOLD-UP CALCULATIONS SIMILAR TO FES.
    - GAS GENERATION DUE TO THERMAL FRONT.
    - EFFECTS OF THERMAL RADIATION ON CAVITY WALLS.
- CONCLUSIONS:
    - AT LEAST 2 DAY HOLD-UP PROVIDED BY SUITABLE THICKNESS OF CORE LADLE.
    - CONCRETE AROUND LADLE OUTGASSED BY THERMAL FRONT BEFORE ARRIVAL OF MELT FRONT.
    - CONCRETE AND STEEL IN CAVITY WALLS PROTECTED FOR 2 DAYS BY SUITABLE THICKNESS OF REFRACTORY.
    - SEPARATE SCOPING CALCULATIONS MAY BE OVERLY CONSERVATIVE.
    - NEED FOR INTEGRATED CALCULATIONS TO DETERMINE BEST ESTIMATE.
    - EXISTING MELT FRONT CODES NOT SUITABLE, HENCE DEVELOP MELSAC CODE AT BNL.

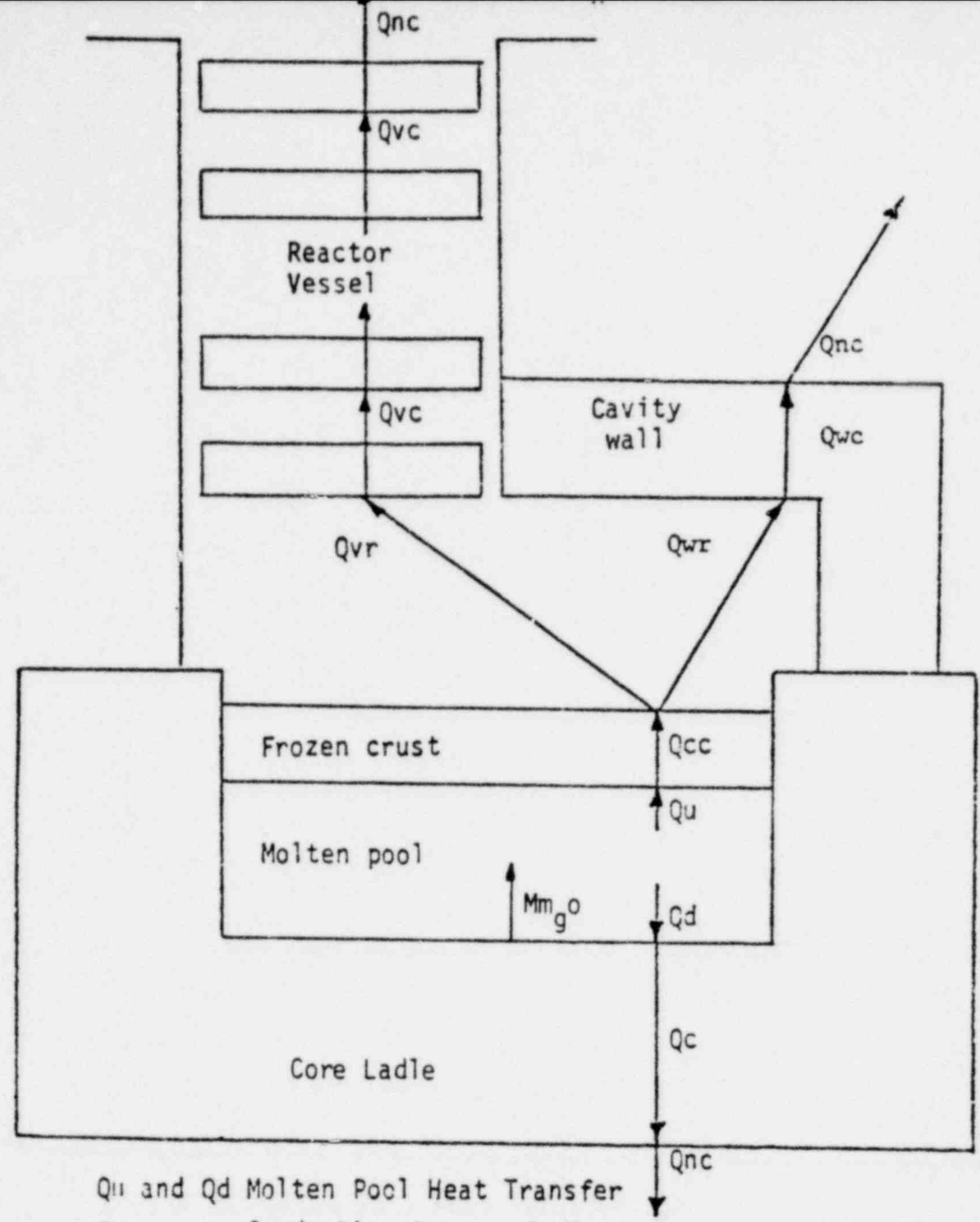


Heat and Mass Transfer in Reactor Cavity

THE MELSAC COMPUTER CODE

MODEL:

