CERTIFIED 8/25/79

MINUTES OF THE ACRS SUBCOMMITTEE ON
THE FLOATING NUCLEAR PLANT
WASHINGTON, DC
JUNE 27, 1979

ACRS-1651

The ACRS Subcommittee on the Floating Nuclear Plant met with representatives of the NRC Staff and Offshore Power Systems (OPS) in Washington, D. C. on June 27, 1979, to continue its review of the Offshore Power Systems application for a manufacturing license for the Floating Nuclear Plant. The specific topic of the meeting was the review of the proposed magnesium oxide core ladle design. A notice of the meeting appeared in the Federal Register on June 12, 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. Siess, Acting Subcommittee Chairman, introduced the members of the Subcommittee and noted the purpose of the meeting was to review the proposed core ladle design. 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

2.1 Status of the FNP Application

Mr. Blair Haga, OPS, briefly discussed the FNP manufacturing license application status noting the following points:

- Application was made in early 1973.
- Atomic Safety and Licensing Board hearings are essentially complete. The ACRS must complete its review before the final hearings can be completed.

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o The OPS core ladle report has been amended to incorporate the NRC Staff's comments.

o OPS has answered all the NRC Staff's questions with regard to the ECCS; however, there are still some questions with regard to the McGuire application which involve a complete review of the ECCS/UHI.

Mr. Haga noted that OPS has formed a task group working closely with Westinghouse to evaluate the Three Mile Island Accident. He noted that it would be several years before an FNP was manufactured and that any TMI-related items on the FNP could be handled during the manufacturing period.

Mr. Haga stated that the FNP core ladle is a direct result of the liquid pathway study reviews with the ACRS. He noted that the Final Environmental Statement, Part 3, imposed the requirement for the core ladle on the FNP to mitigate environmental consequences. Since this is an environmental issue, they do not intend to do an exhaustive test and design effort for the ladle, but will apply the best available technology, which is the approach identified in the NEPA. Special emphasis is being given to ensure that the core ladle will not compromise the existing public health and safety requirements.

2.2 Status of the NRC Staff Review

Mr. Ralph Birkel, NRC Staff, discussed the status of the NRC Staff's review, noting that they have finished the environmental review and have issued Final Environmental Statement, Part 3, which requires the concrete beneath the reactor vessel be replaced with a pad constructed of magnesium oxide. He noted that OPS submitted Topical Report No. 36A59, FNP Core Ladle Design and Safety Evaluation. The NRC Staff is now evaluating Topical Report No. 36A59. He noted that the NRC Staff has not reviewed the structural aspects of the core lau.. design due to shortage of manpower resources.

Mr. Birkel noted that the NRC Staff has about 10 items to finish in its review and that if adequate priority is given, they should finish the review and issue SER Supplement No. 3 some time during the summer of 1979. The safety aspects of the review were covered in a 1975 SER. The present NRC Staff core ladle review will be performed to ensure that the changes being made to accommodate environmental requirements do not have any deleterious safety implications that would change or void any of their earlier conclusions. He added that the NRC Staff feels that the results of the liquid pathways study was a keystone to the FES, Part 3, which reflects environmental conditions which the NRC Staff is obliged to address.

Mr. Birkel suggested that in order to effectively benefit from the Subcommittee Meeting, that the ACRS issue a letter to Mr. Gossick on the proposed core ladle design, similar to letters issued on the liquid pathways study.

The Subcommittee discussed the difficulty in separating environmental from safety concerns. Dr. Siess noted that the FES discusses a number of benefits from delaying a melt-through and that some of these benefits relate to safety. Mr. Haupt, NRC Staff, noted that when the question of whether it was appropriate to include the Class 9 accident in the FNP Environmental Report was brought before the Commission, they supported the NRC Staff; however, no order was ever issued, possibly because OPS agreed to incorporate a core ladle, making the order unnecessary. Haupt indicated the core ladle is an environmental matter simply because it was covered in the environmental report. Mr. Birkel noted that even without the core ladle the 10 CFR 100 calculated doses are acceptable for the FNP.

2.3 NRC Staff's Evaluation of the OPS FNP Core Ladle

Mr. Andrew Marchese, NRC Staff, discussed the background of the NRC requirement for the core ladle and also provided a brief status report on their evaluation of the ladle. He noted that OPS, in Topical Report No. 36A59, confirmed and concurred with the NRC Staff's conclusion in FES III, Appendix E, that magnesium oxide appears to be the most promising candidate for the refractory material of the core ladle and that there was reasonable assurance that a delay of melt-through of molten core debris on the order of two days was feasible. Marchese stated that based on their review, so far, together with the Applicants commitments, the Staff is of the opinion that OPS has met the FES III requirement in the area of delaying core melt-through. The NRC Staffs' "best estimate time" for the proposed design for delaying a full core melt-through is 5 to 8 days.

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Mr. Marchese discussed several NRC related R&D informational needs concerning the interac ions of the molten core debris materials with the refractory sacrificial materials (See Attachments 1 and 2). RES is considering the NRR research request on this item for completion over the next three years.

Mr. Etherington noted several items discussed earlier which he thought needed further consideration. These included:

- o Thermal shock of magnesite
- o Use of magnesite in ladle lining
- o Bonding of brick without mortar
- o Applicability of steel-making experience to core-melt
- o Water intrusion into the magnesite
- o Expansion of bricks at the joints
- o Lack of ventilation of the concrete

3.0 Technical Presentations by Offshore Power Systems

3.1 Summary of the Core Ladle Design Requirements

Dr. Dee Walker, OPS, discussed the OPS interpretation of the FES III functional requirements for the core ladle design. OPS has transformed the FES III requirements into the following set of functional requirements:

1. Maximize debris retention time within available space constraints
   a) Minimum delay time - about 2 days
   b) Minimum refractory thickness - 4 feet
2. Ladle materials shall not form a large volume of gas which can sparge through melt.
3. Ladle volume shall be sufficient for core materials and 25% lower vessel steel.
4. Ladle design shall not compromise the integrity of the containment boundary or platform.
5. 15 mrem/hr radiation limit retained for adjacent compartments outside containment (during plant operation).

Dr. Walker discussed the method they used to calculate the core melt debris volume estimates for sizing the ladle. Details of these calculations are provided in Attachments 3 and 4. He also discussed the properties of candidate materials for the core ladle, including magnesium oxide (See attachment 5).

3.2 Description of Ladle Design

Mr. Clint Dotson, OPS, provided a description of the ladle and discussed the location of the ladle within the FNP (See Attachments 6 & 7).

Mr. Dotson presented a ½ inch to 1 foot scale model of the portion of the FNP including the components around the reactor and the ladle. A scale model of the entire FNP, which is about 35% complete, is available in Jacksonville, Florida for inspection.

Mr. Dotson defined the following design requirements they have established for construction of the ladle.

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 shielding 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 requireemnts.
6. The ladle shall be as thick as practicable within the various 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 basin motions ($1/2^{o}$).

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.

3.3 Boundary Constraints

Mr. Dotson discussed the boundary constraints for the ladle. He noted that the space beneath the reactor vessel available for placement of refractory material for retarding the debris is bounded by the containment boundary pressure bulkheads, the platform primary strength bulkheads, and the incore instrument cables (See Attachments 8 & 9).

3.4 Configuration and Arrangement

Mr. Dotson discussed the configuration and arrangement of the ladle. The size of the ladle was selected to utilize the available space within the various design constraints and still maintain a thickness of not less than four feet in any direction, with a pool volume sufficient to accommodate molten core debris of approximately 920 cubic feet. Attachments 10 and 11 provide the details of the configuration of the ladle and the arrangement of the magnesium oxide bricks.

Mr. Dotson discussed the comparison of a configuration of an existing furnace design and the proposed ladle. The arch design and interlocking brick were used in both.

Mr. Dotson noted the FNP ladle brick will be installed inside a steel box (See Attachment 12). The steel box will provide waterproofing and a gap between the ladle and surrounding concrete for venting gases released from the concrete. Dotson indicated that experience in the steel industry shows that locking of the brick from thermal expansion should minimize or eliminate the possibility of the bricks coming loose or floating out, especially considering the tongue-and-groove arrangement.

OPS has suggested that a 14-foot brick wall be erected directly above the ladle and around the entire perimeter of the lower cavity. This brick wall will shield the concrete from the thermal radiation produced from the melt debris and will limit the primary platform steel temperature such that the structure can withstand the expected loading conditions for the duration of the debris retention. Attachment 13 shows the proposed location of the vertical brick wall.

3.5 Platform Motion and Loading Conditions

In response to questioning from Dr. Siess concerning the design basis for the volume of the ladle, Dotson noted that it would take a little more than 1/2 degree tilt at the design basis debris volume to spill over the inner edge of the ladle. A 1/2 degree tilt is a design basis to occur no more than 10% of the time at the worst site under normal wind and wave conditions. It would take a 2 degree tilt, a 100-year condition, to get to the upper lip.

Mr. Dotson discussed the loading conditions for the reactor cavity structure and core ladle during normal plant operation and design basis events (See Attachment 14).. He also discussed the loading conditions for the reactor cavity structure and core ladle subsequent to a postulated reactor vessel melt-through (See Attachment 15).

The Subcommittee discussed the possibility of large amounts of moisture entering the steel box surrounding the brick and the possible effects on the performance of the ladle. The Applicant does not consider this to be a significant problem.

