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Docket 50-320

EVALUATION OF LONG-TERM POST-ACCIDENT CORE COOLING OF THREE MILE ISLAND UNIT 2

NRC Staff Report

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Division of Systems Safety
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NRC STAFF REPORT - TMI -2

Achieving and Maintaining Long Term Core Cooling

1. INTRODUCTION AND SUMMARY

1.1 Introduction

This NRC staff report addresses acceptability of the proposed method for long term core cooling of Three Mile Island, Unit 2. The licensee has proposed* to adopt the B&W recommendation to utilize natural circulation core cooling in accordance with the "Base Case Summary," see Figure 1.1. Other alternative methods of cooling are available, including high and low pressure injection or recirculation. They do not offer the same assurance of reliability and fission product containment as the proposed natural recirculation mode of long term cooling. The present method of cooling with one reactor coolant pump running is also preferred over the other alternatives, but it has the uncertainty of eventual pump and instrument degradation by the environment inside of containment.

The proposed mode of long term cooling involves a sequence of events initiated in early April following initial actions to stabilize the reactor after the accident on March 28. This sequence is designed to place both steam generator secondary cooling systems and the reactor coolant system in a water solid condition for a closed cycle cooling mode, thus keeping the highly radioactive primary coolant inside containment while preparations for plant decontamination are completed. For this preferred mode of operation, B&W has recommended that a feedwater flow of about 5000 gpm per steam generator be provided. Heat removal to the ultimate heat sink will be via intermediate heat exchangers,

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* "Safety Analysis Report for Transition to Natural Circulation," transmitted by letter from J. G. Herbein of Metropolitan Edison Company to Denwood Ross of NRC, April 12, 1979.

including the condenser for steam generator A in the first few months. These conditions on the secondary side of the steam generators are proposed to optimize primary side core flow, core temperature rise, and average reactor coolant temperature and to result in adequate natural circulation cooling of the core.

The staff has evaluated the proposed "Base Case Summary" plan and has considered contingency plans proposed by the licensee. Our evaluation has included consideration of various subjects, summarized below in this Section and discussed at greater length in later Sections, which might affect natural circulation cooling capability. The staff approves the licensee's plans with certain conditions as described in this report and finds that there is no undue risk to public health and safety in the preferred mode of long term cooling.

On the basis of current understanding of the accident scenario and available data, the staff reports here on its evaluation of the condition of the core and the core flow resistance as it might affect ability to cool the core by natural circulation. The natural circulation cooling capability of TMI-2 for the estimated core flow resistance and a variety of other conditions is evaluated and a comparison of the Base Case and off-nominal plant configurations is presented. The potential for and effects of natural convection core cooling are addressed, and the staff recommendations for reactor performance acceptance criteria upon initiation of natural convection are presented.

Also, a discussion of the short and long term potential for evolution of noncondensible gas is included in this report. Based on current information, it is not expected that quantities of gas large enough to affect natural circulation cooling capability will form at the expected operating conditions, as explained in this report.

The staff has also addressed other potential questions regarding long term cooling in any mode such as boron precipitation, and boron dilution and recriticality, and has reviewed the proposed contingency alternatives in the event of a loss of natural circulation cooling.

The planned hardware modifications in the TMI-2 balance of plant for implementing the long term natural circulation cooling mode have been evaluated day be day by a team of NRC engineers working closely at the site with the Licensee and its contractors. The staff has also considered the process and diagnostic instrumentation requirements associated with operation in the natural circulation mode. Attachment 2 summarizes the results of this review effort.

1.2 Summary

The staff has performed a safety evaluation of the transition to natural circulation shutdown cooling of the TMI-2 reactor core. Our evaluation consisted of a review of the Met Ed/GPU SAR submittal of April 12, 1979, several telephone conversations with B&W technical personnel to clarify the technical content of that SAR, and extensive independent calculations by the staff, national laboratories (PNL, ORNL, INEL, Sandia) and others. We have concluded that there is a high probability that natural circulation cooling of the TMI-2 core can be accomplished using either one or both steam generators in either the steaming or water solid modes. Criteria to accomplish the transition and to evaluate the acceptability of natural circulation cooling performance using only instrumentation which is expected to remain functional for the long term have been defined. Alternative cooling modes have also been considered for the unlikely event that natural circulation cooling fails.

The staff evaluation has considered core conditions ranging from a normal unblocked core with an average resistance factor (K) of 9.4 to a core highly blocked (>99%) in the central region by approximately five feet of debris consisting of fuel and zirconium oxide fragments and with 90% blockage in the peripheral regions, which gives an equivalent average core K of 3760 (96% blockage) or 400 times normal.

Our best estimate of the core resistance model after considering both flow and thermocouple data can be represented by an average core K of 1810 (93% blockage) or 200 times normal. Predicted core flows for this range of core resistances for the current one pump operating condition with one steam generator solid and for the minimum natural circulation condition with one steam generator solid follows.

Average Core Resistance	One Pump (lbs/sec) Core Flow	*Natural Circulation Flow (lbs/sec)
Normal	11,060	380
Best Estimate (200 X Normal)	3,610	231
Maximum (400 X Normal)	3,145†	200**

The calculated natural circulation flow rates as a function of core resistance (K) are linear on a log-log plot. The minimum natural circulation flow rate of 200 lbs/sec corresponds to an average core temperature rise of 13.5F at 3 Mwt decay heat level.

It should be noted that the ratio of calculated one pump flow rate to the calculated natural circulation flow rate for a high core resistance is approximately 15.7:1 and independent of the high core resistance considered.

The core ΔT indicated by individual incore thermocouples during late April remained approximately constant after a large drop in the core inlet temperature (234F to 175F). Thus, a closed channel enthalpy rise model provides a reasonable basis for predicting the incore thermocouple temperatures. Assuming no change in core flow distribution

* Based on Core Inlet Temperature = 103°F and 3 Mwt Decay Heat Level
** Extrapolated from Table 3.3
† Extrapolated from Figure 2.9

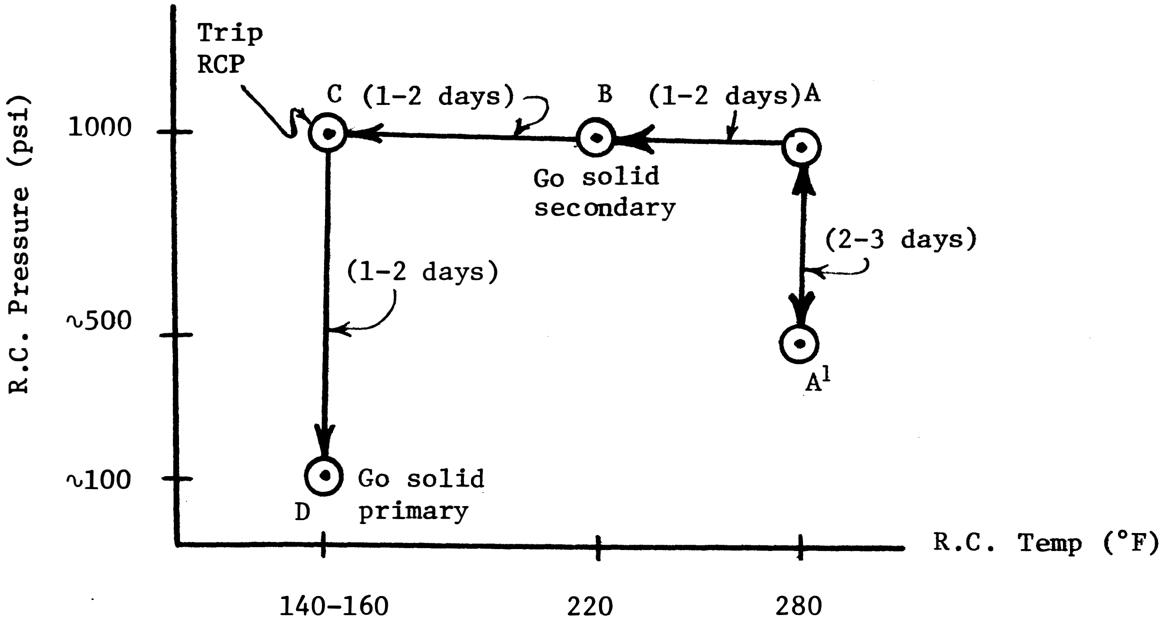
when the operating pump is tripped, the enthalpy rise in each thermocouple channel can be expected to increase by a factor of 15, corresponding to the decrease in flow. Predicted equilibrium thermocouple temperatures as a function of decay heat rate on the day the pump trips are given in Figures 1.2 and 1.3.

The staff has estimated that an incore thermocouple reading of 1000F is indicative of some core material at 1300F or higher. Due to poor strength properties of core structural materials at this high temperature and due to an increasing propensity for additional oxidation of zirconium, 1000F on the incore thermocouples is considered to be an important limit for effective core cooling. Since up to two thermocouples are expected to approach this limit in the event of early initiation of natural circulation, appropriate precautionary actions are specified in Table 4.3 for the occurrence of two or three incore temperatures above 1000F.

A second important limit for acceptable natural circulation cooling is the prevention of bulk boiling in the core. For operation with two steam generators, a criterion to maintain 100F or greater subcooling at the hot leg RTDs is acceptable. For one steam generator operation, low flow or no flow conditions would best be indicated by incore thermocouples. The staff recommends that natural circulation cooling be terminated if the average of incore thermocouple readings exceeds saturation temperature.

Additional criteria are discussed in Section 4.3.

The staff has also concluded that the alternative operating modes of high pressure injection or decay heat removal are less desirable and probably no more efficient than natural circulation cooling. Therefore, natural circulation should not be terminated prematurely.



(1) Degas at A; Lower Pressure (A→A1) while degassing, then return to A.

(2) Continue Design/Installation of static and active systems for primary makeup/pressure control and secondary cooling system for "B" S/G.

(3) Reduce temperature (A→B) by steaming on "A" S/G.

(4) Take "A" S/G solid -(i.e., all liquid, no vapor) drop primary temp. to minimum (B→C)

(5) Trip RC Pump "A" - Establish natural convection - Establish cooling to "B" S/G if available.

(6) Drop primary pressure to selected value (C→D)

(7) Take primary system solid - (i.e., all liquid, no vapor) Control pressure & makeup with static or new active system

<u>END POINT</u>

Primary - Natural Circ, all liquid, Long-term P/V Control
Secondary - All water, Long-term Heat Dump System

Figure 1.1

Core Exit Coolant Temperature
vs.
Time

during Natural Circulation with
one Steam Generator Steaming and
one Steam Generator Isolated