- INCORPORATES FEEDBACK EFFECT ON MOLTEN POOL H/T OF HEATING STRUCTURES AROUND LADLE.
- MOLTEN POOL INITIALLY PURE UO_2.
- AS LADLE MELTS, POOL DILUTED WITH MgO.
- CONDUCTION AHEAD OF MELT FRONT MODELED.
- CRUST CAN FORM ON UPPER POOL SURFACE.
  (CRUST COMPOSITION DEPENDS ON LOCAL POOL CONDITIONS)
- THERMAL RADIATION FROM POOL SURFACE TO REACTOR VESSEL AND CAVITY WALL.
- CAVITY WALL MODELED AS SLAB WITH ONE-DIMENSIONAL HEAT CONDUCTION.
- REACTOR VESSEL MODELED AS SERIES OF CONNECTED MASSES.



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_go Mass addition of molten m_go from meltfront.

Model used in MELSAC

THE MELSAC COMPUTER CODE

RESULTS OF SCOPING STUDY:

- LADLE PENETRATION TIME RELATIVELY INSENSITIVE TO POOL HEAT TRANSFER CORRELATIONS.
- WITH LOWER EUTECTIC TEMPERATURE OF $MgO-UO_2$ MIXTURE AS LADLE MELTING POINT - POOL RAPIDLY COOLS AND FREEZES.
- LADLE PENETRATION TIME INCREASED BY LESS THAN 10% IF NATURAL CONVECTION IS MODELED FROM BACK OF STRUCTURES (RATHER THAN ADIABATIC).
- ADDITIONAL HEAT SINK OF VESSEL INCREASES LADLE PENETRATION TIME BY 23%.
- LADLE PENETRATION RELATIVELY INSENSITIVE TO VARIATIONS IN EFFECTIVE EMISSIVITY BETWEEN POOL SURFACE AND REACTOR CAVITY STRUCTURES.
- WALL CONFIGURATION HAS APPRECIABLE EFFECT ON PROTECTING UPPER REACTOR CAVITY.

BEST ESTIMATE CASES

- WALL CONFIGURATION SCOPING STUDY:
- ASSUMPTIONS SAME AS CASE A:
  (EXCEPT FOR WALL CONFIGURATION)
  - CASE B - 0.91 M (3 FT) MgO AND 0.31 M (1 FT) CONCRETE
  - CASE C - 1.07 M (3.5 FT) MgO AND 0.15 M (0.5 FT) CONCRETE
- RESULTS (CASE B):
  - CONCRETE (<1473 K, 2190°F) AND STEEL (<810 K, 1000°F) IN CAVITY WALL PROTECTED FOR 2 DAYS.
  - LADLE PENETRATED IN 5.8 DAYS.
  - REACTOR VESSEL MELTED IN 1.8 DAYS.
  - FRACTION HEAT STORED IN WALLS AND VESSEL:

    0.76 AFTER 1 DAY
    0.5 AFTER 5.8 DAYS

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CONCLUSIONS

- GREATER than assumed by OPS HEAT TRANSFER TO STRUCTURES ABOVE POOL.
- AT LEAST 0.91 M (3 FT) MgO REQUIRED TO PROTECT CAVITY WALLS FOR 2 DAYS. OPS suggests [illegible]
- CORE HOLD-UP ~ 6 DAYS
- VESSEL MELTS ~ 2 DAYS
- DILUTION OF POOL BY STEEL AND ZIRCALLOY WOULD TEND TO INCREASE TIME SCALE OF ABOVE EVENTS.

APPROACH

- INTRODUCTION - STAFF
- UPDATED THERMAL EVALUATION OF FNP CORE LADLE - BY STAFF CONSULTANTS AT BNL
- UPDATED MATERIALS INTERACTION EVALUATIONS - BY STAFF CONSULTANTS AT THE MATERIALS SCIENCES LABORATORY OF AEROSPACE CORP
- PRESENTATION OF THE ABOVE EVALUATIONS SHOULD ANSWER ALL OF ACRS QUESTIONS* UNDER PART (A), ITEMS RELATED TO THE IMPACT THAT THE CORE LADLE WILL HAVE ON OTHER CONTAINMENT STRUCTURES
- STRUCTURAL CRITERIA - STAFF
- SUMMARY - STAFF

*REFERENCE: LETTER, R. F. FRALEY TO H. R. DENTON, "ACRS REVIEW OF THE FLOATING NUCLEAR PLANT CORE LADLE DESIGN," DATED JULY 25, 1979