3.6 Melt Penetrating Calculations and Gas Generation Estimates

Dr. Henry Stumpf, OPS, discussed their estimate of magnesium oxide erosion due to a molten pool. They assumed a fixed fraction of decay heat directed into the magnesium oxide bed in both the horizontal and vertical directions from 10% to 100%. Other assumptions were uniform heat flux in the horizontal and vertical directions and no conduction into the magnesium oxide ahead of the melt front. The heat source used assumed:

- o Entire core melt
- o 20% of decay heat lost through volatiles
- o Entire heat from zirconium-water reaction was used.

Erosion depths are shown on Attachment 16, showing 6 days penetration time for the eight-foot thickness assuming 100% of the heat being directed into the melt.

In response to questions concerning erosion rates, Mr. Arthur Chait, Harbison-Walker, noted that the bottoms of ladles, which had been contact with liquid metals, typically erode in a uniform fashion. He indicated that he had not experienced rapid erosion, as might be expected with a core melt. Mr. Chait said that with uniform erosion he would expect wear to include a combination of slagging and melting.

In response to a question from Dr. Siess concerning the expected ratio of heat into the melt versus radiation upward, Stumpf expected about 80% of the heat to be directed to the melt. He expected very little heat to the surrounding walls after one or two hours, once the wall temperature reaches a temperature near that of the pool.

Mr. Stumpf discussed outgassing of the basaltic concrete beneath the ladle and on the sides. He indicated that you would probably begin to outgas the concrete long before the melt front reaches the concrete. The gas from the concrete would be vented out the side of the ladle so it would not bubble through the molten pool. Stumpf noted that if you outgas all of the concrete under the ladle, it would increase containment pressure roughly 6½ psi. He added that outgassing the walls surrounding the pool and above the pool would add another 9 or 10 psi to the containment pressure.

Mr. Bob Brusoloff, OPS, indicated they have not calculated the containment temperature as a result of a core melt accident. Mr. Haga said this was not being addressed since the ladle is a requirement of the liquid pathway study. The Subcommittee expressed some concern that delaying the melt-through may increase the chance of earlier containment rupture and higher airborne releases.

Mr. Stumpf discussed their calculation of the thermal effects of radiation from the pool to the side walls. For this calcualtion, OPS assumed that the heat radiation from the pool was equal to the total decay heat being generated at that time. As the walls heat up they will radiate heat back to the pool so that the net transfer is small once the walls reach equilibrium pool surface temperature. Attachment 17 shows expected wall temperatures of brick and concrete. Stumpf noted that since the melting point of basaltic concrete is around 2650^o R, and since the expected temperature due to upward radiation is above this temperature, that all surfaces in the cavity will have to be lined.

Mr. Marchese noted that the staff is evaluating the advantages and disadvantages of pumping water into the core melt debris. He noted that an advantage is that it would remove heat but the disadvantages are steam explosions and sparging of radioactivity. He added, that the decision has not been made whether to require the operator to turn off ECCS pumps once it is known the core has melted. It was noted that since the sump water contains, the activity sparged by the containment spray system, pumping the sump onto the debris would likely add to the air release and to the liquid release; however, there is also the possibility that it would cool the core and prevent burn-through.

In response to questions from Mr. Etherington concerning the ability of the structures to support the vessel following a core melt, Haga said they are only concerned that the vessel is supported for the specified length of time until burn-through of the platform occurs.

Mr. Stumpf discussed containment pressure build-up due to outgassing of the concrete (See Attachment 18). He estimated the entire decomposure of 2000 cu. ft. of basaltic concrete around the vessel and 3000 cu. ft. on the surrounding walls would add 16 psi to the containment pressure.

3.7 Dose Estimates for Regions Surrounding the Ladle

Dr. Walker discussed the dose estimates during normal operation for the lower cavity walls and occupied spaces below the cavity. He noted that the design dose rate limit is the same as it was before the ladle was incorporated in the design. The calculated dose rate beneath the ladle with the current design is 6 mrem/hr. He added, that there was a margin of about two feet of MgO which could be removed and still meet the 15 mrem/hr limit.

3.8 Related Experience with Refractory Materials in the Steel Making Industry

Mr. Chait discussed some of Mr. Etherington's earlier comments. He noted that magnesite is not normally used in ladles in the steel industry. He added that magnesite is very sensitive to thermal shock; however, a single dump would only cause thermal shock to the initial surface. The FNP ladle has chemically bonded brick which will take the initial thermal shock.

Mr. Chait discussed his experience involving inspection of ladles and furnaces following use, noting that even though the bricks crack, they do not float up on top of the melted metal or fall out when the ladle is turned upside down.

Mr. Noble Seaborne, OPS, discussed experience from steel-making furnaces and operations and related this experience to the FNP ladle. Seaborne discussed comparable sizes, drop heights, and temperatures in the steel-making industry with core-melt debris in the FNP.

4.0 Miscellaneous Discussion

The Subcommittee discussed some of the Sandia comments on Topical Report 36A59 which were presented in a letter dated May 23, 1979. In response to comments on the Sandia letter , OPS and Harbison-Walker representatives noted the following:

o The decay heat from complete oxidation of the zirconium is assumed

o The decay heat from chromium oxidation was not considered. OPS agreed to look into this.

o Examination of samples from industrial processes which are at lower temperatures than a core melt, shows that chemical attack does play a significant role in the failure of magnesium oxide bricks.

o There is a lot of experience that would indicate that flotation of the brick will not be a problem.

o The ratio of the total weight of MgO to UO_2 is about 100 to 1. The ratio of the total weight of MgO to molten steel is about 750 tons to 200 tons.

o Experience has shown that refractories that have slagged and been altered were damaged during cooldown; therefore, inspecting cooled bricks for damage may not be representative of the conditions occurring during a core-melt accident.

5.0 Conclusions/Remarks

Mr. Haga discussed the importance to OPS of obtaining a manufacturing license for the FNP. He noted that they are preparing proposals to be submitted this summer to utilities which have expressed a very strong interest in building an FNP. He did not think that any utilities would make any decision on the FNP until such time as OPS has a manufacturing license.

Dr.Siess discussed Mr. Birkel's request to have the ACRS perform a review of the preliminary ladle design and to write a letter to Mr. Gossick commenting on the design before the NRC Staff finishes its review. Dr. Siess said he would discuss this request with the ACRS at the July 1979 ACRS Meeting. If the ACRS agrees, OPS and the NRC Staff would likely come to the August 1979 ACRS Meeting to discuss the preliminary design of the core ladle and any other liquid pathway study related items.

The meeting was adjourned at 3:25 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.

Headquarters, Washington, DC 20546. Telephone 202/755-3252.

June 5, 1979.

Russell Ritchie,

Acting Associate Administrator for External Relations.

[FR Doc. 79-18183 Filed 6-11-79; 8:45 am]

BILLING CODE 7510-01-M

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Subcommittee on Radiological Effects and Site Evaluation; Meeting

The ACRS Subcommittee on Radiological Effects and Site Evaluation will hold an open meeting on June 27, 1979 in Room 1046, 1717 H St., N.W., Washington, D.C. 20555 to discuss changes in the NRC research program budget and several other matters in the areas of radiological effects and site evaluation. Notice of this meeting was published on May 24, 1979 (44 FR 30177).

In accordance with the procedures outlined in the Federal Register on October 4, 1978 (43 FR 45926), 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:

Wednesday, June 27, 1979

8:30 a.m. until 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 pertinent to this review with representatives of the NRC Staff and invited speakers from outside NRC.

The Subcommittee may then caucus to determine whether the matters identified in the initial session have been adequately covered and whether the project is ready for review by the full Committee.

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. Ragnwald Muller, (telephone 202/634-1413) between 8:15 a.m. and 5:00 p.m., EDT.

Dated: June 6, 1979.

John C. Hoyle,

Advisory Committee, Management Officer.

[FR Doc. 79-17830 Filed 6-11-79; 8:45 am]

BILLING CODE 7590-01-M

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 June 27, 1979, in Room 1167, 1717 H Street, N.W., Washington, DC 20555 to continue its review of the Offshore Power Systems' application for a manufacturing license for the Floating Nuclear Plant. The specific topic of this meeting will be the review of the proposed core ladle design using magnesium oxide bricks. Notice of this meeting was published on May 24, 1979 (44 FR 30177).

In accordance with the procedures outlined in the Federal Register on October 4, 1978, (43 FR 45926), 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:

Wednesday, June 27, 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.

The Subcommittee may then caucus to determine whether the matters identified in the initial session have been adequately covered.

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. Quittschreiber, (telephone 202/634-3267) between 8:15 a.m. and 5:00 p.m., EDT.

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.)

Dated: June 6, 1979.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 79-18048 Filed 6-11-79; 8:45 am]

BILLING CODE 7590-01-M

POSTAL RATE COMMISSION

[Docket No. MC79-3]

Red-Tag Proceeding, 1979; Deferring Procedural Deadlines

June 5, 1979.

As directed in the Order Denying Motion to Postpone Proceeding issued on April 27, 1979, the United States Postal Service submitted a testimonial filing in this docket on May 31, 1979. A preliminary examination of the Service's filing reveals that it contains technically

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PRESENTATION SCHEDULE
FLOATING NUCLEAR PLANT SUBCOMMITTEE MEETING
JUNE 27, 1979
WASHINGTON, D.C.