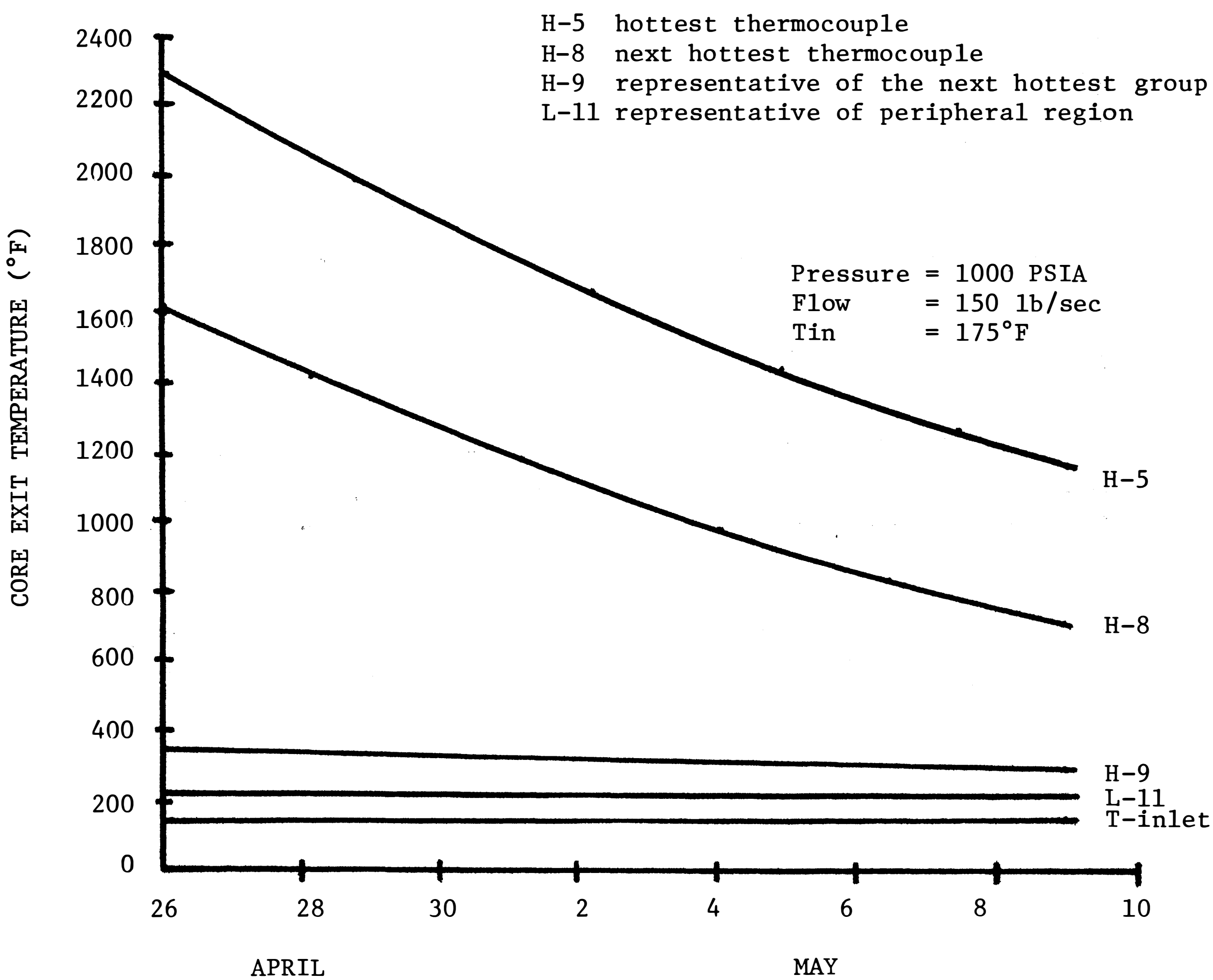


Figure 1.2

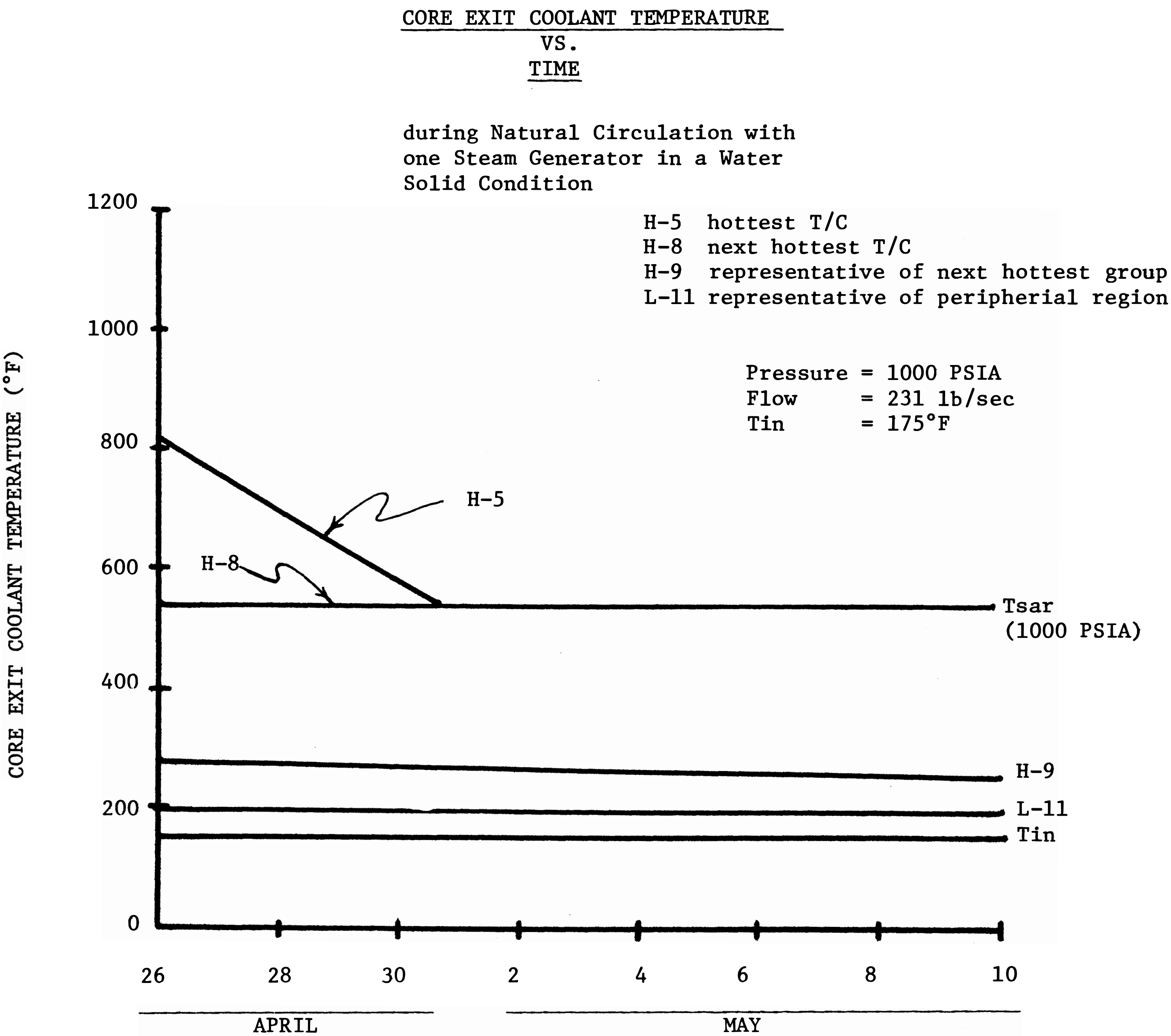

CORE EXIT COOLANT TEMPERATURE
VS.
TIME
during Natural Circulation with
one Steam Generator in a Water
Solid Condition
H-5 hottest T/C
H-8 next hottest T/C
H-9 representative of next hottest group
L-11 representative of peripherial region
Pressure = 1000 PSIA
Flow = 231 lb/sec
Tin = 175°F
CORE EXIT COOLANT TEMPERATURE (°F)
1200
1000
800
600
400
200
0
H-5
H-8
Tsar
(1000 PSIA)
H-9
L-11
Tin
26
28
30
2
4
6
8
10
APRIL
MAY

Figure 1.3

2.0 CORE COOLING

2.1 Assessment of Core Conditions

The staff assessment of the present TMI-2 core geometry is presented in Attachment 1. It is based upon the preliminary data available from the plant as of April 13, 1979. An understanding of core damage is an important factor in assessing the adequacy of core cooling in the proposed natural circulation mode, as explained in subsequent sections. The assessment can be summarized as follows:

The cladding for many or all fuel elements may have ballooned and ruptured early in the accident. This mode of initial defecting is probably irrelevant in light of later, more extensive damage by oxidation and embrittled fracture of many fuel elements.

In the hot upper central region of the core, fuel temperatures probably exceeded 1750°C releasing large quantities of fission products; radiochemical analyses indicate that about 30% of the total core inventory of noble gases was released into the primary coolant system. The 1750°C temperature is less than the UO_2 melting point. Temperatures sufficient to cause fuel pellet melting (2800°C) were probably not reached.

It is estimated that about 40% of the Zircaloy cladding reacted with water. The region of most severe oxidation probably was localized above the 2 to 6 ft elevation, with more severe oxidation in the central bundles than in the peripheral bundles. Significant melting of partly oxidized cladding may also have occurred, and this material would have solidified when core cooling was re-established. The severely oxidized (and perhaps fused) cladding probably fragmented upon quenching. The resulting fuel element debris is probably composed of pieces ranging from millimeter size to whole sections of fuel rods.

The temperature of unfueled components would lag the temperature of fuel rods by only about 20°F so that they also should have experienced temperatures above about 1700°C. Consequently, Zircaloy components in the hot region of the core should have oxidized, and components with Inconel, stainless steel, and Ag-In-Cd should have melted. Because of many layers of protective sheathing in the instrument tube the incore thermocouple tubes have survived even in the damaged core region, although the outer sheath of the instrument tube may be badly damaged.

Nearly all of the broken and oxidized fuel element debris should remain trapped in the upper core region because the fuel assembly end fittings at the top of the core have a grillage that would act as a screen. Furthermore, the compaction of fuel debris is limited because the fuel pellets are fabricated with a packing fraction of about 46% and the theoretical maximum packing fraction (for a bed of spherical particles) is only about 63%. It is very likely that fuel debris is also trapped in some mixing cups surrounding the incore thermocouples, contributing to non-uniform thermocouple readings.

2.2 Staff Analysis of Core Cooling

In order to evaluate current core cooling and the planned transition to natural circulation cooling, calculations were performed by the NRC staff, and results of calculations performed at Pacific Northwest Laboratory (PNL), Idaho National Engineering Laboratory (INEL), and Oak Ridge National Laboratory (ORNL) were discussed with the staff. The calculations can be summarized as follows:

(1) The staff performed analyses with the computer code Cobra IV to develop a thermal-hydraulic model of the core based on the core exit coolant temperature measurements obtained from the post-accident in-core thermocouples with one reactor coolant pump running. Pacific Northwest Laboratory performed similar calculations using Cobra IV and ORNL attempted similar calculations using the

SABRE code. The latter calculations are incomplete; however, preliminary results are consistent with PNL and NRC calculations.

(2) Hand calculations have been performed to further develop the core model based on the knowledge obtained from temperature and flow measurements, thermal-hydraulic design data, and COBRA IV calculations. Considerations in the modeling included:

(a) an agglomerate of fuel particles packed in the mixing cup surrounding the in-core instrument thimble at the axial core exit location of the thermocouple.

(b) suspension of fuel particles in a partially fluidized bed with the present one-pump flow conditions,

(c) accumulation of fuel debris in the bottom of the reactor vessel.

(3) Analyses are in progress to estimate the kinetics of core damage by modeling the events that occurred in the course of the accident. This effort includes hand calculations by the staff to obtain preliminary estimates based on the transient data and more detailed computer calculations using IRT at Brookhaven National Laboratory (BNL) and RELAP4 at INEL.

(4) Calculations were performed at INEL using RELAP4 to evaluate natural circulation flow with several possible modes of steam generator operation during long term cooling. Natural circulation test data from Oconee were studied to confirm the calculated flow rates.

(5) Calculations of the effect of core flow blockage on water flow in the active and inactive loop have been performed by INEL (using RELAP), Technology for Energy Corporation (TEC) and by the staff.

These calculations have been compared with post accident and startup test measurements of flow in the TMI-2 reactor coolant system in order to estimate the extent of core flow blockage.

(6) The time history of post-accident thermocouple data from TMI-2 has been examined using a simulation of the damaged core to evaluate the capability of the model to predict cooling conditions. The behavior of thermocouple data during the transfer from reactor coolant pump 1A to pump 2A on April 7 has been interpreted by the staff to generally confirm the estimated condition of loose fuel particles suspended in the upper regions of the core. As shown in Figure 2.1, high thermocouple readings existed in Region A prior to tripping pump 1A. After start-up of pump 2A, an improved flow condition around the core exit thermocouples of Region A was indicated by decreases of greater than 100F in exit temperature from several of the instrumented assemblies. At the same time, core exit temperatures rose on the order of 50F in several symmetrically located fuel assemblies in Region B of Figure 2.1. This cannot be totally explained if the fuel in the upper region of the core were stationary since the switch in inlet flow distribution would account for a maximum of about 30 percent increase or decrease in core ΔT. However, it can be partially explained by collapse of the fluidized bed during the flow coast down transient and by a redistribution of fuel particles axially or radially for the new flow distribution obtained with Pump 2A. The trends of the thermocouple data both prior and subsequent to the pump trip also are consistent with this conceptual core model.

(7) Calculations were performed to determine if the few high thermocouple readings that have existed since stable core cooling was reestablished could be caused by debris in the mixing cup surrounding the thermocouple. For calculational purposes, a spherical mass was assumed to form around the end of the thermocouple. The

calculations show that even for the maximum diameter sphere which fits into the mixing cup, the temperature difference from inside to outside the sphere is only about 60°F. This is not enough to explain the high thermocouple readings that have been observed during the first two weeks following the accident, assuming heat transfer to subcooled fluid surrounding the mixing cup. Assumptions used in the calculation are given in Table 2.2.

2.3 Discussion of Staff's Thermal Hydraulic Model

The staff's thermal-hydraulic model consists of a one-eighth core, coarse mesh COBRA IV model with one radial mesh point per assembly and 13 axial nodes. Flow area reductions were included in the assemblies which showed the largest temperature difference across the core. The amount of area reduction was chosen so that the calculated core exit coolant temperature matched the thermocouple readings taken at 5 PM on 3/30/79. The measured thermocouple readings were corrected for the bias which was observed at the plant (see Section 2.4, below). When the thermocouple readings are corrected for such bias, the core temperature rise in the peripheral assemblies appears to be approximately consistent with a relatively unblocked core. The measurement uncertainties for the thermocouples and the resistence temperature detectors in the coolant loops (RTDs) are so large that the temperature differences between a normal core and a slightly blocked core would be masked.

The staff model takes account of increased resistance to cross-flow by greatly reducing the cross-flow areas. The extent of flow area blockage in the staff model is shown in Figure 2.2. This model has been used to calculate the core exit coolant temperatures for the period 3/30/79 to 4/10/79 and has also been used to calculate the coolant conditions under several possible modes of operation for natural circulation. Figures 2.3 and 2.4 present the calculated and measured core exit coolant temperatures as a function of time.

The staff's consultants at Pacific Northwest Laboratory also performed thermal-hydraulic calculations for the damaged TMI-2 core. The PNL thermal hydraulic model consists of a coarse mesh, full core model with 21 radial regions each representing eight or nine fuel assemblies. The PNL radial modeling is shown in Figure 2.5. The model includes increased flow resistance and reduced flow area in the hot regions at high elevations, as shown in Figure 2.6. As was the case with the staff calculations, it was necessary for PNL to severely restrict cross flow in order to match the measured core exit coolant temperatures. The area changes and flow resistances shown in Figure 2.6 were chosen to match the thermocouple readings from 4/7/79.

Both the staff model and the PNL model indicate that extensive blockage to both axial and radial flow is required to explain the measured temperatures. In order to infer a core condition from the area reductions and resistance increases used in the models, we have used the flow and pressure drop characteristics of various possible flow blockage configurations.

For example, calculations were performed to estimate what type of particles might be suspended in the core due to upflow from one reactor coolant pump and to estimate the depth of particulate matter required to cause the inferred flow blockage (i.e., blockage inferred from readings of thermocouples above the core).

The average velocity of fluid in the core is estimated to be about 3.5 ft/sec with one pump running and with a normal core geometry. As can be seen from Figure 2.7, this velocity is high enough to suspend particles of 0.15 inch, or less, in equivalent diameter. Fuel pellet cracking, as is known to occur during normal operation, would produce UO_2 fragments in a size range that could be be suspended by a 3.5 ft/sec upflow. As fabricated, the pellets are about 0.37 inches in diameter and approximately 0.5 inches long.

The pressure drop required to fluidize a particulate bed of UO_2 ranges from 2.5 psi/ft for 40% porosity to 2 psi/ft for 50% porosity. Therefore, if the frictional core pressure drop is between one and twelve psid, a fluidized bed between one-half foot and six foot thick could be suspended.

The hydraulic resistance of a fluidized bed depends on the porosity of the bed and the shape of the particles. Therefore a range of shapes and porosities were considered in calculating the resistance coefficient for the fluidized bed. The porosity was considered to vary from 40% to 50% and shapes including cylindrical sectors, wedges and parallelopipeds were considered. If the pressure drop per foot of bed (psi/ft) is expressed as

$$\frac{\Delta P}{L} = \frac{K}{A^2} \frac{W^2}{288\, g_c\, \delta}$$

Where W = mass flow, lb/sec

A = area occupied by bed of particles, ft^2

δ = fluid density, lb/ft^3

g_c = gravitional constant, ft/sec^2

K = resistance coefficient, $^1/ft$

L = bed length, ft

then the most likely range for the resistance coefficient, K is $37{,}000 \leq K \leq 138{,}000$ for each 12 inches of debris.

In summary, we found that a flow resistance (K-factor) of 37,000 to 138,000 could be produced by each 12 inches of debris, where the debris consists of fuel and clad fragments ranging down to 1/8 of an inch in diameter. The flow resistances used by PNL (600,000 and 100,000) would be equivalent to at least 4.3 feet of debris for the hottest region and between 2.7 feet and 0.7 feet of debris in the lower temperature regions. The peripheral region is assumed to be free of a significant amount of debris in the PNL model. The staff model was used to study peripheral regions with two possible configurations, unblocked and 90% blocked. Using a blockage of 90% in the peripheral region, the total core flow resistance as inferred from the thermocouple readings is 3760 or 406 times normal.