SUMMARY

- FNP CORE LADLE CONCEPT IS FEASIBLE
- LADLE CAN BE ENGINEERED TO PROVIDE RETENTION OF A MOLTEN CORE FOR AT LEAST A PERIOD OF TIME IN THE RANGE OF TWO DAYS TO ONE WEEK
- OPS IS DEVELOPING A COUPLED HEAT TRANSFER CALCULATIONAL MODEL
- ANY SIGNIFICANT DIFFERENCES BETWEEN THE APPLICANT AND STAFF MODELS CAN BE RESOLVED DURING EARLY PHASES OF THE FINAL DESIGN
- EXPERIMENTAL RESULTS FROM APPLICABLE CORE MELT RESEARCH PROGRAMS WILL BE INCORPORATED INTO CALCULATIONAL MODELS
- ONCE A CALCULATIONAL MODEL IS AGREED UPON, THE LADLE CONFIGURATION CAN BE OPTIMIZED TO PROVIDE THE LARGEST POSSIBLE CORE RETENTION TIME CONSIDERING ALL THE FACTORS IN ITEMS A.2 THROUGH A.7 OF THE COMMITTEE'S LETTER OF JULY 25, 1979

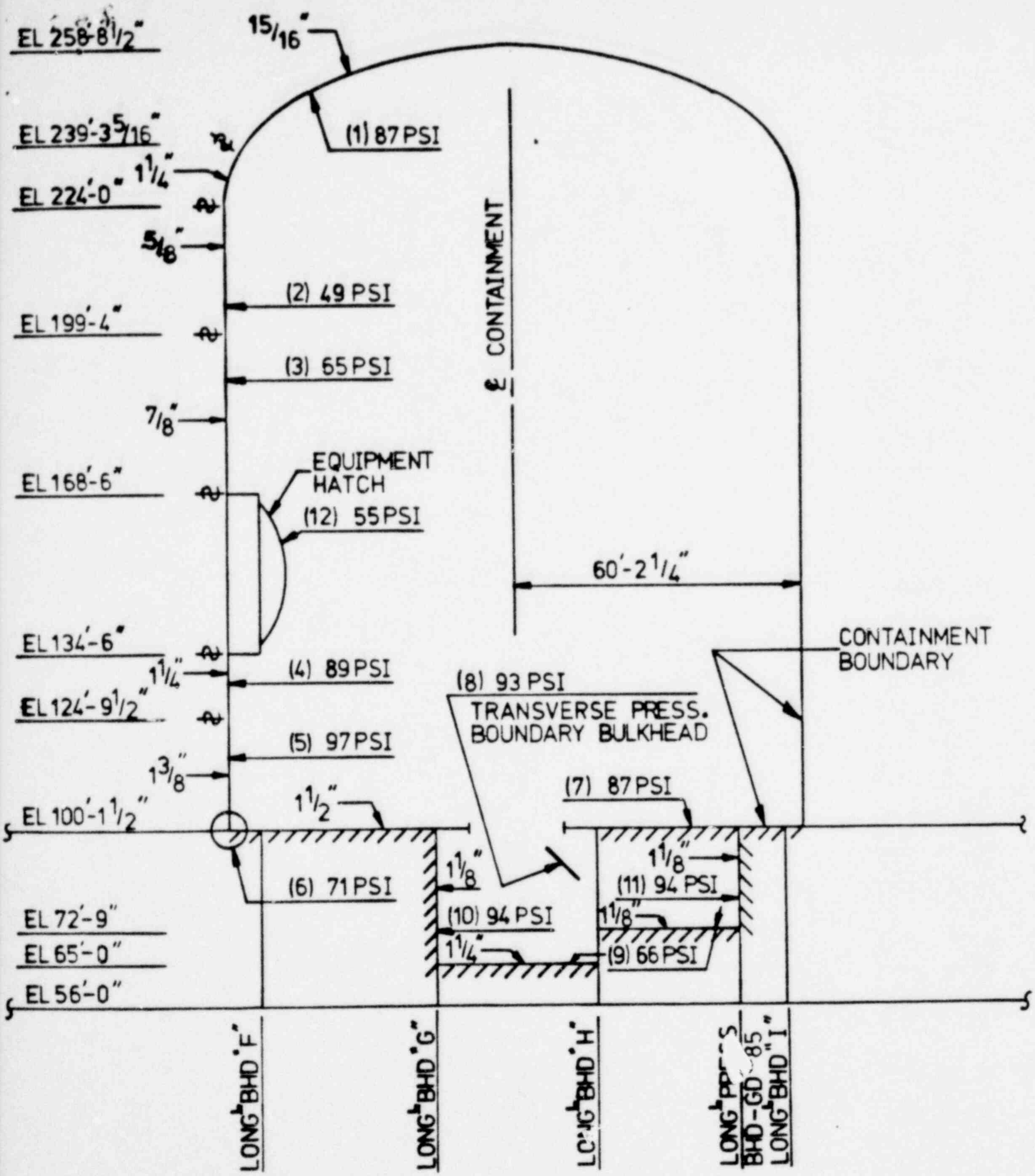
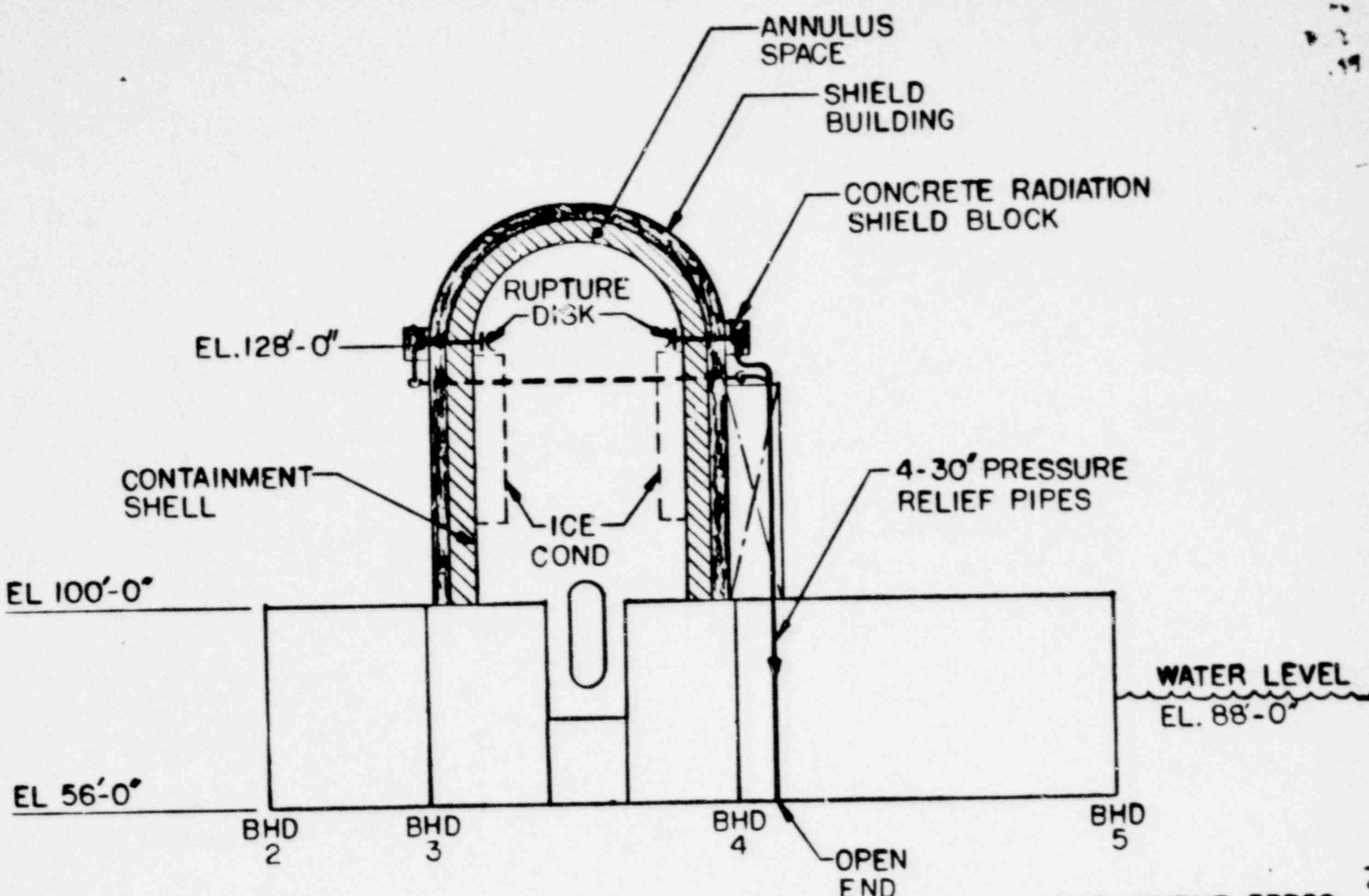


FIGURE #20

TRANSVERSE SECTION FRAME "3(b)"

ESTIMATED CONTAINMENT BOUNDARY FAILURE PRESSURES-(PSI)

F-11



CONTAINMENT PRESS.
RELIEF PIPING
SECTION VIEW AT
CONTAINMENT (COL.G.I)
LOOKING TO BHD.H

92