POOR ORIGINAL

		PRESENTATION TIME	APPROXIMATE TIME
	MEETING WITH OFFSHORE POWER SYSTEMS AND THE NRC STAFF (OPEN SESSION)		
1.0	Subcommittee Chairmans Opening Remarks		8:45 am
2.0	Introductory Remarks		
2.1	Applicants Summary of Functional Design Requirements and Materials Selection	15 min	8:50 am
2.2	NRC Staff Discussion of Review Schedule and Major Problems in Review, if any	5 min	9:20 am
3.0	Technical Presentations by Applicant with NRC Staff Response		
DOTSON 3.1	Description of Ladle Design		
3.1.1	Design Contraints	10 min	9:30 am
3.1.2	Configuration	10 min	9:50 am
3.1.3	Ladle Support	10 min	10:10 am
3.1.4	Platform Structural Consideration	10 min	10:30 am
	Coffee Break	10 min	10:50 am
3.1.5	Platform Motion and Seismic Design Criteria	10 min	11:00 am
3.1.6	Miscellaneous Considerations	10 min	11:20 am
STUMPF 3.2	Melt Penetration Calculations and Gas Generation Estimates	20 min	11:40 am
WALKER 3.3	Dose Estimates for Region Beneath Ladle	10 min	12:20 pm
	Break for Lunch		12:40 - 1:40 pm
SEABORNE 3.4	Discussion of Related Experience with Refractory Materials in Steel Making Industry	45 min 30	1:40 pm
4.0	Caucus		3:10 pm
	- Conclusions/Remarks - Discuss Future Meetings Schedule		
	Adjournment		3:20 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% of the presentation time, has been made for questioning by the Subcommittee.

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ACRS FNP
SUBCOMMITTEE MEETING
JUNE 27, 1979
WASHINGTON, DC

ATTENDEES LIST

ACRS

C. Siess, Acting Chairman
P. Shewmon, Member
W. Mathis, Member
H. Etherington, Member
I. Catton, Consultant
G. Quittschreiber, Staff*

* Designated Federal Employee

NRC

R. Birkel
C. Haupt
A. Marchese
M. Silverberg

OPS

B. Haga
D. Walker
C. Dotson
H. Stumpf
N. Seaborne
R. Brusoloff
R. Capo

Harbison - Walker Refractories

A. Chait

Sandia Laboratory

D. Powers

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ATTACHMENT C

DOCUMENTS PROVIDED TO THE
SUBCOMMITTEE FOR THIS MEETING

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

2. Sandia Report, dated May 23, 1979, Review of Information Needs for Design of a MgO Core Retention Device.

3. View graphs shown at the meeting are provided as Attachments 1 through 13. A complete set of all handouts are provided in the meeting transcript and in the ACRS Office file for this meeting.

MOLTEN CORE DEBRIS MATERIALS INTERACTIONS
WITH REFRACTORY SACRIFICIAL MATERIALS
R & D INFORMATIONAL NEEDS

- RATE AND EXTENT OF CORE MELT PENETRATION INTO THE SACRIFICIAL MATERIAL, INCLUDING PENETRATION INTO CRACKS BETWEEN BRICKS OF THE SACRIFICIAL MATERIAL;
- EXTENT OF THERMAL SHOCK CRACKING AND/OR SPALLATION OF THE SACRIFICIAL MATERIAL;
- QUANTITY AND COMPOSITION OF VAPORS AND GASEOUS PRODUCTS RELEASED FROM THE SACRIFICIAL MATERIAL;
- HIGH-TEMPERATURE THERMOPHYSICAL PROPERTIES, ESPECIALLY THE LOWEST MELTING POINT EUTECTIC TEMPERATURE OF THE SACRIFICIAL MATERIAL AFTER EXPOSURE TO CORE MELT DEBRIS;

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R&D INFORMATIONAL NEEDS (CONTINUED)

- DETERMINATION OF ANY SIGNIFICANT CHEMICAL INTERACTIONS OR DISSOLUTION PROCESSES, INCLUDING CHARACTERIZATION OF THE REACTION PRODUCTS, PHASES PRESENT AND EXTENT OF SLAG-LINE ATTACK;

- EFFECT ON THE ABOVE IF FLOODING THE SACRIFICIAL MATERIAL WITH WATER PRECEDES THE INTRODUCTION OF CORE MELT MATERIAL;

- DETERMINATION OF A SUITABLE LAYERED BRICK CONFIGURATION TO PREVENT FLOATATION OF THE SACRIFICIAL MATERIAL BY CORE MELT DEBRIS AND ALSO WITHSTAND THE MECHANICAL IMPACT IF THE LOWER REACTOR VESSEL HEAD SHOULD DROP;

- MEASUREMENT OF THE HEAT TRANSFER SPLIT (UPWARD/DOWNWARD/SIDEWARD) FROM THE CORE MELT MATERIAL WITH AND WITHOUT THE EFFECT OF FLOODING THE MELT WITH WATER.

D. Walker

CORE MELT DEBRIS VOLUME ESTIMATES

(FOR 10% AND 25% OF LOWER INTERNAL AND REACTOR VESSEL HEAD STEEL MELTED)

MATERIAL	SOURCE	WEIGHTS LBS.	DENSITY $\#/FT^3$	VOLUME (10% CASE) FT^3	VOLUME (25% CASE) FT^3
UO_2	FUEL (100%)	223,000	550	405	405
ZIRCALLOY-4	CLADDING (100%)	51,000	375	136	136
STEEL	CORE (100%)	11,500	400	29	29
	LOWER REACTOR VESSEL HEAD (10 OR 25%)	22,500*	400	23	56
	LOWER INTERNALS (10 or 25%)	60,000*	400	60	150
SILVER, ETC.	CONTROL RODS (100%)	5,000	540	9	9
TOTAL		373,000		662	785

* 25% VALUES IN WEIGHT COLUMN

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CORE DEBRIS VOLUME ESTIMATES WITH IRON AND ZIRCALLOY OXIDI[illegible]

(ASSUMES 25% OF LOWER INTERNALS AND LOWER HEAD MELTED)

FRACTION IRON OXIDIZED	FRACTION ZIRCALLOY OXIDIZED	VOLUME FT^3
0.0	0	785
0.0	.75	848
0.0	1.0	869
0.1	1.0	886
0.25	1.0	911

DENSITY FOR ZrO_2 TAKEN AS 313 $\#/FT^3$; FOR FeO TAKEN AS 300 $\#/FT^3$

D. Walker

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PROPERTIES OF CANDIDATE LADLE MATERIALS

MATERIAL	MELTING POINT (^{o}C)	DENSITY (G/CM^3)	SPECIFIC HEAT ($CAL/G^{o}C$)	HEAT OF FUSION (CAL/G)	VOLUMETRIC HEAT ABSORPTION CAPABILITY VHAC (CAL/CM^3)
ALUMINUM OXIDE	2037	4.0	0.272	256	2939
GRAPHITE	2760	1.9	0.458	-	2218
MAGNESIUM OXIDE	2852	3.5	0.313	428	4168
SILICA	1728	2.32	0.266	31	1065
THORIUM OXIDE	2800	9.95	0.07	72	2448
TITANIUM CARBIDE	3076	4.8	0.207	283	3976
URANIUM OXIDE	2815	11.0	0.07	71	2675
ZIRCONIUM OXIDE	2760	5.7	0.155	169	3416

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D. Walker

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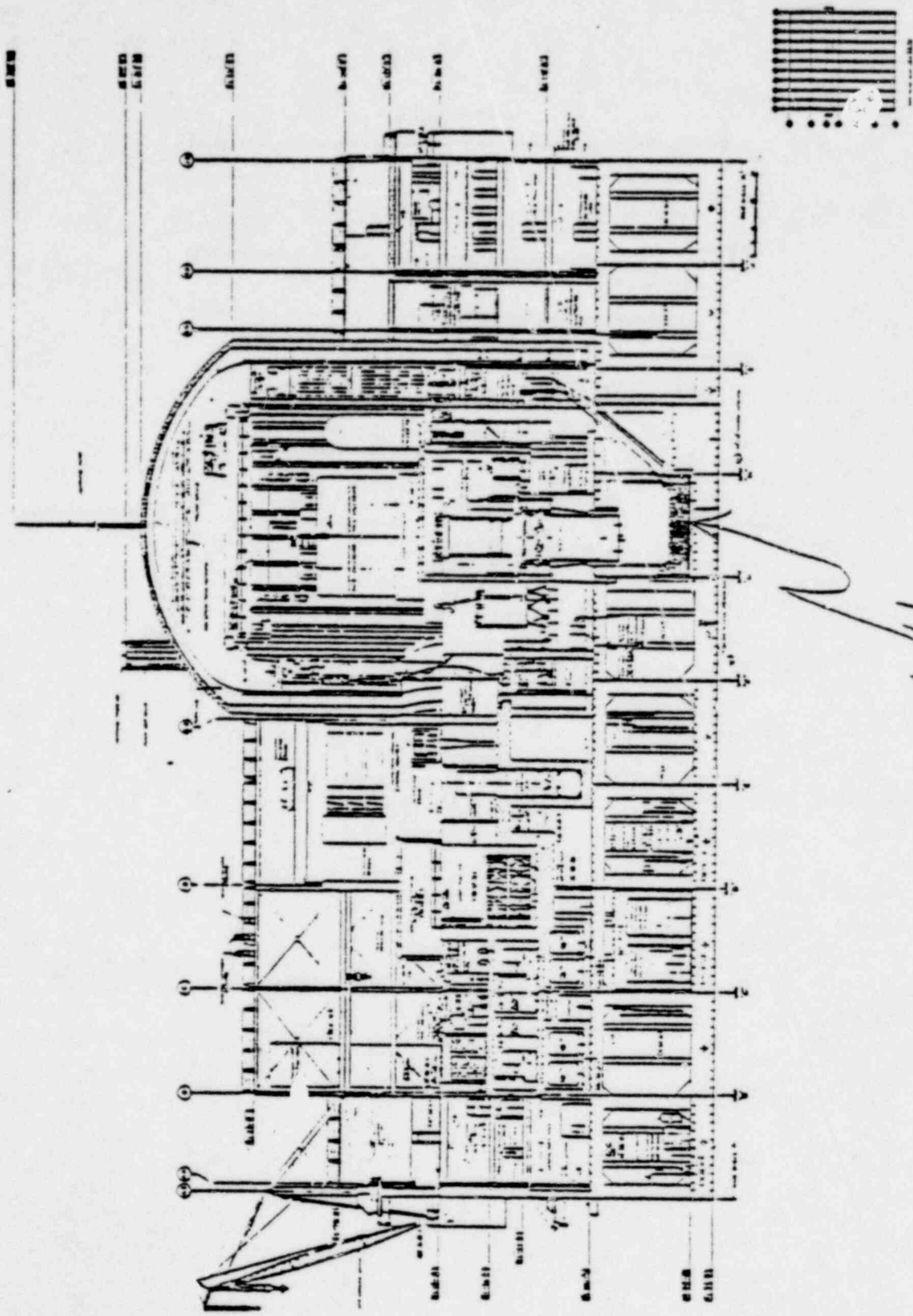