In order to calculate the core exit temperature readings following the change to pump 2A on April 7, it was necessary to change the flow blockage model. Some or all of the observed temperature changes may have been associated with axial motion of debris; that is, material that was near the thermocouples (e.g., just under the fuel bundle upper end fitting) might have settled to a lower elevation. If this had been the case, then the temperature of the material might not have been affected when pump 1A tripped; rather, the temperature at the thermocouple location might have been lowered as a result of cross flow from cooler regions. For this reason both the early blockage model (with pump 1A running) and the later blockage model (with pump 2A running) have been used in extrapolating to natural circulation conditions, as described in Section 4, below.

Figure 2.8 shows a comparison between the temperature behavior predicted by the later blockage model and the measured thermocouple readings since the startup of pump 2A. The figure clearly indicates periods of time during which the readings are changing more rapidly than would be expected from the changes in decay heat with time. This could be the result of continuing movement of debris in the region of the hot assembly (H-8). However, there appears to be a smooth and steady decrease in the readings from 1600 hours on 4/9/79 until 0800 hours on 4/11/79. This reduction parallels the calculated results and suggests at least a temporary end to the movement of material near the exit of assembly H-8. This trend has continued up to the present time.

2.4 B&W Thermal - Hydraulic Analysis*

A conservative evaluation of core blockage was made by B&W, based on the use of the core exit thermocouples. Based on the measured temperature data, the core flow in the current mode with one reactor coolant pump operating was estimated to be less than 1 x 10^6 lbs/hr. This represents very nearly total blockage. The comparable calculations by the staff indicate about 2 x 10^6 lbs/hr.

In order to better understand the apparently high core exit thermocouple readings, B&W performed analyses of the thermocouple data. The chromel-alumel thermocouples are located in instrument thimbles which extend through the core. The thermocouple junction is within a mixing cup in the fuel assembly upper end fitting, approximately 9 inches above the active core. The analysis postulated the presence of fuel accumulation in the upper end fitting and the mixing cup. Fuel debris with a radius of 3 to 4 inches is needed to produce the highest observed temperature readings in the early days following the accident when core temperatures had stabilized (ΔT = 300F from thermocouple to subcooled coolant external to the instrument thimble). This amount of fuel could fit within the upper end fitting, which has an interior width of seven inches. Debris within the smaller diameter mixing cup could result only in a small (10F) increase in temperature.

Tests were performed by B&W to determine the effect on thermocouple accuracy of exposing the instrument to 2000F for four hours. Four test thermocouples read within 5% of their calibration values over a range of 200F to 1000F after this exposure. Thermocouple data from TMI-2 during normal operation were examined for evidence of a systematic bias. The possibility of de-calibration since the accident was also evaluated. It was concluded from the results of the thermocouple tests and evaluations that the temperature readings are generally accurate, but have a possible +5F bias. Further upward bias of the thermocouple readings is believed by B&W to be due to fuel debris

* See Licensee's SAR of 4/12/79, Section 3.7

packed around the thermocouple. They cite the shift in thermocouple readings during the switch from pump 1A to pump 2A to further support the debris theory.

From these studies, B&W concluded that the existance of flow blockage to reduce the core flow to 1×10^6 lbs/hr, as can be predicted from the thermocouple data, was improbable.

Rather than a blockage model based on the thermocouple data, B&W advances another model of core thermal hydraulics. This model applies the B&W PUMP code to the TMI-2 plant for different reactor coolant pump configurations and corrects for the pre-accident flow split measurements. Core resistance was then increased until the calculation predicted the hot leg, post-accident flows as measured by the two flow meters in the present one-pump operating condition. This calculation results in a prediction of core flow of 13×10^6 lb/hr and a core pressure drop of 18 psi for the one-pump operating condition. These conditions are calculated with an average core flow resistance approximately 200 times the normal resistance, or a form loss coefficient of 1100 for a single-node core representation or 1650 for separate modeling of core and bypass. This model thus yields a more optimistic view of core flow than do the models based on the thermocouple data.

B&W considered the current estimate of material available for core blockage to determine if their best estimate core resistance was feasible. They concluded that 167 cubic feet of debris spread evenly across the core in a three-foot-thick packed bed would produce a form loss coefficient of 1700, which is in good agreement with the total core resistance calculated using the measured flow splits. Therefore, this is considered to be their best estimate of current core conditions.

The Licensee's submittal of the B&W analysis does not address the validity of the post-accident, one-pump flow data tubes. These tubes are located in the hot legs; typical post-accident readings for the one-pump operation are 24% forward flow in the active loop and 17% reverse flow in the inactive loop. The staff has been informed in telephone conversations with B&W engineers that the one-pump flow measurements, including reverse flow, are accurate to within one percent. The validity of this information is of primary importance in evaluating the actual core resistance that now exists.

B&W has calculated an unrecoverable core pressure drop of 18 psid for the present one-pump operation. The pressure drop was determined by adjusting the core resistance in the hybrid computer code PUMP until the calculated flow in the hot leg of the loop A matched the measured flow in the "A" hot leg of TMI-2. The calculation was confirmed by comparing the reverse flow calculated for loop B with the measured reverse flow for loop B. B&W claims that measured and calculated reverse flow agreed very well. Further confirmation of these relatively large loop flows can be obtained by comparing the post-accident power input to the pump with its normal power input. The pump is drawing approximately 10,000 HP or 7.46 Mw with approximately 6.3 Mw of pump heat input to the water. Normal pump heat is 4.5 - 5.0 Mw.

Because B&W's calculated core coolant velocity is low (1.26 ft/sec on the average) most of the core pressure drop must be due to form loss. B&W calculated a dimensionless form loss coefficient, K, of 1100 for the combined core flow and bypass flow, as follows:

$$\Delta P = K \frac{V^2}{288\ \delta\ g_c}$$

where ΔP = core pressure drop in psi

V = core coolant velocity in ft/sec

δ = coolant density in lbm/ft^3

g_c = gravitational constant in ft/sec^2

The measured core pressure drop for four-pump operation under normal conditions is 14.5 psi. Initially, B&W assumed that 50% of the pressure drop was due to friction and 50% due to form loss. They thus calculated a normal form loss coefficient of 5.47 and noted that the form loss coefficient for the damaged core was approximately 200 times the normal form loss coefficient. Other form loss coefficients reported by B&W (1100 and 1650-1700) correspond to the calculated 18 psid pressure drop across the damaged core using different models in the pressure drop calculation. The K=1100 was for a model which lumped the core and bypass regions. The K = 1650 -1700 was for a model whch considered separate core and bypass flow.

B&W provided a parametric study of natural circulation for form loss coefficients ranging from 18 times normal with 5% bypass to 1000 times normal with 5% bypass. The best estimate case of 200 times normal with 30% bypass was also included. They conclude that even with the most pessimistic form loss coefficient of 1000 times normal, the core flow during natural circulation is adequate to cool the core.

The staff has evaluated the core and loop flow analysis by B&W. The measured flow rates in the active and inactive hot legs have been studied in an attempt to infer core blockage on the basis of changes in these values from those expected in an unblocked core. Calculations of loop flow rates with various amounts of core blockage have been performed by the staff, by INEL and by TEC. Figure 2.9 shows the calculated changes in loop flow with increasing core blockage. As discussed above, the measured flow rates in the active and inactive loops are 24% forward flow and 17% reverse flow, respectively. Table 2.1 presents the measured inactive loop flows on each of three reactor protection system channels for pump 1A running and for pump 2A running.

These measured inactive loop reverse flow rates of about 17% were used to infer an amount of core blockage by using the information on Figure 2.9. The core flow blockage corresponding to 17% reverse flow in the inactive loop is approximately 96%. This is in reasonably good agreement with the results of B&W calculations which indicated that a core resistance of 200 times normal is required to produce this value of inactive loop flow; i.e., a core resistance of 200 times normal corresponds to approximately 93% core blockage.

Based on measured core pressure drop of 14.5 psid and a calculated frictional pressure drop of 3.8 psid for normal four-pump operation, the staff has calculated a normal-core form loss coefficient of 9.4. Assuming that the reported TMI-2 core pressure drop of 18 psid is correct and that the core flow is 13×10^6 lb/hr as estimated by B&W, the resistance coefficient is 1810 or approximately 200 times normal. The most optimistic estimate of core conditions is based on the assumption that the peripheral region of the core is unblocked. This assumption gives a core frictional pressure drop of 1.5 psid and a core average flow of 21×10^6 lb/hr; this corresponds to a form loss coefficient of 57.5 or 6.1 times normal. The value of core resistance which is inferred from the thermocouple readings is 3760 or 400 times normal.

The range of core resistance factors for the cases discussed above is given in Table 2.3. This complete range of core resistances was considered in the natural circulation calculations reviewed in Section 3, below.

2.5 Staff Evaluation of Core Cooling

On the basis of the available measurements, our own calculations, our consultants calculations, and B&W's calculations, we have developed the following conclusion. It is reasonably certain that the central region of the core is severely damaged and almost entirely blocked to normal coolant flow. This blockage is probably due to a 3 ft to 6 ft thick layer of fuel and structural debris. The lower elevations in this central region of the core might be relatively undamaged and unblocked. The condition of the core peripheral region is more difficult to evaluate. The core exit thermocouples in the peripheral region indicate temperatures approximately 10°F higher than the indicated cold leg temperature. However, these readings are thought to be slightly biased as described in Section 2.4. Therefore the ΔT measurements in the peripheral region are not sufficiently accurate to distinguish between unblocked and partially blocked (i.e., up to approximately 90% blockage). The hot leg venturi flow meter readings have been used by B&W and by the staff to infer a total core blockage of approximately 95%. This amount of blockage is consistent with the core exit thermocouple readings if the center region is more than 99% blocked and the peripheral region is approximately 90% blocked. This appears to be an intuitively reasonable core description and is consistent with or conservative relative to other core descriptions that have been developed (Attachment 1). We conclude that the core resistance is in the range of 200 times normal to 400 times normal. Since the loop flow measurements are expected to provide better information on the average core resistance, we conclude that the best estimate of core resistance is close to the 200 times normal value as calculated based on the measured loop flows. The value of core resistance inferred from the thermocouple readings (i.e., 400 times normal) has also been considered in the bounding calculations of local core temperatures which might occur during natural circulation.

TABLE 2.1

INACTIVE B LOOP FLOW RATE MEASUREMENTS

April 1, 1979 reactor coolant pump 1A running

Reactor Protective System Channel	Flow Rate
A	-6850 lb/sec
B	---
C	-6889 lb/sec
D	-6794 lb/sec
Average Value	-6844 lb/sec (16.9% of full)

April 9, 1979 reactor coolant pump 2A running

Reactor Protection System Channel	Flow Rate
A	-6870 lb/sec
B	------
C	-6844 lb/sec
D	-6541 lb/sec
Average Value	-6752 lb/sec (16.7% of full)

TABLE 2.2

Assumptions Used in The Calculation

of The Spherical Mass

(1) One dimensional conduction

(2) Decay heat = 0.5% of 2772 Mw

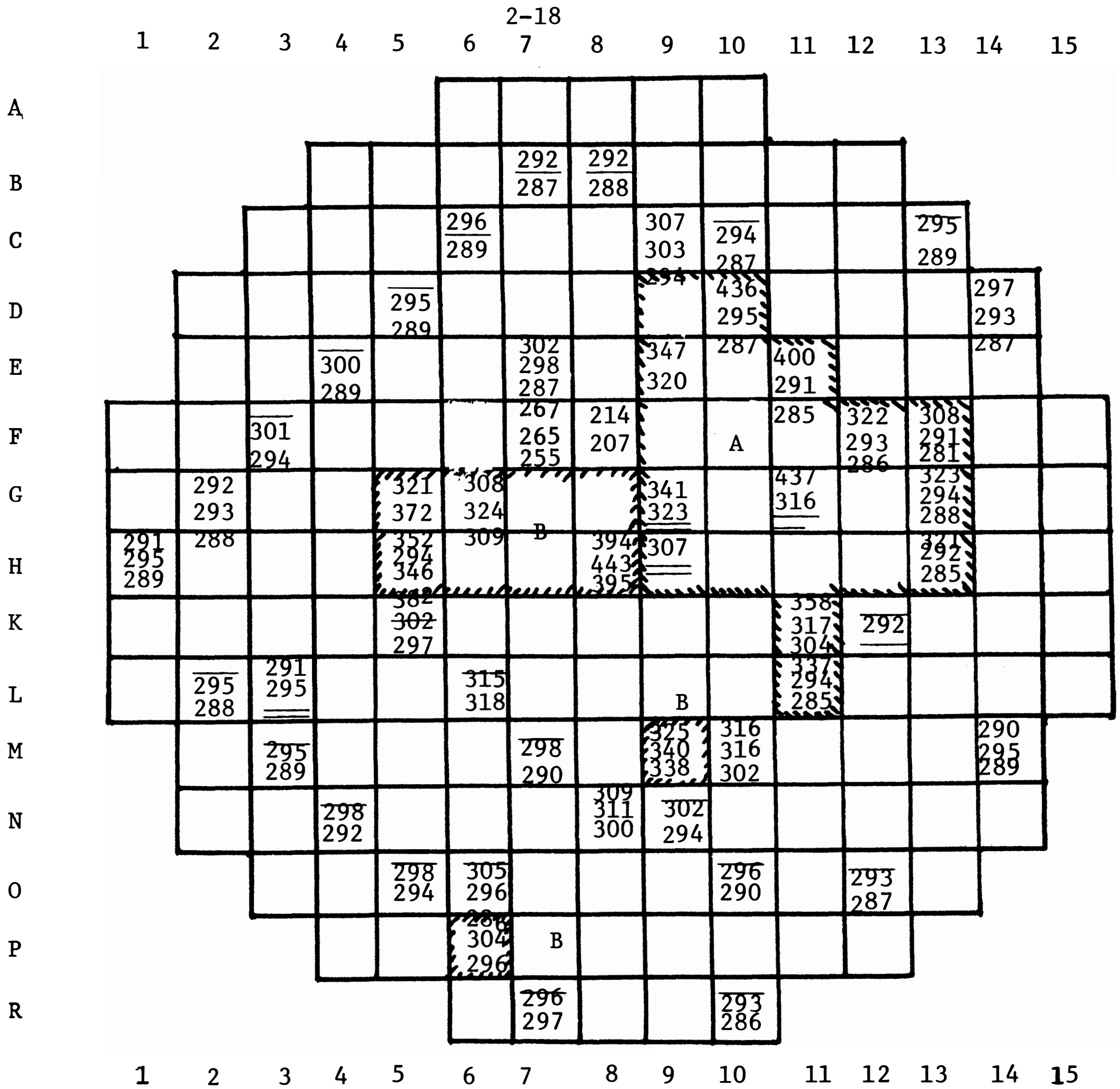
(3) Thermal conductivity of UO_2 = 2 Btu/hr-ft-°F

(4) The ratio of volumetric heat generation to the thermal conductivity does not vary with porosity.

TABLE 2.3

STAFF CALCULATED CORE RESISTANCE FACTORS

Condition	W Total lb/Hrx10^{-6}	ΔP Form (PSID)	$K = \frac{(\Delta P)\ (288) g_c}{\delta \quad V^2}$	W CORE, lb/Hrx10^{-6}	V ft/sec (core area = 49.17 ft^2)
4 Pump (Design)	130.1	10.7	9.4 (normal)	120.2	15.47
1 Pump (Normal)	62.02		9.4 (normal)	39.8	7.38
1 Pump B&W Calculated Core ΔP	39.7	18	1810 (200 X normal)	13.0	1.26
1 Pump 40% Core Blockage	-	1.5	57.5 (6.1 normal)	21	2.04
1 Pump, 95% Blockage from TCs	-	-	3760 (400 X normal)	-	-



XXX XXX XXX	4/06/79 @ 1000 hrs w/PUMP 1A (Tin = 285F) 4/07/79 @ 1205 hrs w/PUMP 2A (Tin = 285F) 4/11/79 @ 0800 hrs w/PUMP 2A (Tin = 280F)

\\\\\\\ A indicates blocked region w/Pump 1A

/////// B indicates blocked regions w/Pump 2A

Figure 2.1

TMI-2 Incore Thermocouple
Indications of Flow Blockage

TMI-2 Flow Blockage Model

Based on T/C Readings from 3/30/79

	8	9	10	11	12	13	14	15
H	90% .87	98% 1.19	99% 1.30	99% 1.31	90% 1.22	95% 1.22	90% 1.32	90% .94
K		99% 1.27	99% 1.30	99% 1.29	96% 1.20	98% 1.22	90% .97	90% .79
L			99% 1.30	98% 1.25	99% 1.03	90% .96	90% .62	90% .49
M				99% 1.15	90% 1.08	90% .93	90% .72	
N					90% .99	90% .85	90% .55	
O						90% .59		
P								
R								

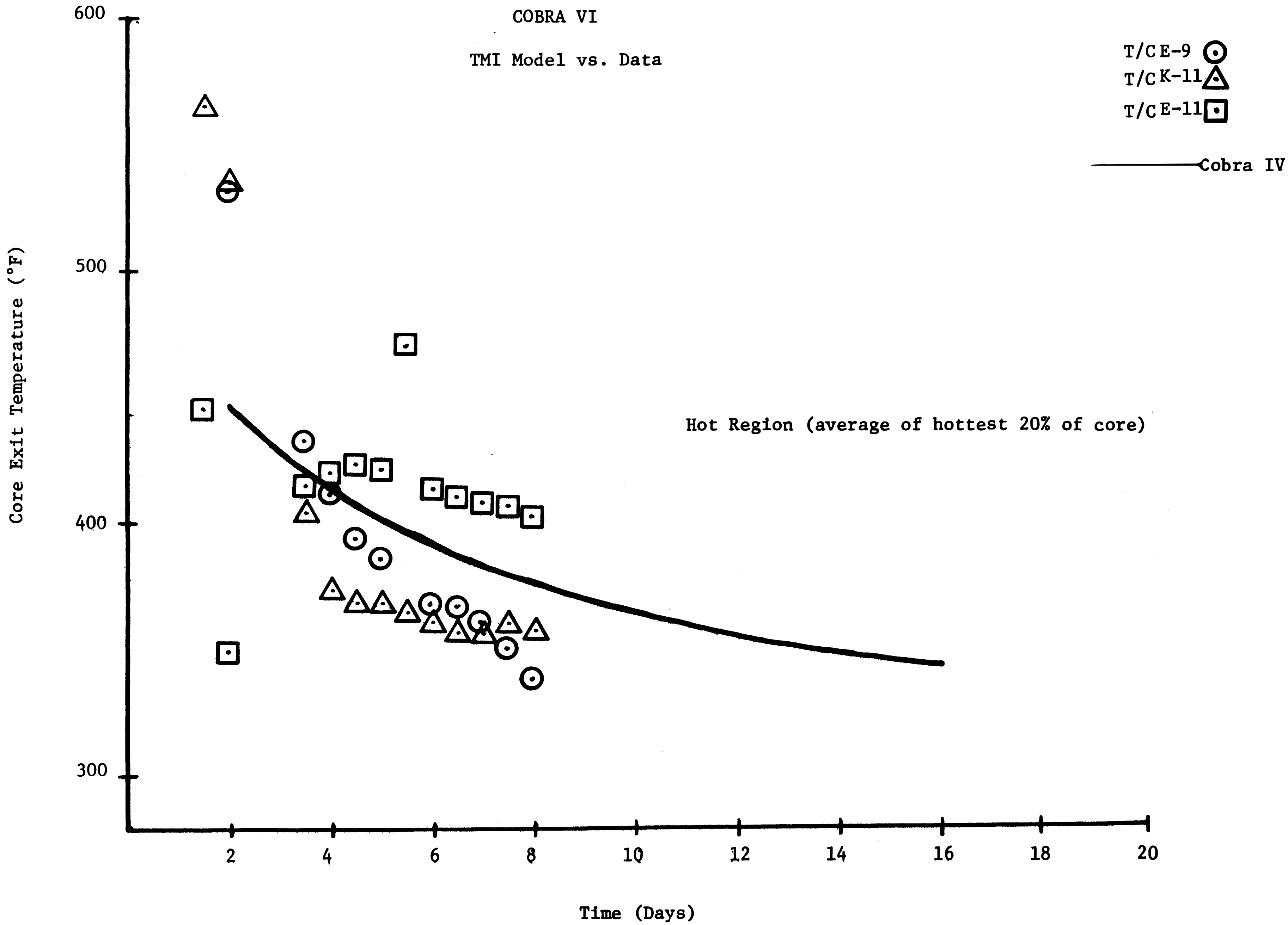
$\boxed{X.XX}$ RADIAL RPD

76 Assemblies Blocked
101 Assemblies Unblocked

CONSUMERS POWER COMPANY
MIDLAND PLANT UNITS 1 & 2

4 EFPD 100-30-100 Design
Transient Power Distribution
at 136 EFPH, 100% FP

Figure 2.2



COBRA VI
TMI Model vs. Data
T/C E-9
T/C K-11
T/C E-11
Cobra IV
Hot Region (average of hottest 20% of core)
Core Exit Temperature (°F)
600
500
400
300
2
4
6
8
10
12
14
16
18
20
Time (Days)

Figure 2.3

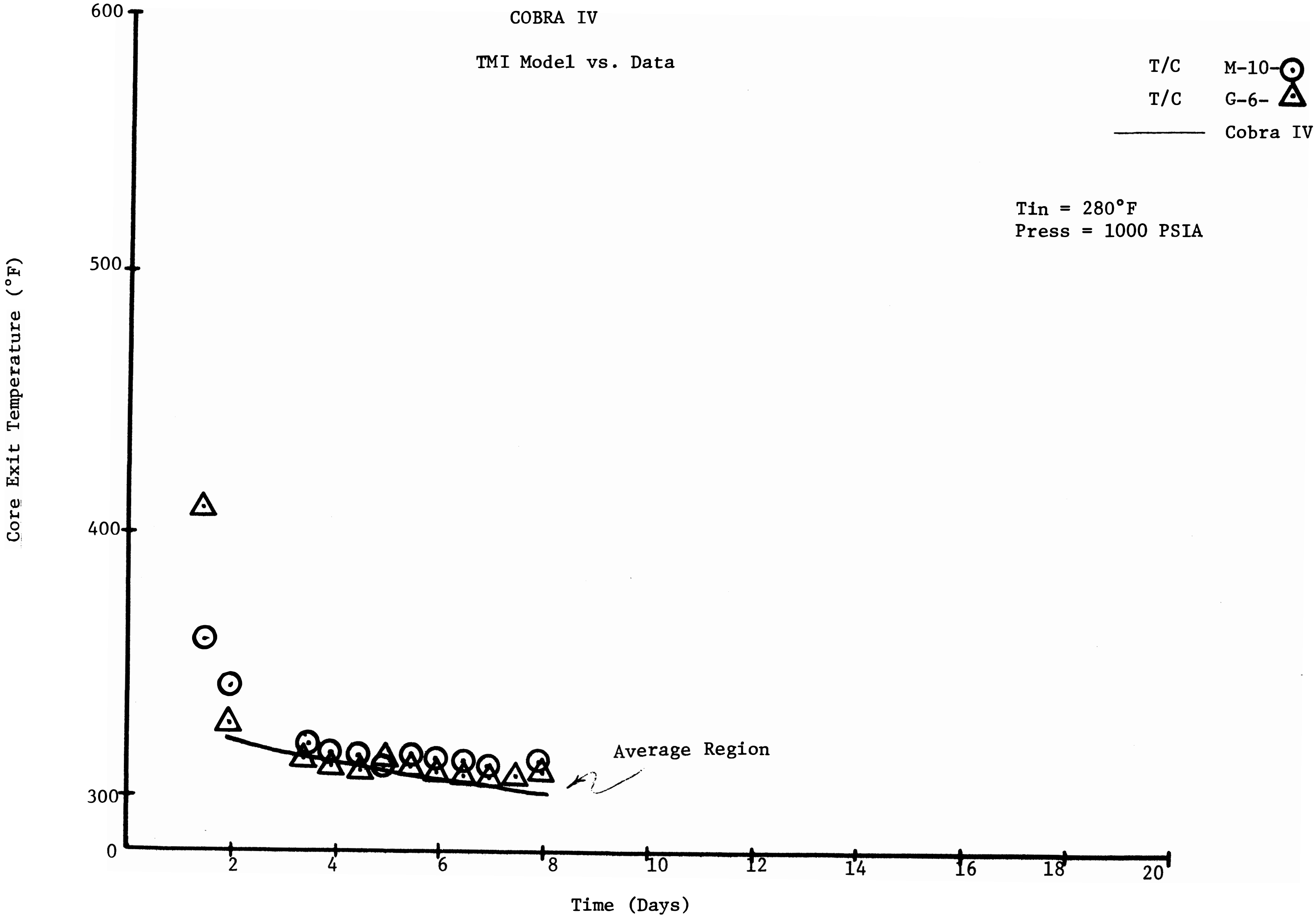
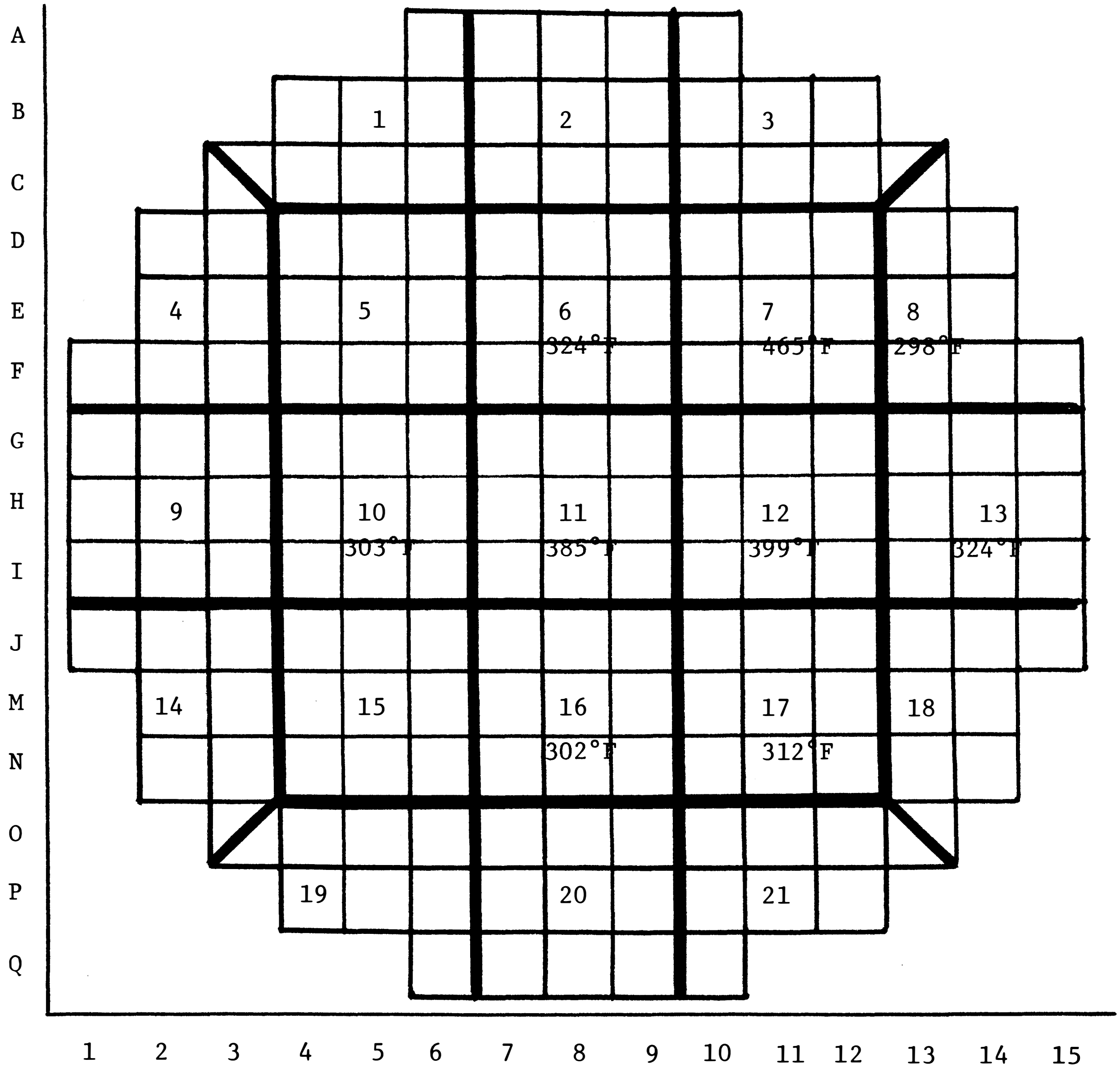

COBRA IV
TMI Model vs. Data
T/C M-10-
T/C G-6-
Cobra IV
Tin = 280°F
Press = 1000 PSIA
Core Exit Temperature (°F)
600
500
400
300
0
2
4
6
8
10
12
14
16
18
20
Average Region
Time (Days)

Figure 2.4



Region I Channels 8, 10, 16, 17

Region II Channels 6, 11

Region III Channels 7, 12

Unblocked Area Channels 1, 2, 3, 4, 5, 9, 14, 15, 18, 19, 20, 21

PNL COBRA IV Model for TMI

Figure 2.5

Flow Resistance and Flow Areas

for

PNL COBRA IV Model

	Region I	Region II	Region III
End Fitting			
Grid 8	K = 500,000 Area = .43 x Nominal	K = 50,000 Area = .43 x Nominal	K = 50,000 Area = .43 x Nominal
Grid 7	K = 50,000 Area = .43 x Nominal	K=25,000 Area = .43 x Nominal	K= 25,000 Area = Nominal
Grid 6	K = 50,000 Area = .43 x Nominal	K = 25,000 Area = Nominal	K = 25,000 Area = Nominal
Grid 5			
Grid 4			
Grid 3			
Grid 2			
Grid 1			
End Fitting			

Region I Gap Size = 10^{-3} in.