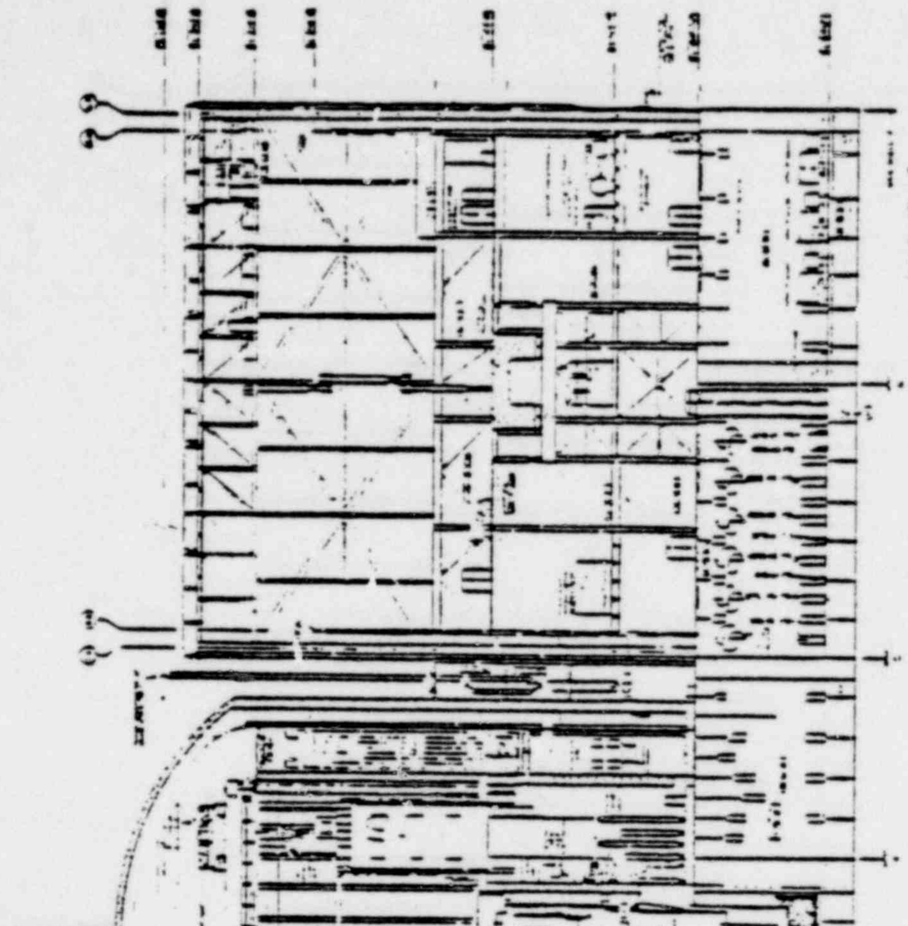
ladle

POOR ORIGINAL

1366 040

6

Dotson

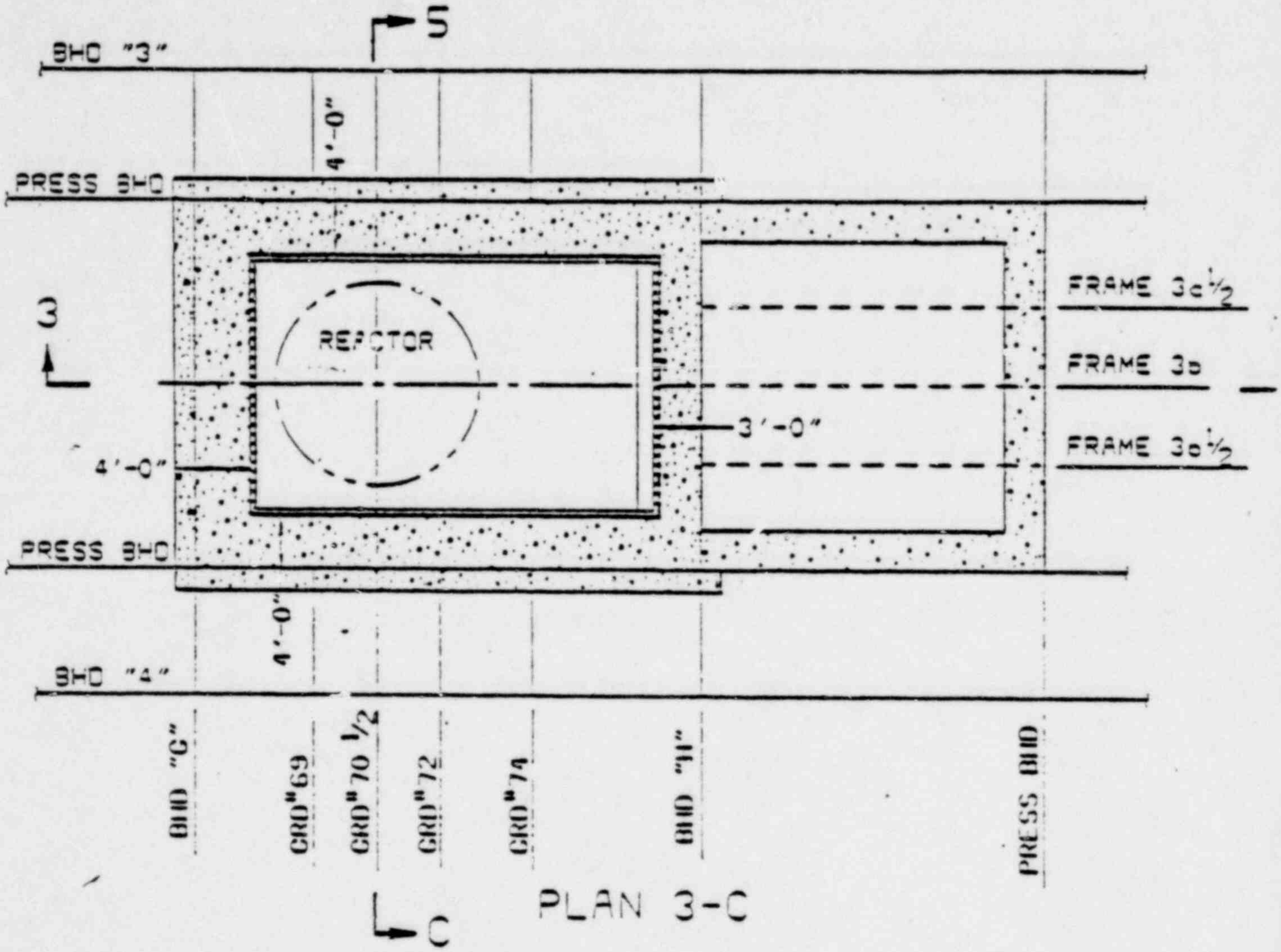


POOR ORIGINAL

1366 041

7

Dolsm



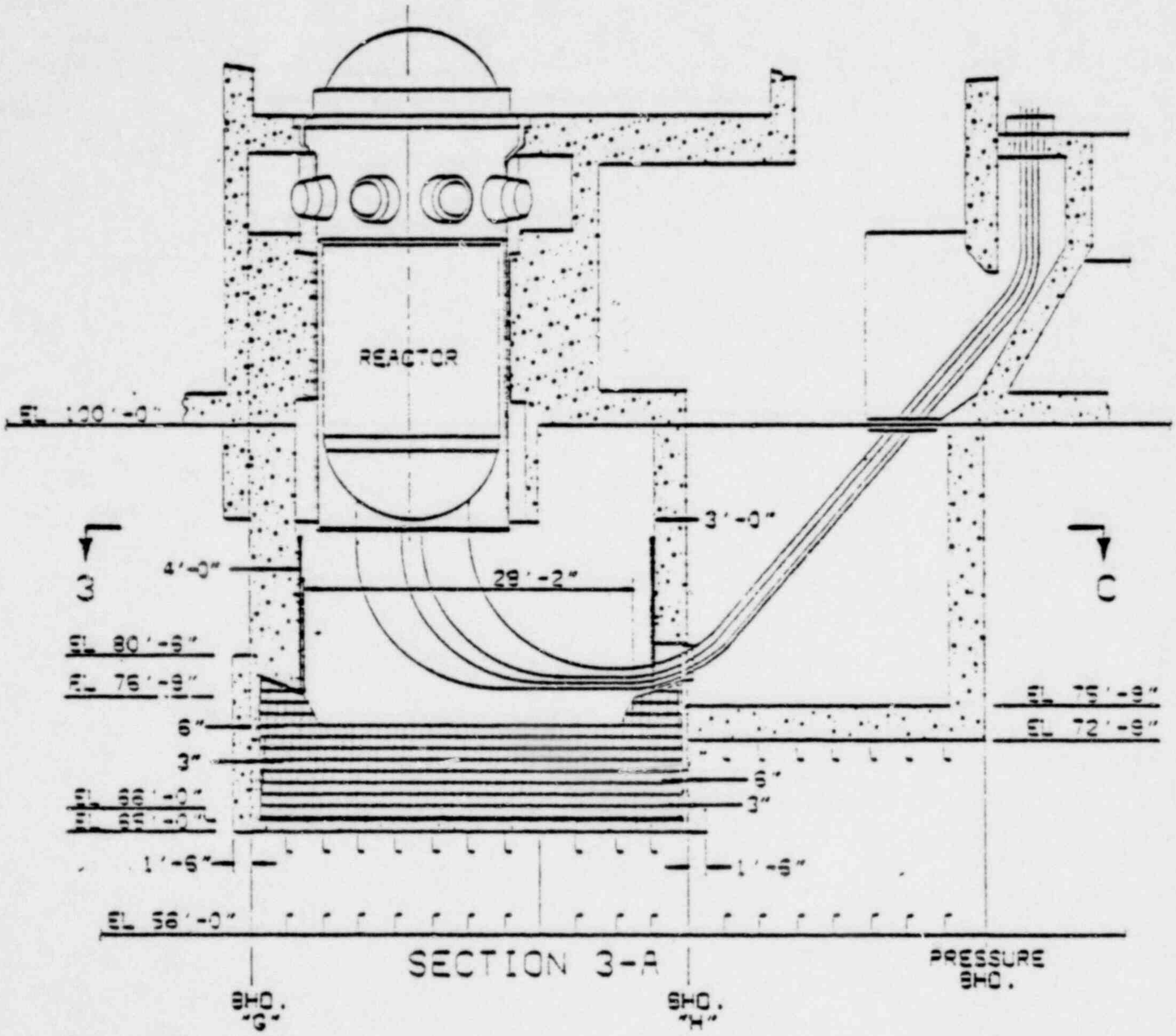