Region II Gap Size = .03 in

Region III Gap Size = .03 in.

Figure 2.6

Figure 2.9

Fluid Velocity Required to Suspend Spherical UO_2 Particles in 280°F Water at 1000 PSIA

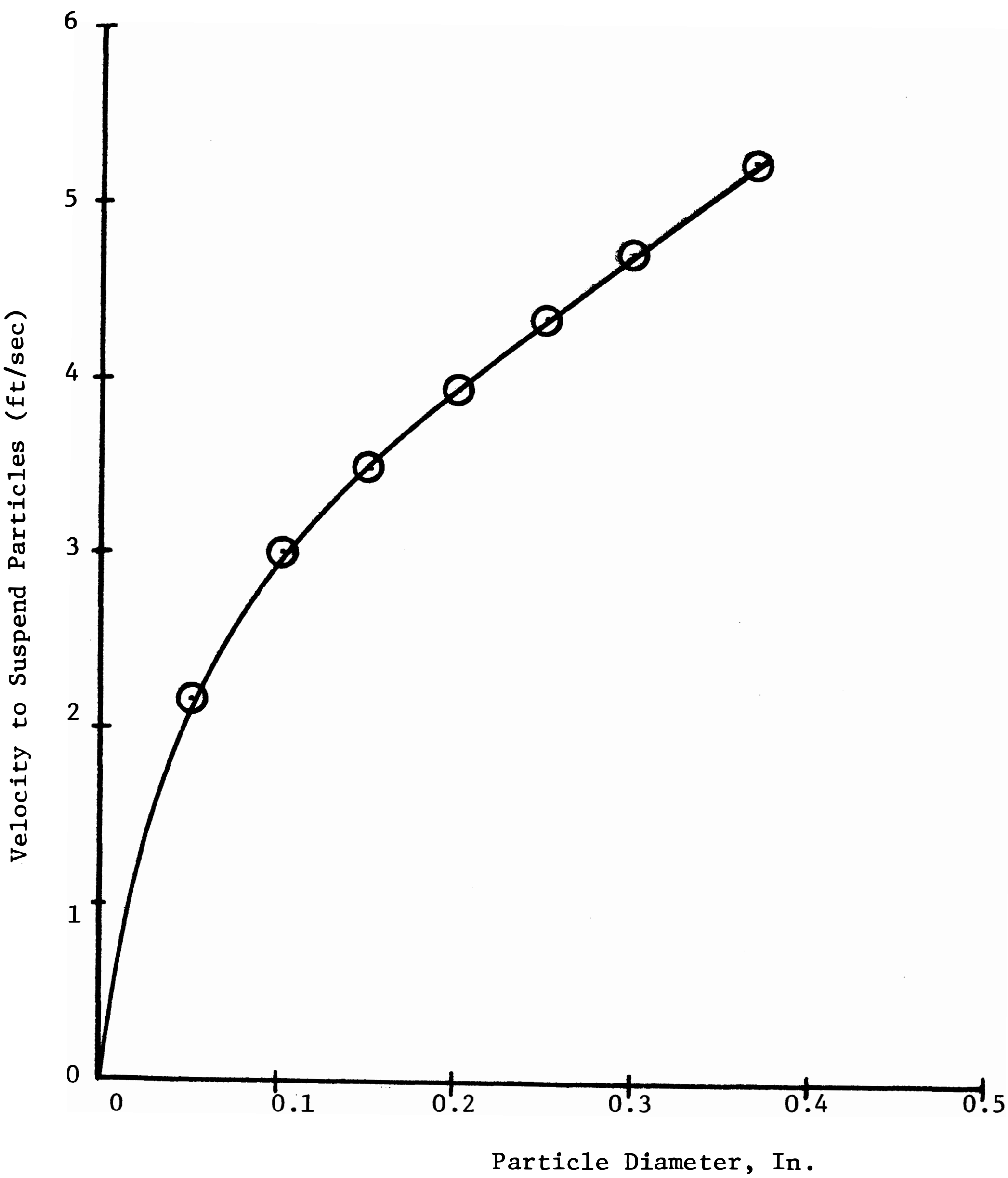


Figure 2.7

Core Exit Coolant Temperatures
vs
Time

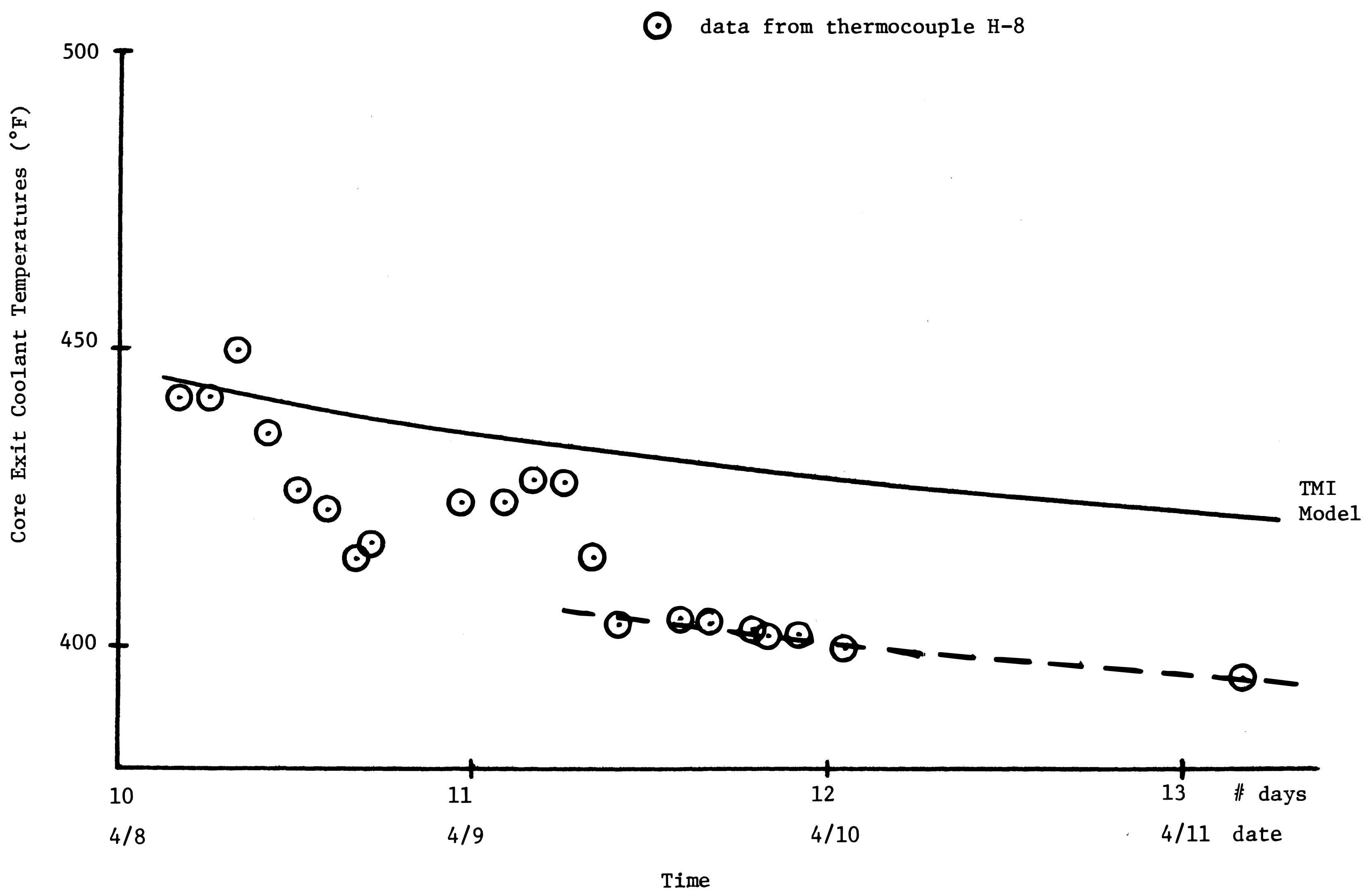


Figure 2.8

Active and Inactive Loop Flow
as a Function of
Core Blockage

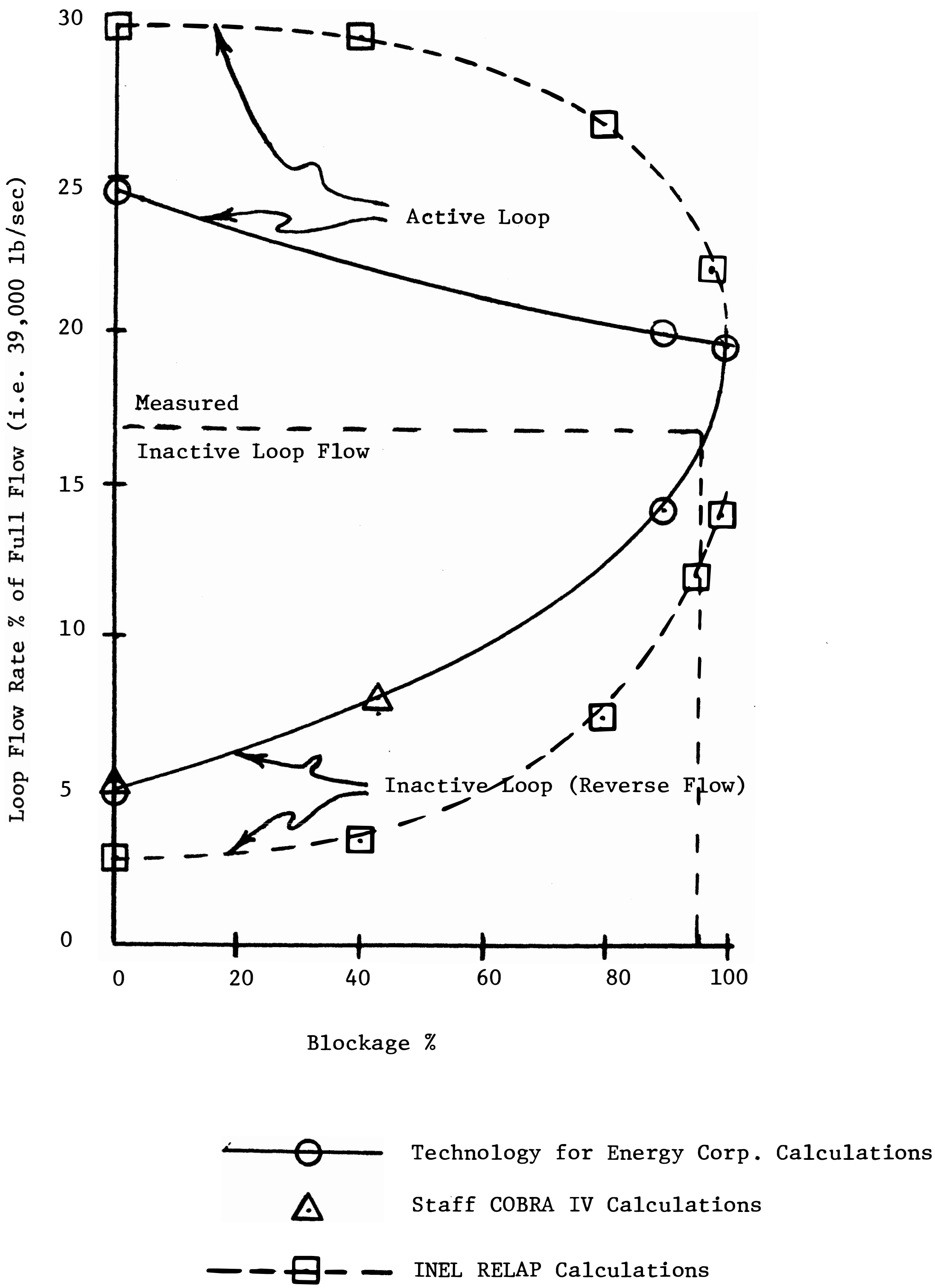


Figure 2.9

3.0 Long Term Natural Circulation

Natural circulation flow rates in the reactor coolant system for post-accident TMI-2 conditions have been calculated by B&W and by the staff's consultants at INEL for four different plant configurations. The following section addresses these different modes of natural circulation. The INEL calculations were performed with RELAP. The B&W calculations are reported in the Licensee's SAR of April 12, 1979. Each of these models calculates natural circulation flow by modeling the system volumes and elevations and by balancing the frictional and elevation pressure drops throughout the system. The RELAP model used for these calculations is the same model used by INEL for Loss of Coolant Accident analysis.

3.1 Alternative Steam Generator Configurations

a) One Steam Generator Steaming and One Isolated

Both B&W and INEL calculations indicate that this is the least effective mode of natural circulation. One B&W calculation has been reported for this condition; four INEL calculations were performed. Table 3.1 presents the results of these calculations; they are reasonably consistent. The lower flow in the B&W calculation appears to be the result of the higher value of core resistance that was used.

b) Both Steam Generators Steaming

Table 3.2 presents the results of the B&W and INEL calculations for this plant configuration. As with the previous case, the results of the B&W and INEL calculations are reasonably consistent.

c) One Steam Generator Water-Solid and One Isolated

This configuration is an expected long term mode of operation while modifications are being made to the secondary system. The results of the

B&W and INEL calculations for this configuration are given in Table 3.3; they are reasonably consistent. The importance of the core flow resistance can be seen by noting the sensitivity of the results to changes in this parameter.

d) Both Steam Generators Water-Solid.

This is the configuration proposed for the long term cooling mode. Table 3.4 presents the results of B&W and INEL calculations for this mode of operation. The B&W results show approximately half as much flow as the INEL results. This difference is primarily due to the differences in core flow resistances used in the calculations.

3.2 Staff Evaluation

On the basis of comparisons with the natural circulation test data taken at Oconee (a plant similar in design to TMI-2) we conclude that either the B&W or the INEL codes provide reasonable estimates of the reactor coolant system flow rates during natural circulation as a function of core flow resistance. In addition, we conclude that the mode of operation proposed in the Licensee's SAR of April 12, 1979, which calls for the use of both steam generators in the water-solid closed-cycle cooling mode, is the most favorable mode of long term operation in that it provides larger core flow than the others studied. Based on the calculated flow rates we conclude that any of the above modes of natural circulation operation will provide sufficient flow to maintain the average temperature at the core exit below saturated conditions. The expected local coolant temperatures with natural circulation using one or two steam generators in the water-solid mode is addressed in the next section of this report.

Table 3.1

Natural Circulation with One Steam Generator
Steaming and One Isolated

Source	S.G. Level	Core Resistance	Power	Flow Rate in Core
B&W	30 ft.	60 x Nominal	5 MWt	145 lb/sec to 306 lb/sec
INEL	30 ft.	Nominal	5 MWt	430 lb/sec
INEL	30 ft.	Nominal	3 MWt	350 lb/sec
INEL	30 ft.	35 x Nominal	5 MWt	320 lb/sec
INEL	3 ft.	Nominal	5 MWt	350 lb/sec
INEL	12 ft.	Nominal	5 MWt	380 lb/sec
INEL	30 ft.	400 x Nominal	5 MWt	98 lb/sec
INEL	30 ft.	10,000 x Nominal	5 MWt	12 lb/sec

Table 3.2

Natural Circulation with Both
Steam Generators Steaming

Source	S.G. Level	Core Resistance	Power	Core Flow Rate
B&W	30 ft.	Nominal	5 MWt	417 lb/sec to 638 lb/sec
B&W	30 ft.	10 x Nominal	5 MWt	306 lb/sec to 444 lb/sec
B&W	30 ft.	60 x Nominal	5 MWt	222 lb/sec to 333 lb/sec
INEL	30 ft.	Nominal	5 MWt	500 lb/sec

4.0 Core Cooling by Natural Circulation

4.1 B&W Analysis

Operation of the TMI-2 core with natural circulation cooling has been evaluated by B&W and judged acceptable, as indicated in the Licensee's SAR of April 12, 1979. Minimum core flow requirements have been established based on a margin of three times the flow rate required to prevent bulk coolant temperature from exceeding the saturation temperature. Predictions of natural circulation core flow for assumed flow resistances of 60 to 1000 times the resistance of a normal 177 fuel assembly core show adequate flow to meet the required flow rate for the operating mode in which one steam generator is water-solid on the secondary side (see Table 3.3).

The effects of local boiling due to flow blockage and high temperature at fuel debris locations have been considered by B&W. It is expected that any increased local flow resistance due to boiling will be compensated by increased local convection due to the density change of vaporization. If fuel debris is located near the incore thermocouple junctions, thermocouple readings will indicate the localized heating effects rather than the bulk fluid temperature. Therefore, B&W and the Licensee recommend that it not be required that all incore thermocouple readings be maintained below the saturation temperature in the natural convection mode.

Transient natural circulation analyses were performed by B&W at core decay heat levels of 2 and 3 megawatts to evaluate the change in the core flow rate and the change in temperatures following the loss of forced circulation flow.

The core flow rate calculated by B&W drops to a minimum at about one minute into the transient, then rises and stabilizes between 10 and 20 minutes. The core outlet temperature reaches a maximum value at 4 to 5 minutes into the transient and a lower stable condition at about ten minutes at a value that will depend upon the initial primary coolant temperature. Equilibrium values for flow and temperature should be reached within the first one-half hour of the transient.

The B&W analysis has also considered the time available to establish and evaluate natural circulation flow conditions during the transition from one pump operation. For the preliminary values of reactor temperature and pressure at the time of transition, more than one hour would be required to heat water in the core region to the saturation temperature even if no core flow could be established. An additional hour would be required (at 3 MWt) to boil out enough coolant to drop the level from the outlet nozzle level to the top of the core. If the core were to become instantaneously uncovered, more than one hour would be required (at 3 MWt) for the adiabatic heat-up of the core average temperature from 200°F to 2000°F. These estimates indicate that reasonable time is available for decisions regarding alternative modes of core cooling (see Section 7, below).

Criteria for the core exit thermocouples and the hot leg RTD's have been proposed by B&W and the Licensee for monitoring the transition to and maintenance of adequate natural circulation cooling. These criteria are presented in the Licensee's SAR of April 12, 1979 (Figure 6 of Attachment 1 and Reference 2 of Attachment 4). The staff believes these criteria may be too conservative in the sense of causing natural circulation to be discontinued prematurely. The staff's recommended criteria are provided in Section 4.3 below.

4.2 Staff Analysis

The staff's model of core thermal hydraulics described in Section 2 was used to calculate coolant temperatures during natural circulation. A range of natural circulation flow rates was derived from Section 3 of this report.

Coolant temperatures were calculated for three assumed plant configurations, corresponding to a flow rate of 231 lb/sec for the case of natural circulation with one steam generator in a water solid condition and a flow rate of 251 lb/sec for the case of natural circulation with both steam generators in a water solid condition. The coolant temperature calculations were

Table 3.3

Natural Circulation Flow with One Steam Generator Solid and One Isolated

Source	S.G. Level	Core Resistance	Power	Core Flow Rate
B&W	Solid	60 x Nominal	3 MWt	194 lb/sec to 306 lb/sec
B&W	Solid	60 x Nominal	2 MWt	167 lb/sec to 278 lb/sec
B&W	Solid	Nominal	3 MWt	380 lb/sec
B&W	Solid	200 x Nominal	3 MWt	231 lb/sec
B&W	Solid	1000 x Nominal	3 MWt	142 lb/sec
INEL	Solid	Nominal	5 MWt	420 lb/sec

Table 3.4

Natural Circulation Flow With
Both Steam Generators Solid

Source	S.G. Level	Core Resistance	Power	Core Flow Rate
B&W	Solid	60 x Nominal	5 MWt	222 lb/sec to 333 lb/sec
INEL	Solid	Nominal	5 MWt	568 lb/sec
INEL	Solid	3 x Nominal	5 MWt	∿ 560 lb/sec

performed for various decay heat levels. The results of these calculations are summarized in Table 4.1 and Table 4.2. These results were obtained from COBRA IV calculations and checked with closed-channel hand calculations. These calculations for each plant configuration and each decay heat level were performed with two different sets of assumptions. The first set used the core exit thermocouple data in a conservative manner by assuming that during natural circulation the fraction of natural circulation flow in each region of the core relative to the average natural circulation flow rate was the same as the fraction of pumped flow calculated for each region of the core relative to the average pumped flow. For this calculation it was assumed that only about 6% of the normal pumped flow rate is going through the core. In this calculation the effect of local void generation on the elevation head is conservatively ignored. In addition the increased flow resistance due to two-phase flow conditions is also ignored. The second calculation used core flow thermocouple data in a more realistic manner by using the calculated natural circulation flow rates directly from the RELAP calculation. In this case the increased core flow resistance causes an increased core exit temperature and increased local void generation and therefore an increase in natural circulation flow which partially offsets the adverse effects of the local core blockage.

These analyses indicate that extensive local boiling might occur under natural circulation flow conditions. In addition, several cases analyzed show superheated steam conditions in one or more assemblies. The maximum coolant temperatures were calculated for the case with one steam generator operating. A temperature of 2960F was calculated at the core exit for the core decay power level on April 16, 1979. As indicated in the tables there is a significant sensitivity of this temperature to the reduction in decay heat as a function of time. By April 30, 1979 the highest temperature is calculated to be 1390F. These calculations also show a significant sensitivity to the assumed core flow rate. As seen from the previous section, the value of core flow rate is dependent on the total core flow resistance. It is reasonably certain that the central portion of the core, which has been showing high thermocouple readings, is very nearly completely blocked to flow. However, there is considerable uncertainty as to the condition of the peripheral assemblies. The difference between an unblocked region and a 50% blocked region is only about 1°F in core exit temperature. Since the exit temperatures are not measurable to within one degree, the condition of the peripheral region is somewhat uncertain. The core exit temperatures presented in Tables 4.1 and 4.2 assume 90% blockage in the peripheral region. With this assumption, the total core blockage is approximately 94%.

It should be noted that the hot assembly temperatures in Tables 4.1 and 4.2 only apply to a relatively small fraction of the core. For all of the cases calculated, most of the core exit temperatures are predicted to remain below 1000°F. Temperatures below 1000°F would not be expected to have any adverse influence on the ability to keep the rest of the core cooled. In addition, these temperatures would not result in any loss of strength in the intact structures nor would they be expected to cause any significant movement in the fuel debris. Temperatures in excess of 1000F would, however, continue to drive more fission products out of the UO_2, thus tending to prolong the eventual cleanup of primary coolant. Prolonged local boiling also will lead to boron precipitation tending to further block areas already difficult to cool.

In summary, local boiling and local superheated conditions are possible if the transition to natural circulation takes place during the next few weeks. The maximum local superheated temperatures are all expected to be well below the UO_2 melting point, the Inconel melting point, and the threshold temperature for zirconium water reaction for natural circulation after April 16, 1979 with two steam generators in the water-solid condition. After April 30, 1979 comparable performance would be expected for natural circulation with one steam generator in the water-solid condition.

There is some potential that small pieces of uranium oxide and zirconium oxide particulate debris which are now suspended in the core by forced circulation cooling could fall through the core to the lower plenum once the pump is tripped and core flow drops to natural circulation rates. A simple one-dimensional conduction analysis (conduction in the vessel wall is negligible) was performed to determine the depth of material needed in the lower plenum to cause incipient melting of the vessel wall.

The results of this conservative approach are shown in Figure 4.1 for several values of the controlling parameters. Since approximately 10% of the core UO_2 volume is needed to produce a 5 inch layer of debris in the lower plenum, and since little suspended matter is expected to fall to the lower plenum upon loss of forced circulation, tripping the pump to go to natural circulation is not expected to add more than one inch to the debris layer already in the lower plenum. The amount of debris in the vessel lower head is judged to be small (see Attachment 1).

An additional analysis performed at Sandia Laboratory and reviewed by the staff shows that a debris layer in the lower head would be cooled by cellular convection and that the heat generation in the pile would be too low to cause dryout of the debris. Sandia concludes that the temperature in the debris pile, and hence at the liner of the lower head, is only a few hundred degrees higher than the water temperature. Therefore, we conclude that the debris pile can be cooled sufficiently for any depth of debris of practical concern at TMI-2.

4.3 Criteria and Conclusions

The transition from forced convection core cooling using one reactor coolant pump to the natural circulation mode should be possible with minimum risk of further core damage (zirconium oxidation, fuel melting or structural degradation) and minimum risk of unacceptable activity release. The functional criteria to accomplish these goals should be as follows:

(a) Avoid bulk boiling of core coolant

(b) Avoid prolonged conditions where local temperatures of core materials would be predicted to exceed 1000F

(c) Avoid system conditions that could result in pressure instability with susceptibility to large pressure pulses, e.g., extensive boiling in large regions of the core.

(d) Avoid operations that have the potential for relocation of fuel debris, e.g., starting more than one reactor coolant pump.

(e) Avoid conditions that could result in noncondensible gas evolution which could affect the natural circulation cooling, e.g., pressures below 300 psig, or rapid depressurization.

(f) Avoid boron dilution and the potential for return to criticality.

(g) Avoid rapid or unnecessary additions of cold water to the reactor coolant system (e.g., through normal makeup system) because of the possible adverse effect on natural circulation.

(h) Confirm that a stable core ΔT of 5F to 150F is established during the first half hour of natural circulation.

(i) Maintain sufficient void in pressurizer to accommodate heat up of fluid between core and steam generator when going into natural circulation.

(j) Maintain sufficient level in pressurizer to accommodate primary system shrinkage if reactor coolant pump is restarted.

(k) Keep pressurizer spray line open to vent any gases that could collect in pump or cold leg.

(l) Minimize potential for local boron precipitation which could clog flow passages in local boiling areas of the core.

The B&W criterion for avoidance of bulk boiling in natural circulation is to assure 100F subcooling in the hot leg as measured by the RTDs. The staff believes this to be adequate for operation with two steam generators. However, the thermocouples at the core exit are a more reliable indication of core boiling while bypass flow exists in an idle steam generator. We conclude that natural circulation should be discontinued with a return to forced circulation operation with one reactor coolant pump if the average value of incore thermocouple readings should exceed the saturation limit indicated on Figure 4.2.

Additionally, the B&W criteria provide for a maximum coolant temperature rise of 150F across the core and require a hot leg temperature rise within the first hour during transition to natural circulation. In addition, the hot leg temperature is not to exceed 250F for pressure greater than 500 psia (180F for pressure near atmospheric). The staff finds these limits acceptable. However the Licensee's recommended limits of ten or more incore thermocouples reading lower than saturation temperature is not needed in view of the staff requirement on the average readings. Likewise, the Licensee's local limit of 800F on two thermocouples appears arbitrary since local high temperatures are acceptable. However, no more than three of the thermocouples should be permitted to remain above 1000°F for more than one hour if restart of a reactor coolant pump is a viable option.

The criteria that no more than three thermocouples should be allowed to indicate local temperatures above 1000°F is based on the following considerations:

1. It is our best estimate that only one instrumented assembly will approach 1000°F if natural circulation is achieved on April 24 or soon thereafter; up to five others could show approximately 100°F of superheat. Therefore, three thermocouples with temperatures above 1000°F would indicate that natural circulation is not being achieved as expected.

2. It is likely that during natural circulation many thermocouples will read temperatures lower than the maximum local temperatures. This conclusion is based on early thermocouple data; changes in the thermocouple data following the startup of RCP 2A; and on PNL and ORNL calculations. The early thermocouple data indicated temperature

differences of 300°F or more in adjacent assemblies. After following the trend of the thermocouple data for a long period of time it seems clear that these differences do not represent significant differences in the conditions within the adjacent assemblies, but rather differences in the thermocouple location relative to the maximum temperature in the assembly. While the higher temperatures are probably the result of material filling the entire assembly region up to the thermocouple location, the lower temperatures are probably the result of having some space between the debris bed and the thermocouple location. Both PNL COBRA IV calculations and ORNL SABRE calculations indicate that flow redistribution in a small region (6 to 9 inches) above the debris bed would be expected to produce a difference of 100°F in the temperature at the top of the debris bed.

In addition, this mechanism is a likely explanation of the 120°F drop in indicated temperature of the hot assemblies when RCP 2A was started-up. That is, the 120°F drop in indicated temperature may have been the result of the debris bed settling down to an elevation several inches below the thermocouples.

We conclude that an indicated temperature of 1000°F during natural circulation probably corresponds to a maximum temperature of approximately 1300°F in the debris bed. Exceeding 1300°F over a significant portion of the core is undesirable since the structural Inconel loses strength rapidly as a function of temperature at this elevated temperature. The degree to which these structures are needed to support the core in its present condition is unknown. Some additional zirconium oxidation could also occur at this temperature and would be undesirable.

3. It is believed that the 1000°F criterion is not overly conservative since our thermocouple predictions are based on current thermocouple readings with no core displacement when the pump is tripped. We

expect that some fuel debris will settle under the low natural circulation flow rate and resultant thermocouple readings may be lower than predicted.