PLAN 3-C

POOR ORIGINAL

1366 042

8

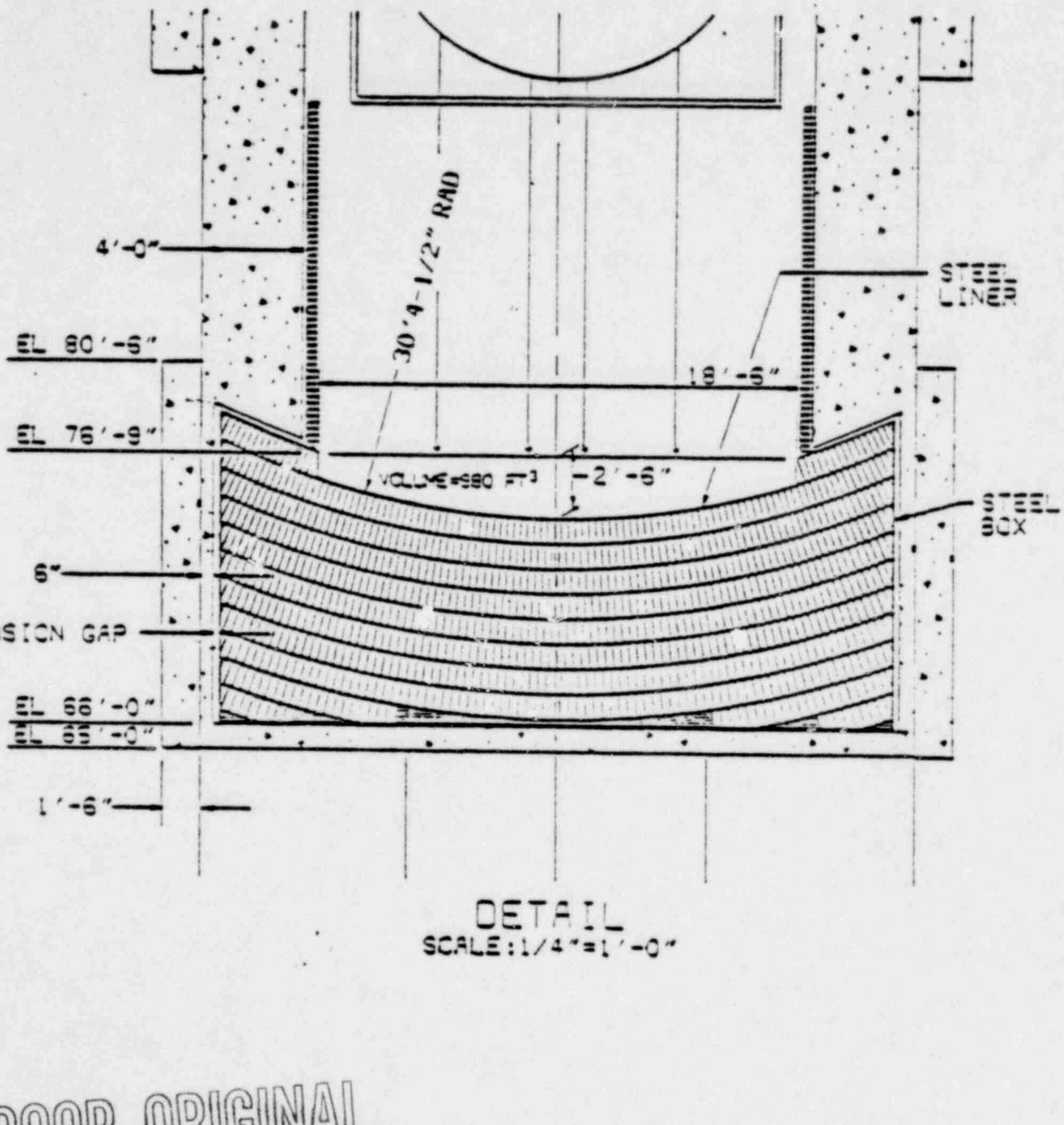
Dotson



SECTION 3-A

1366 043

Dotson



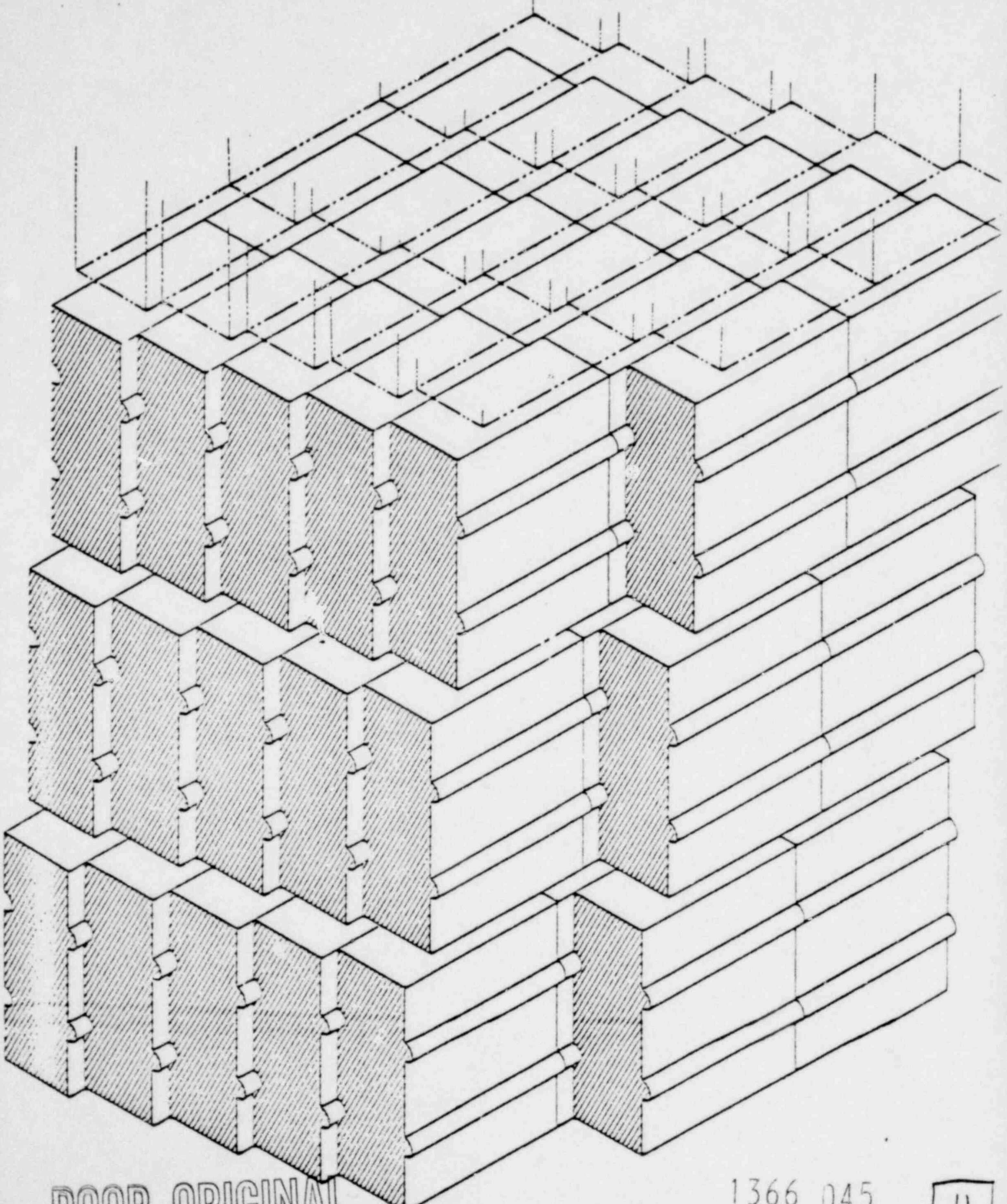
DETAIL
SCALE:1/4"=1'-0"