The primary system should not be in a water-solid condition when transition from forced to natural circulation cooling is accomplished. It is anticipated that local boiling will occur during the transition to natural circulation. Local boiling at present decay heat levels will not be capable of sustaining large voids and the collapse of small voids should not be sufficient to cause core structural damage. However, the resultant small pressure pulses could lead to small scale redistribution of the core debris which would affect flow blockage and the settling of debris.

Minimum primary system pressure required to prevent noncondensible gas evolution and an evaluation of the effects of a coolant voids on natural circulation are discussed in Section 5 of this report.

Calculations of core exit coolant temperatures based on thermocouple data, indicate that local temperatures could exceed 1000°F if transition to natural circulation is attempted prior to about 5/5/79. The high temperatures will not necessarily be indicated on core exit thermocouples since the debris or blockage which is producing the high temperature may drop away from the thermocouple locations under the low flow natural circulation conditions. The planned transition to natural circulation during the first two weeks of May is consistent with the decay time for local debris that should preclude measured temperatures in excess of 1000°F and will minimize the potential need to restart the coolant pumps which could lead to further redistribution of debris. The transition to natural circulation provides less potential for local boiling and coolant superheating the latter it is accomplished. So long as the reactor coolant pump now running remains in a satisfactory status, forced convection cooling should be maintained to take maximum advantage of fission product heat decay, consistent with the need to accomplish an orderly and well planned transition to natural circulation.

The staff has concluded that the ECC injection modes (HPI and RHR) do not offer significant advantages over natural circulation cooling and have several disadvantages (see Section 7, below). Therefore, we do not recommend interruption of the natural circulation mode prematurely in favor of HPI or RHR unless the above criteria are exceeded and improved conditions can be reasonably expected in the alternative cooling modes.

In addition to the stated criteria, systematic failure of core exit thermocouples in a given region should be viewed with caution during the transition to natural circulation as it might signal high local core temperatures and a warning that effective cooling is not being accomplished.

The limitations on process parameters to assure adequate core cooling in natural circulation are summarized in Table 4.4 and Figure 4.2. Provisions for continuous monitoring and recording of incore thermocouples should be included in the plant modifications for natural circulation.

The staff has developed criteria for detecting loss of natural circulation during long term core cooling:

1) Increase in the average reading of all operable core outlet thermocouples by more than 10°F, or

2) Increase in average of the two hot leg temperatures by more than 10°F, or

3) Increase in reactor pressure corresponding to a 10°F rise in saturation temperature, or

4) Marked and unexpected decrease in total heat removed by the secondary side of the steam generators ($F\Delta T$ decrease).

Once natural recirculation is lost, it may well re-establish itself through the mechanisms described in Section 5.0. Criteria for detecting re-establishment of natural circulation cooling are:

1) Constant or decreasing average core outlet temperature (average of all core outlet thermocouples) over a period of ½ hour or more at a temperature below the saturation pressure corresponding to the safety valve set point (assuming an unvented system), and

2) Constant or decreasing average of the two hot leg temperatures over a ½ hour period, and

3) Decreasing reactor pressure over a ½ hour period, and

4) Increase in total FΔT on the steam generator secondary side.

The staff concludes that the proposed transition to natural circulation can be accomplished with minimal risk to the public health and safety, and that natural circulation is the preferred long term cooling mode. The planned transition in early May should minimize the potential for local fuel temperatures in excess of 1000F.

Table 4.1

Core Exit Coolant Temperatures
During Natural Circulation
With One Steam Generator Operating in
a Water Solid Condition

Flow = 231 lb/sec

Tin = 125F

Pressure = 1000 PSIA
X = Steam Quality

Day	Decay Heat Power Level	Calculated Temperatures Case 1 (No credit for local buoyancy)		Calculated Temperatures for Case 2 (Including local buoyancy effect)
4/23	0.08%	Hot assembly =	2060F	545F
		2nd hottest assembly =	545F	345F
		Peripheral Region =	183F	140F
4/30	0.065%	Hot assembly =	1390F	460F
		2nd hottest assembly =	545F	300F
		Peripheral Region =	172F	137F
5/2	0.060%	Hot assembly =	1150F	430F
		2nd hottest assembly =	545F	284F
		Peripheral Region =	168F	136F

Table 4.2

Core Exit Coolant Temperatures
During Natural Circulation Operation
With Two Steam Generators Operating in
a Water Solid Condition

Flow = 251 lb/sec

Tin = 125F

Pressure = 1000 PSIA
X = Steam Quality

Day	Decay Heat Power Level		Calculated Temperatures Case 1 (No credit for local bouyancy)	Calculated Temperatures for Case 2 (Including local bouyancy effect)
4/23	0.08%	Hot assembly =	1120F	545F
		2nd hottest assembly =	545F	330F
		Peripheral Region =	168F	139F
4/30	0.065%	Hot assembly =	600F	415F
		2nd hottest assembly =	545F	280F
		Peripheral Region =	160F	137F
5/2	0.06%	Hot assembly =	545F	375F
		2nd hottest assembly =	444F	245F
		Peripheral Region =	157F	134F

Table 4.4

Criteria for Core Cooling

Operating Modes

1. One Reactor Coolant Pump (RCP) Operating
2. Transition to Natural Circulation During First Hour Following RCP Trip
3. Natural Circulation Thru One Steam Generator
4. Natural Circulation Thru Two Steam Generators
5. High Pressure Injection
6. Decay Heat Removal

Operating Limits	Applicability to Operating Modes
1. Core Exit Thermocouples < 1000F	1,3,4,5, & 6
2. Average of Core Exit Thermocouples < T_{sat}	1,2,3,5, & 6
3. Hot Leg Temperature > 100F Subcooled	1,3,4,5, & 6
4. Hot Leg Temperature < 250F	All Modes when P>500 PSIA
5. Hot Leg Temperature < 180F	All Modes when P<15 PSIA
6. Core ΔT < 150F	All Modes
7. Hot Leg Temperature Increase Within One Hour After Reactor Coolant Pump Trip	3

Monitor Limits	Operating Limit	Action (See below)
1. Average Incore Thermocouples > T_{sat} (Figure 4.2)	2	1
2. Two Incore Thermocouples > 1000F	1	2
3. Three Incore Thermocouples > 1000F	1	1
4. Hot Leg RTD Exceeds 100F Subcool Limit of Figure 5.2	3	1
5. Hot Leg Temperature > 250F	4	1
6. Hot Leg Temperature Minus Cold Leg Temperature > 150F	6	1
7. Two or More Incore Thermocouples Fail Within 30 minute interval	1	3

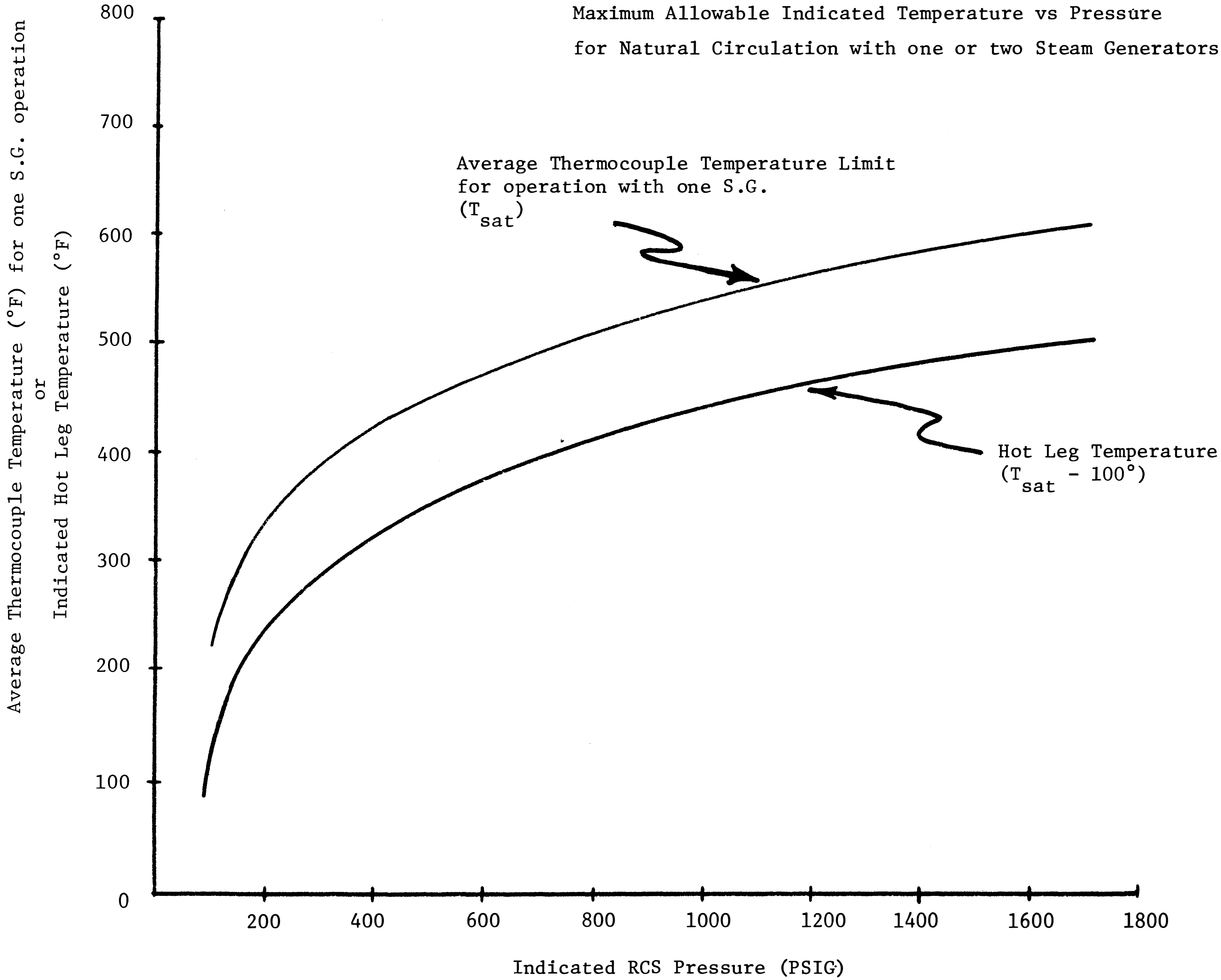

Maximum Allowable Indicated Temperature vs Pressure
for Natural Circulation with one or two Steam Generators
Average Thermocouple Temperature Limit
for operation with one S.G.
(T_{sat})
Hot Leg Temperature Limit
(T_{sat} - 100°)
Average Thermocouple Temperature (°F) for one S.G. operation
or
Indicated Hot Leg Temperature (°F)
800
700
600
500
400
300
200
100
0
200
400
600
800
1000
1200
1400
1600
1800
Indicated RCS Pressure (PSIG)

Figure 4.2

Table 4.4 (Continued)

Actions

1. Return to Mode 1, if possible. Otherwise consider transfer to an alternative cooling mode.

2. Validate incore thermocouple operability by comparison to other nearby thermocouples and by other available means.

3. Danger of local temperatures high enough to melt incore instrument structures. Evaluate and take Action 1 or other emergency actions as warranted.

Thickness of Debris Layer to Heat
Liner of Lower Plenum to Melting
Point (T_{melt} = 2550°F)

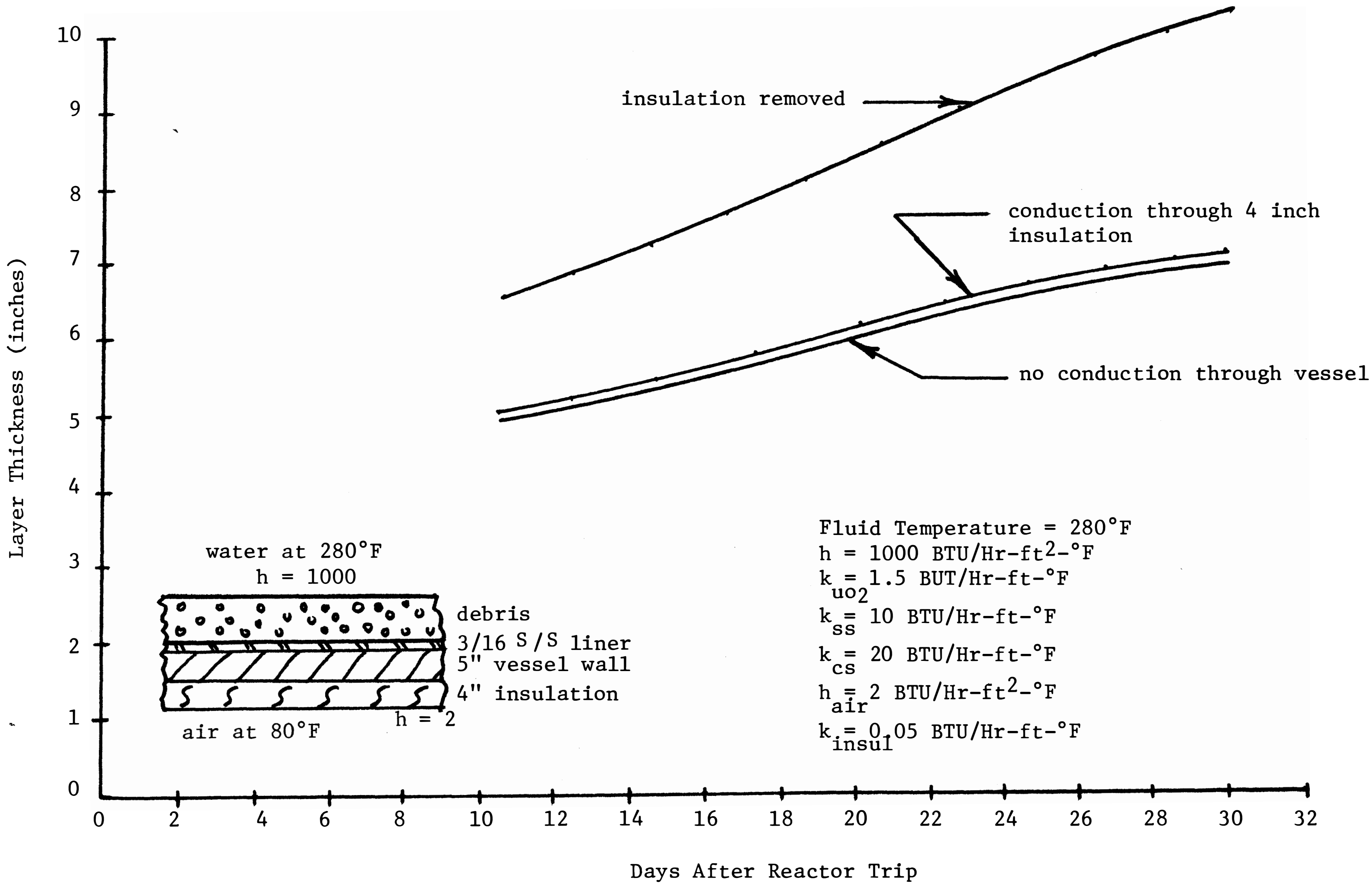


Figure 4.1

5.0 Effects of Non-Condensible Gas on Natural Recirculation Cooling Capability

Measurements during the first week of the accident indicated the presence of a large volume of non-condensible gas in the reactor coolant system. Subsequently the gas was removed through a degassing procedure involving pressurizer spraying and letdown. The procedure included a series of incremental pressure reductions from 1000 psig down to 300 psig to remove any gases trapped at high points in the reactor coolant system. It is currently estimated that no bubble will form by evolution of non-condensible gases now in solution or still residing in RCS pockets, as long as the primary coolant system pressure remains in excess of 300 psig.