POOR ORIGINAL

1366 044

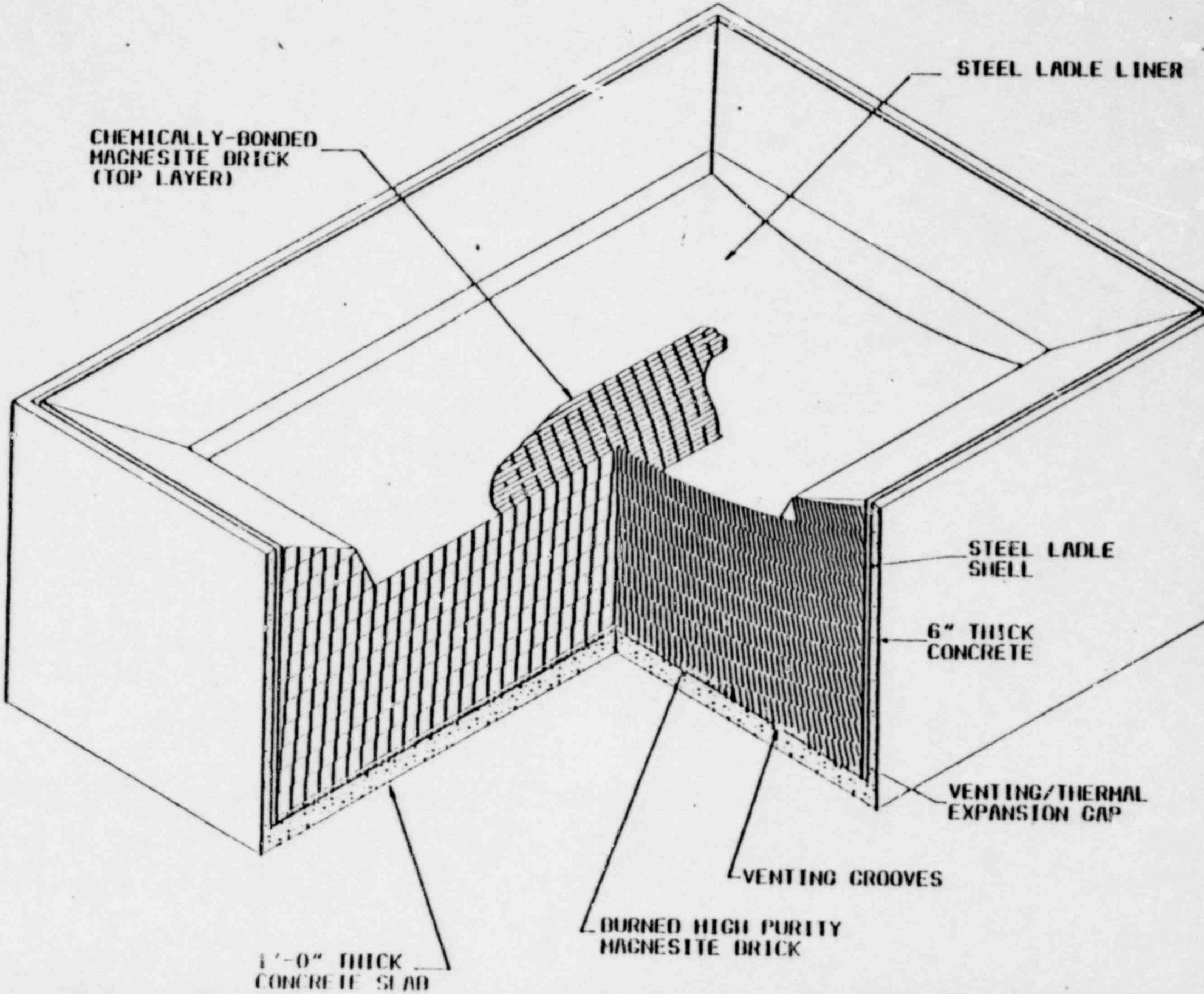
10

Dotson



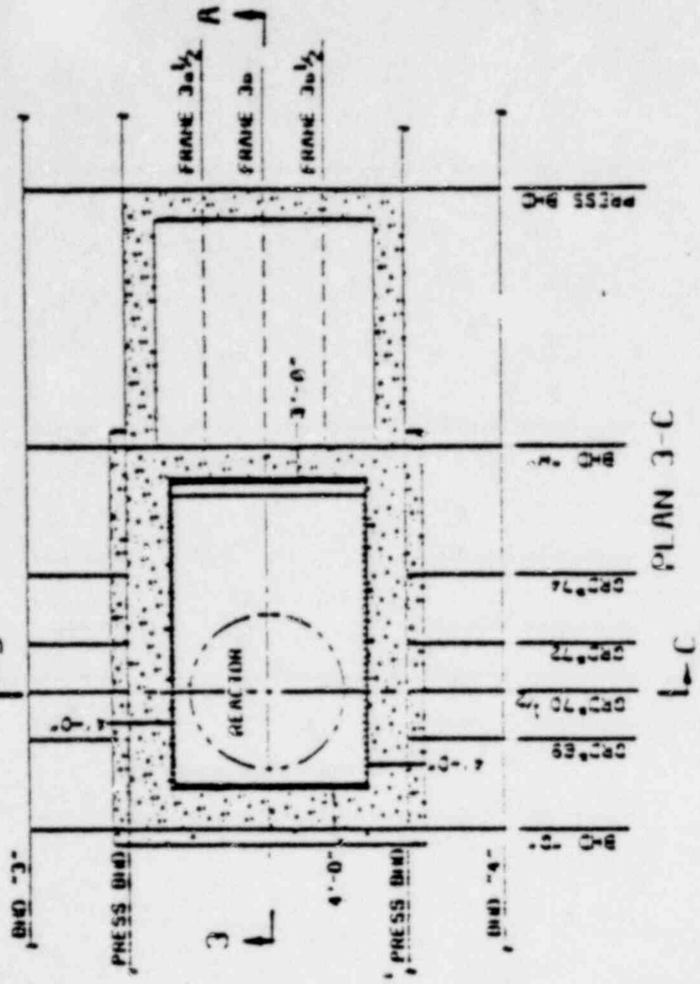
POOR ORIGINAL

1366 045

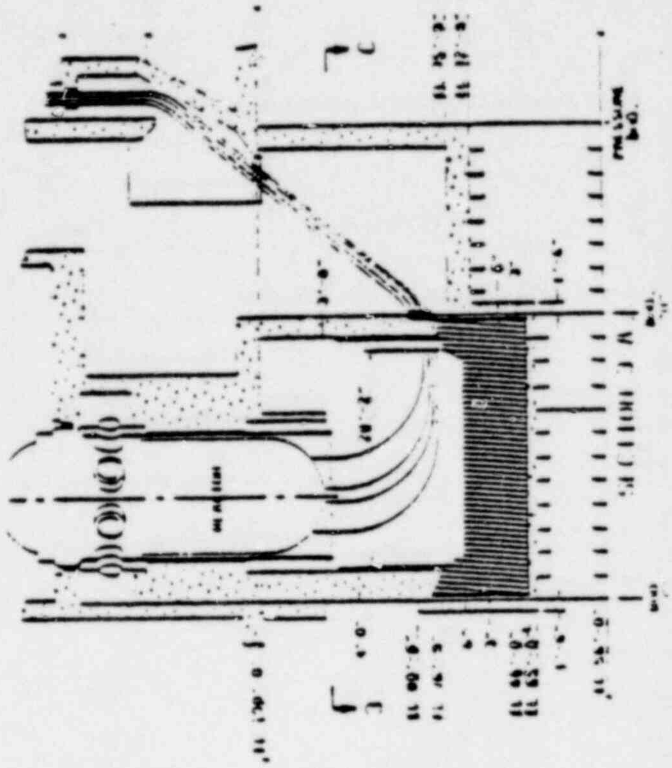

STEEL LADLE LINER
CHEMICALLY-BONDED
MAGNESITE BRICK
(TOP LAYER)
STEEL LADLE
SHELL
6" THICK
CONCRETE
VENTING/THERMAL
EXPANSION GAP
VENTING GROOVES
BURNED HIGH PURITY
MAGNESITE BRICK
1'-0" THICK
CONCRETE SLAB

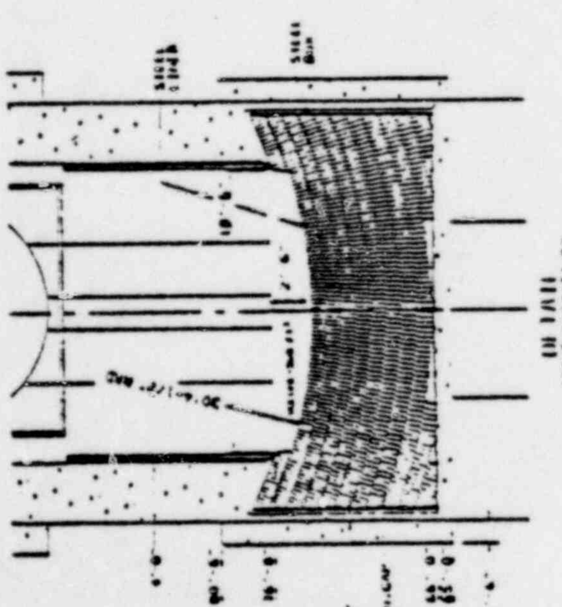
1366 046

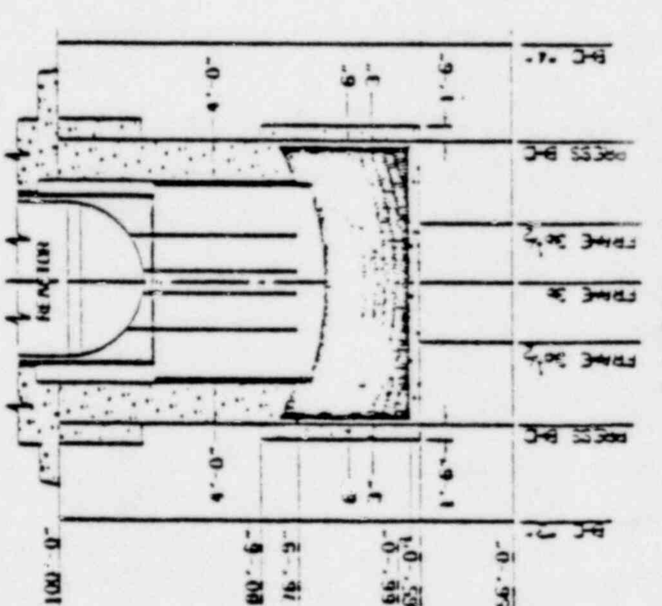
Dotsm



PLAN 3-C







POOR ORIGINAL

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B. LOADING CONDITIONS FOR REACTOR CAVITY STRUCTURE AND CORE LADLE DURING NORMAL PLANT OPERATION AND DESIGN BASIS EVENTS

- THE REACTOR CAVITY STRUCTURE FORMING THE CONTAINMENT PRESSURE BOUNDARY HAS BEEN DESIGNED FOR THE LOADING CONDITIONS AND STRESS CRITERIA OF SECTION 3.8.2 OF THE PLANT DESIGN REPORT (PDR).
- ADDITIONALLY, PRIMARY STRENGTH MEMBERS FOR THE PLATFORM HAVE BEEN DESIGNED FOR THE LOADING CONDITIONS AND STRESS CRITERIA OF SECTION 3.12.2 OF THE PDR.
- INCORPORATION OF THE CORE LADLE WITHIN THE LOWER REACTOR CAVITY SHALL NOT COMPROMISE THE STRUCTURAL INTEGRITY OF THE CONTAINMENT PRESSURE BOUNDARY OR PLATFORM FOR OPERATING AND DESIGN BASIS CONDITIONS INCLUDED IN THE PDR.
- THE LADLE SHALL BE DESIGNED TO REMAIN FUNCTIONAL (I.E. REMAIN INTACT) FOR ALL NORMAL AND OPERATING BASIS LOADING CONDITIONS DURING PLANT OPERATION AND SHUTDOWN, INCLUDING TRANSIENT LOADS DURING TOW.
- DESIGN FOR THE HORIZONTAL COMPONENTS OF THE OPERATING BASIS EARTHQUAKE WILL BE BASED ON STATIC ANALYSES OF THE CORE LADLE AND ITS SUPPORT.
- DESIGN FOR THE VERTICAL COMPONENT WILL BE BASED ON A DYNAMIC ANALYSIS OF THE CORE LADLE AND THE SUPPORTING GIRDERS.
- HORIZONTAL FORCES WILL BE CARRIED TO THE BOTTOM OF THE LADLE SHELL BY FRICTION FORCES BETWEEN THE BRICKS AND BY THE STEEL SHELL ENCASING THE BRICKS WHICH WILL BE ANCHORED TO THE REACTOR CAVITY FLOOR.
- THE LADLE SHALL ALSO BE EVALUATED FOR DESIGN BASIS ENVIRONMENTAL LOADS IN ORDER TO ENSURE THAT THERE WILL BE NO GROSS FAILURE WHICH MIGHT IMPAIR THE FUNCTION OF THE SAFETY CLASS COMPONENTS.

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Dotson

C. LOADING CONDITIONS FOR REACTOR CAVITY STRUCTURE AND CORE LADLE SUBSEQUENT TO POSTULATED REACTOR VESSEL MELT-THROUGH

- THE REACTOR CAVITY STRUCTURE SHALL BE DESIGNED TO WITHSTAND, FOR THE DURATION OF CORE DEBRIS RETENTION, THE LOADING CONDITIONS WHICH RESULT FROM A CORE MELT ACCIDENT IN COMBINATION WITH THE LOADINGS FROM CONTINUOUS BASIS ENVIRONMENTAL CONDITIONS.

  THE LOADING CONDITIONS RESULTING FROM REACTOR VESSEL MELT-THROUGH ARE:

  1. IMPACT LOADS FROM FALLING DEBRIS
  2. DEAD LOADS FROM WEIGHT OF DEBRIS
  3. THERMAL LOADS RESULTING FROM INCREASED TEMPERATURES

- FOLLOWING REACTOR VESSEL MELT-THROUGH AND DURING THE DEBRIS RETENTION PERIOD, THE REACTOR VESSEL CAVITY SHALL REMAIN INTACT TO THE EXTENT REQUIRED TO SUPPORT THE CORE LADLE.

- THE LADLE SHALL BE DESIGNED TO REMAIN FUNCTIONAL, FOR THE DURATION OF CORE DEBRIS RETENTION, FOR LOADING CONDITIONS WHICH RESULT FROM A CORE MELT ACCIDENT IN COMBINATION WITH THE LOADINGS FROM CONTINUOUS BASIS ENVIRONMENTAL CONDITIONS.

- THE LOADING CONDITIONS RESULTING FROM THE CORE MELT EVENT ARE: THERMAL AND MECHANICAL SHOCK FROM FALLING DEBRIS, ADDITIONAL WEIGHT OF MOLTEN DEBRIS ON LADLE BED AND THERMAL LOADS FROM MOLTEN DEBRIS.

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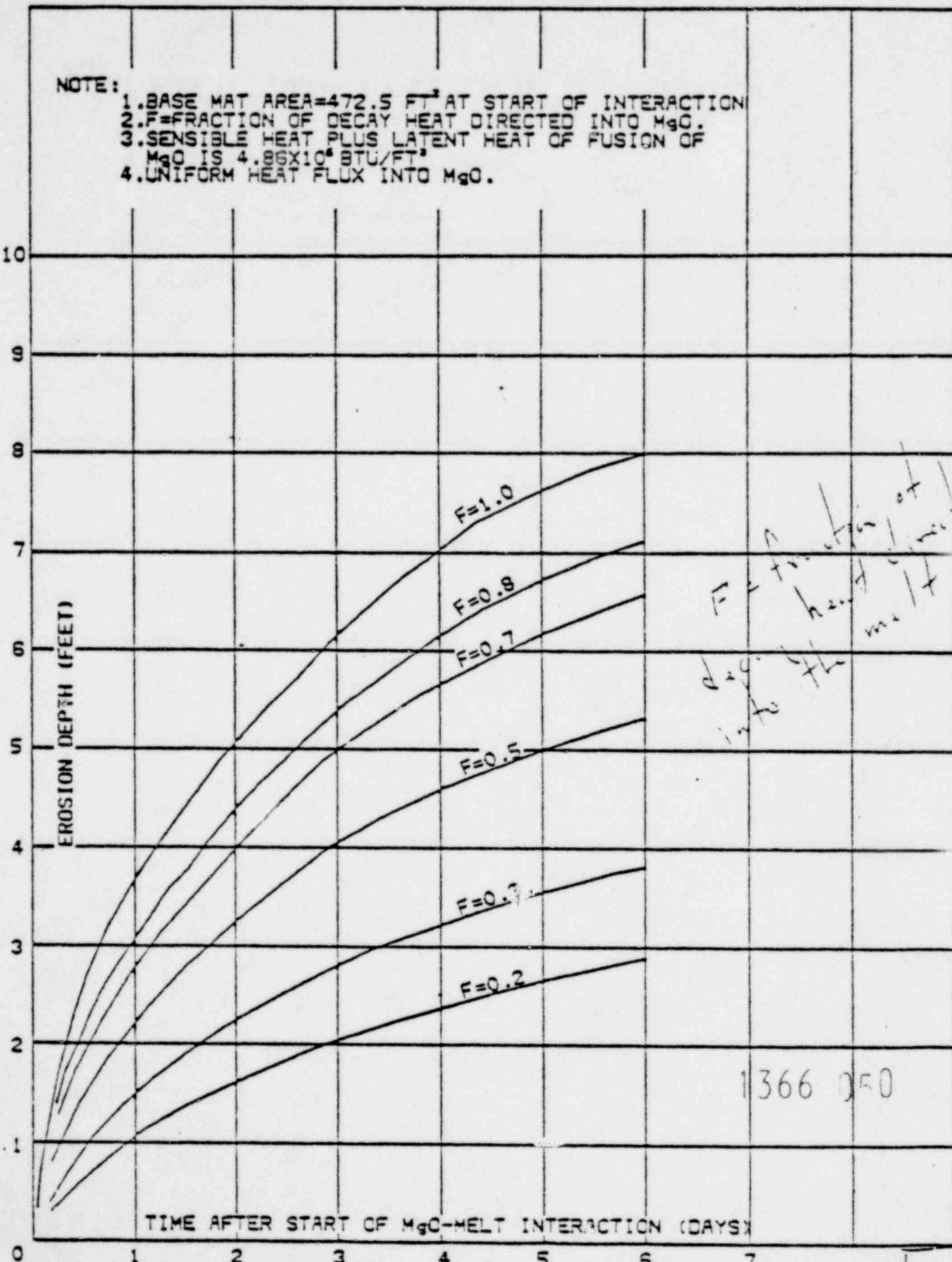
Dotson

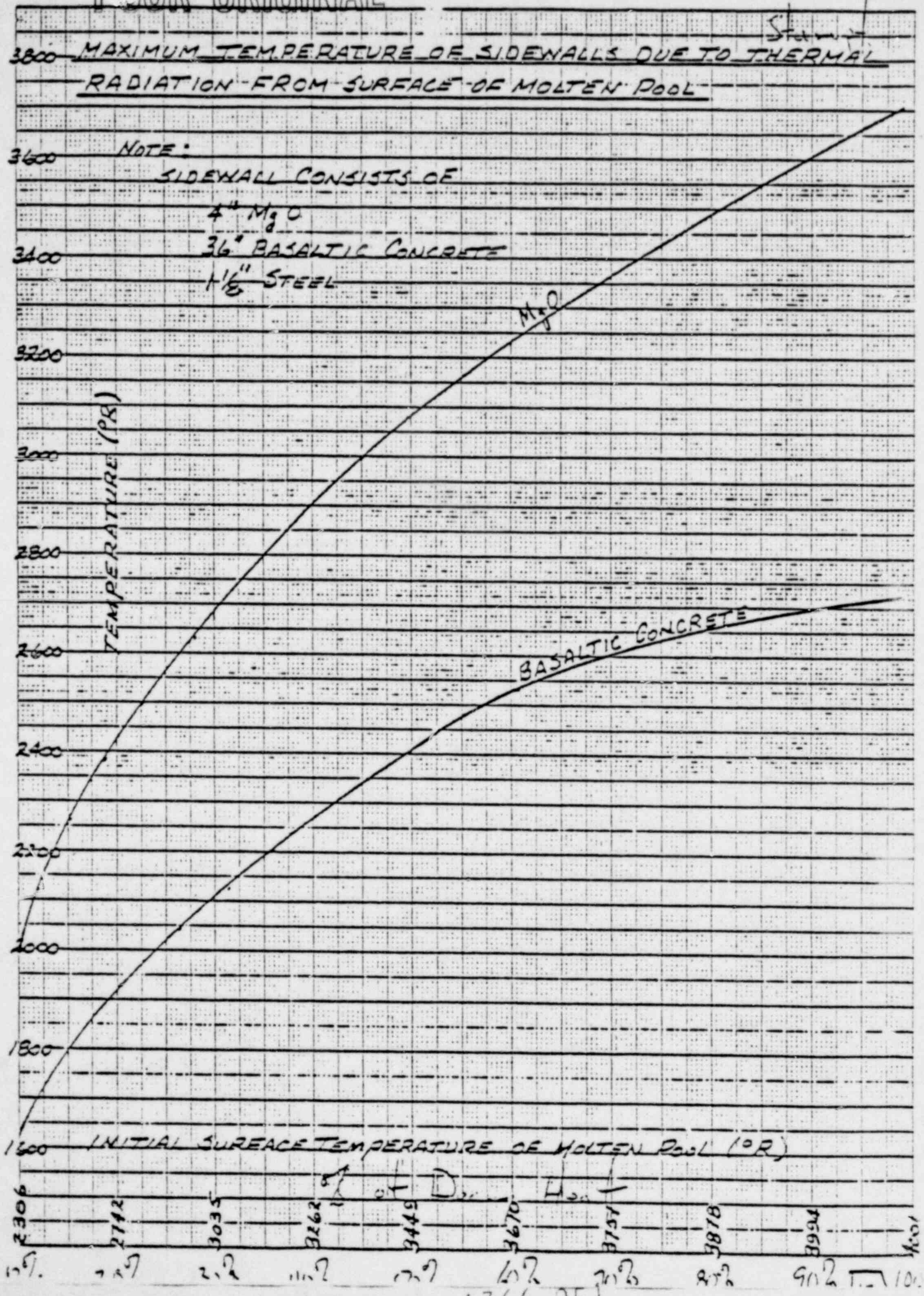
POOR ORIGINAL

EROSION DEPTH AS A FUNCTION OF TIME
START OF MgO-MELT INTERACTION

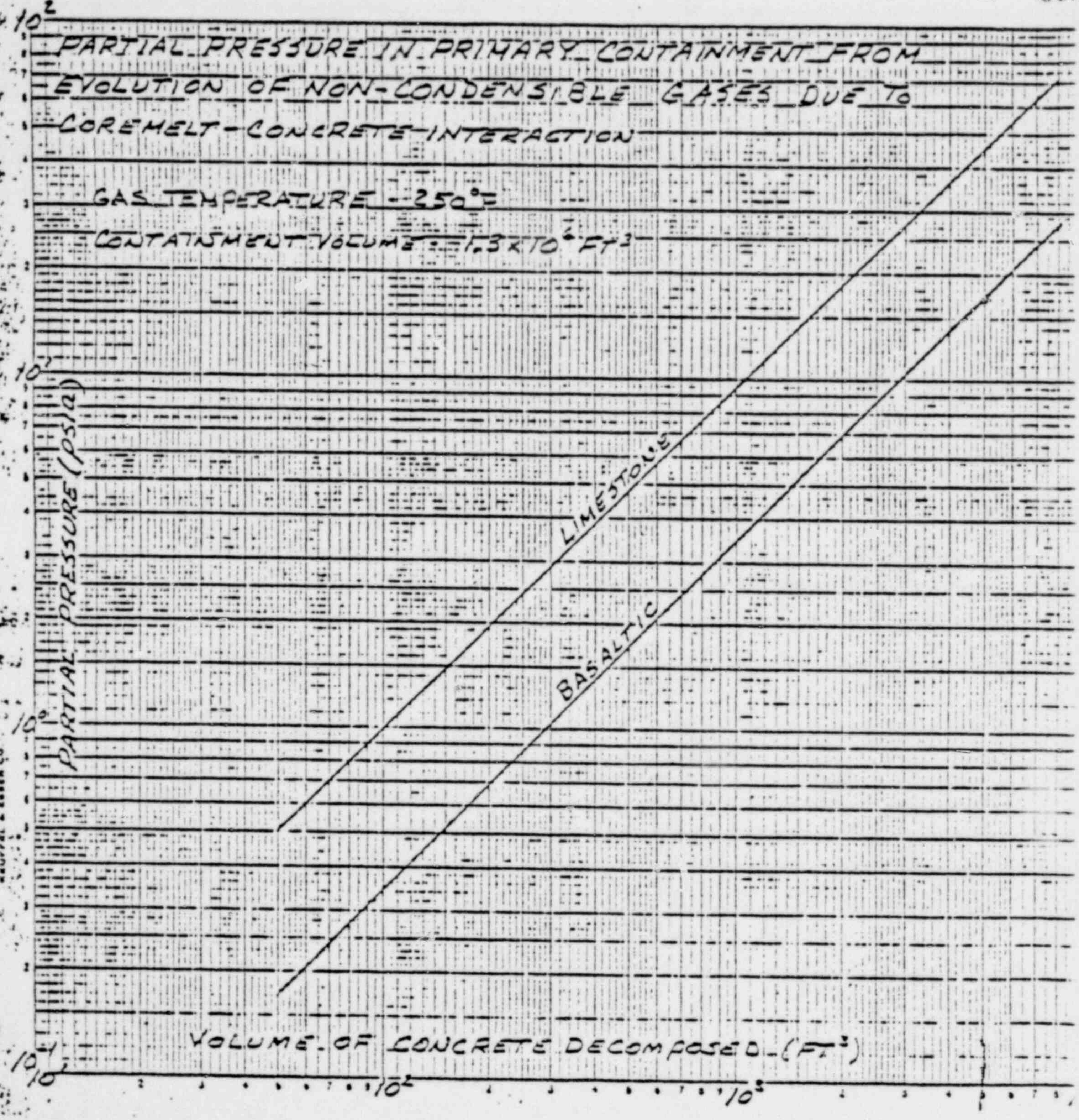


1366 050

I16


MAXIMUM TEMPERATURE OF SIDEWALLS DUE TO THERMAL RADIATION FROM SURFACE OF MOLTEN POOL
NOTE:
SIDEWALL CONSISTS OF
4" MgO
36" BASALTIC CONCRETE
1 1/8" STEEL
MgO
BASALTIC CONCRETE
TEMPERATURE (°R)
INITIAL SURFACE TEMPERATURE OF MOLTEN POOL (°R)
3800
3600
3400
3200
3000
2800
2600
2400
2200
2000
1800
1600
2306
2742
3035
3262
3449
3670
3757
3878
3994

5000 c-ft total decompose   Stumpf



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