Consideration has been given to the potential effects of radiolysis of the reactor coolant water, i.e., the decomposition of the water to form hydrogen and oxygen.

In gamma and neutron fields typical of nuclear power reactor coolant systems, a hydrogen concentration of 17 cc/kg water is needed to suppress radiolysis of the primary coolant (Ref: US Patent 2937981, 5/24/60). In operating plants the usual concentration of hydrogen is maintained between 25 cc/kg and 35 cc/kg.

In the TMI plant, the saturation concentrations of hydrogen are listed below for the preliminary points in the Base Plan (described in Figure 1.1):

Operating Point "A" (280°F, 1000 psia)	1670 cc/kg
Operating Point "B" (220°F, 1000 psia)	1540 cc/kg
Operating Point "C" (140°F, 1000 psia)	1430 cc/kg
Operating Point "D" (140°F, 100 psia)	140 cc/kg.

These values are considerably higher than the concentration of hydrogen required for the suppression of radiolysis of water at the operating conditions of the plant. It can be concluded, therefore, that no significant radiolysis will take place at the operating conditions defined by points A, B, C and D as long as the primary coolant remains saturated with hydrogen. Some radiolysis may be expected if the concentration of hydrogen is reduced below the saturation limit and it reaches a value lower than 17 cc/kg.

The Licensee obtained pressurized samples of primary coolant from TMI-2 on 4/22/79 to determine the amount of hydrogen and other gases in solution. The preliminary analysis of this coolant sample indicates the following:

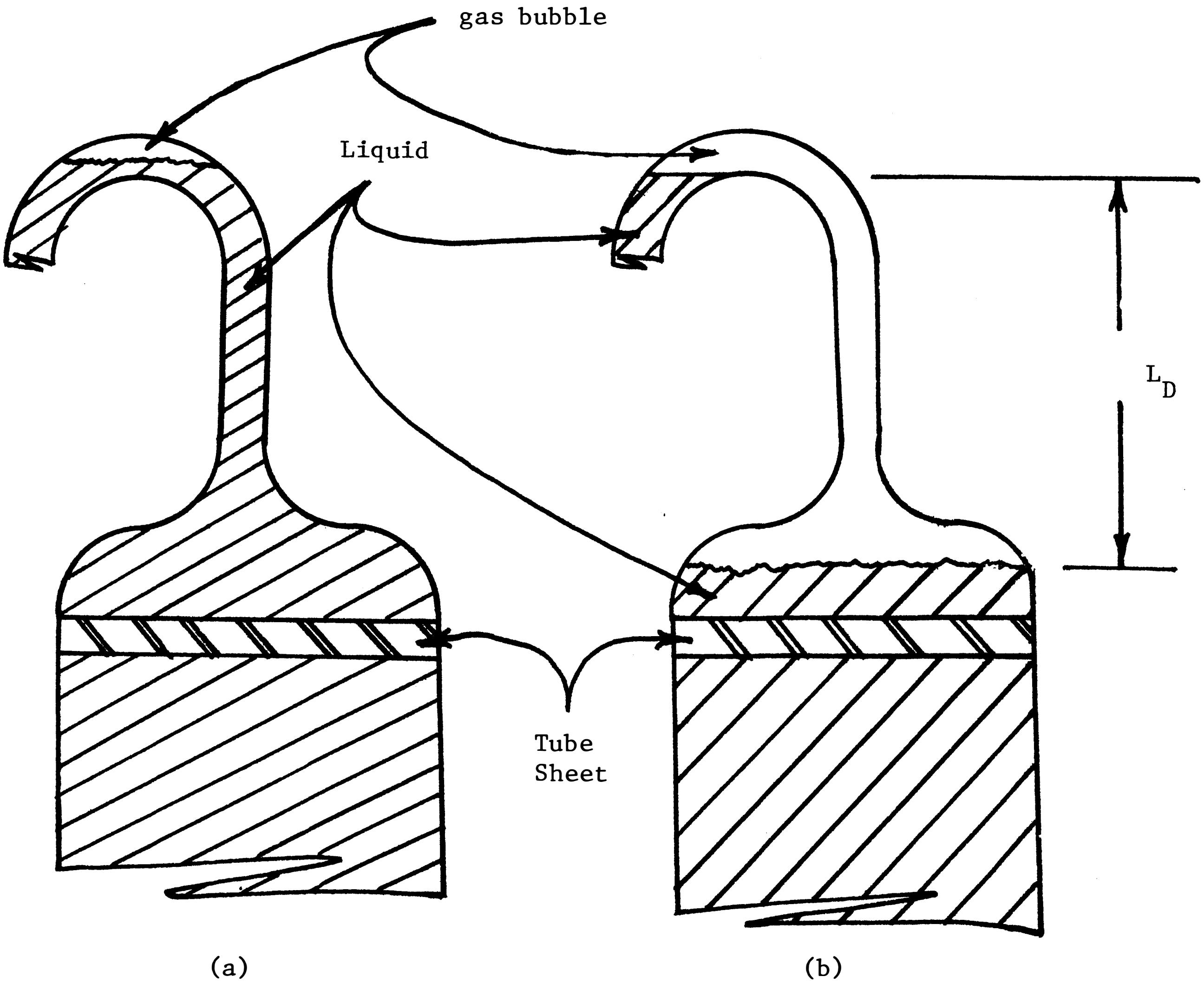
Hydrogen	- 23.8	cc/kg
Oxygen	- <1.1	cc/kg
Nitrogen	- <9.9	cc/kg

Natural convection cooling in the primary system (as described for the Licensee's proposed long term cooling mode and evaluated above) assumes the presence of no free gas volume in the primary coolant system for pressures greater than 300 psi for the reasons described above. However, the staff has nevertheless evaluated the effects of non-condensible gas that might unexpectedly come out of solution, as described in the following paragraphs.

If the system pressure and the core exit plenum and hot leg temperatures allow gas to come out of solution, the formation of gas pockets could interfere with natural circulation. The gas could be driven back into solution by increasing pressure to maintain sufficient subcooling. The time required for plant operators to accomplish this reverse process is on the order of hours which is within the available time frame for operator actions following loss of natural recirculation. Nevertheless we arbitrarily assumed that the released gas might remain out of solution for the following evaluation.

The released gas would collect at high points. These points are (a) the upper portion of the reactor vessel, (b) the "candy canes" near the top of the steam generators, and (c) the reactor coolant pumps. For natural circulation flow, the gas volume in the steam generator "candy canes" is of particular concern since it could inhibit coolant flow to the steam generators. If the volume is small as shown in Figure 5.1a, there should be little flow impedance. However, if the volume is large enough to fill the piping, as illustrated in Figure 5.1b, the flow could be at least temporarily stopped.

Natural circulation flow tests with an intact core and steam generators in a steaming mode at conditions where the decay heat rate is about 0.2 percent of design power have shown that the ratio of coolant flow to core power is roughly five times that for the reactor at full power with all pumps running. The resulting small temperature increase across the core (10-15°F) produces a small driving head for natural circulation which could be stopped by a gas volume large enough to produce a liquid depression, L_D, in the "candy cane" of less than 6 inches (see Figure 5.1b). However, with cessation of natural circulation flow, the core and hot leg temperatures would increase and produce a higher driving head. In addition, the cold portion of the system (including most of the steam generator and RCS piping cold legs) will be kept at about 100 to 150°F as the result of the flooding of the secondary side of the steam generators. The resulting increase in driving head should be sufficient to reestablish natural circulation flow for relatively large volumes of gas. Preliminary calculations indicate that gas volumes that could extend down into the upper plenum of the steam generator may not permanently stop natural circulation flow at hot leg temperatures well below the saturation temperature, providing RCS pressures are brought back to the 1000 to 1500 psig range. As noted previously, the gas volume would shrink and eventually disappear as the gas goes back into solution, provided sufficient subcooling is obtained by the increase in system pressure.



Gas Bubble in Steam Generator "Candy Cane"

Figure 5.1

6.0 Long Term Control of Reactor

6.1 Long Term Control of Reactor Coolant Level and Pressure

Pressurizer level indication is most important during the transition from forced to natural convection flow. During this transition, RCS temperature and pressure will change. Pressurizer level indication will be desirable to control the pressure and to indicate the evolution of gases in the system which might inhibit establishment of natural recirculation as discussed in Section 5. If the plant is water solid when the transition is made, the control of system pressure and inventory is more difficult.

In the long term, when the RCS is operating in a natural circulation mode, the pressurizer level indication is not as desirable as above because the system can be kept water solid.

The Licensee has proposed a method for RCS pressure (and inventory) control during long term operation in the water solid natural circulation mode. This system would serve as a backup to the CVCS and maintain reactor coolant system pressure with the pressurizer filled solid with water. Primary coolant system pressure would thus be maintained even with the loss of pressurizer instrumentation and inoperative pressurizer heaters. Also, the pressure control system will be designed to provide adequate NPSH to the reactor coolant pumps if they are needed, and to suppress potential bubble formation which could inhibit natural recirculation as discussed in Section 5.

The standby reactor coolant pressure control system will consist of passive components (a series of water storage tanks and a surge tank with nitrogen blanket for pressure control) and active components (charging pumps). The system will control reactor coolant pressure over the range of 100 psig to 750 psig.

The intent is to use the passive reactor coolant pressure system which would be at first locally controlled. Later, additional instrumentation and remote control will be incorporated to automate this system. The active pressure control portion would resupply water to the surge tanks with added capability of providing additional makeup water directly to the RCS if needed.

The above modification will establish a flowpath of makeup water and pressure control through the normal makeup lines that interface with the reactor coolant loop cold legs. Chemical control of the degassified borated water used in the pressure control system will be provided by the present chemical addition system.

Connections will be provided to accommodate the addition of boric acid, H_2, demineralized water and hydrazine, LiOH, and NaOH.

6.2 Core Cooling Without Primary System Convection

After some months of fission product decay, an alternative to natural convection coding will be available. It should be possible to cool the TMI-2 core without circulation (forced or natural convection) through the reactor coolant system. That is, the core could be cooled by simply maintaining an essentially water solid primary system and conducting and convecting heat from the reactor vessel to the containment atmosphere.

From the calculations based on the specific operating history of TMI-2, the decay heat (Figure 6.2 shows decay heat from 0 to 130 days following shutdown) is expected to decrease to about 0.3, 0.14, and 0.05 MW at six months, one year and two years after the accident, respectively. If the reactor coolant system is kept at a temperature of about 100°F, the heat loss by conduction or convection to containment would be small relative to the heat that can be removed by the steam generators. However, after six months, it appears that decay heat could be removed by heat loss to the air and earth surrounding the reactor vessel in the containment, while still maintaining the reactor coolant temperature well below normal operating temperatures (the primary system would require pressurization to suppress boiling).

Since data for TMI-2 were not available, the heat loss to containment from the vessel was estimated by scaling values from other PWRs. It is estimated that the heat loss to containment from the reactor vessel, the primary coolant piping, control rod drive mechanisms, steam generators, steam piping and uninsulated parts of the rector coolant system pumps is about 1.5 MW at design power conditions. Roughly one half of this heat loss can be attributed to the control rod drive mechanisms which have cooling water coils and forced convection air cooling. The heat loss from the control rod drive mechanisms is herein conservatively assumed to be negligible due to conservatively assumed loss of the cooling water and forced air flow. On this basis, the heat loss from the reactor coolant to containment air is estimated to be about 1.6×10^{-3} MW/°F. Assuming the containment air ventilation fans are lost, a conservative estimate of the heat loss by conduction from containment to the surrounding air outside containment is 3×10^{-3} MW/°F. From these values, it is estimated that the 0.3 MW decay heat at six months after shutdown could be transferred directly to 70°F air outside of containment with a reactor coolant temperature of about 400°F.

Better estimates of the heat losses under these conditions can be inferred later from plant records at TMI-2.

It is possible that a significant increase in heat removal could be obtained by using the cooling water to the control rod drive mechanisms which would be available for the long term operation. It is noted, however, that natural circulation flow in the reactor coolant system during this mode of operation (steam generators isolated) would be impeded since the major portion of the decay heat would no longer be removed at the top of the steam generators. Further analysis of the natural circulation flow and adequacy of core cooling will be needed to demonstrate the feasibility of this backup mode of operation.

6.3 Overpressure Protection/Thermal Shock

With the RCS temperature low, around 150°F as in natural recirculation cooling, the reactor vessel pressure must not be excessive due to the possibility of brittle fracture. System pressure should be carefully monitored and controlled to avoid overpressurization.

The rapid addition of cold RCS makeup should be avoided since the makeup inlet points are located in the RCS cold legs, and the cold fluid would travel into the reactor inlet nozzles, down the downcomer, then into the inlet plenum. The cold makeup could cause thermal shock and brittle fracture potential for the reactor vessel since there is no heating of the fluid until it enters the core. Therefore, continuous rather than intermittent RCS makeup should be considered. Consideration should also be given to provisions for heating of makeup water to the long term primary coolant temperature.

Fracture mechanics calculations have been performed for several cases that could be encountered in the planned cooldown of TMI-2. In all cases, the possible atypical weld metal in the lower head is limiting. Nevertheless, assuming reasonable mixing of the water, our calculations show that there is no need for concern about brittle fracture of the vessel unless extremely unlikely conditons occur.

Appendix G calculations were first performed using all of the conservative Appendix G assumptions. These include a 1/4T flaw, the Appendix G K_{IR} curve, and a factor of 2 margin on pressure stresses. This gave a minimum temperature of 160°F for 1000 psig pressure and a cooldown rate of 50°F/hr.

Next, thermal stresses and stress intensity factors were calculated for the proposed cooling parameters. This gave a slightly higher cooldown rate and slightly higher thermal stresses and stress intensity factors.

Again, using the Appendix G factor of 2 margin on pressure stresses, the K_{IR} curve, and the 1/4T flaw, the minimum temperature to comply with Appendix G was 170°F.

If the pressure is below about 900 psig, Appendix G requirements and margins would be met at 150°F.

Calculations were also performed assuming a pressure increase to 2500 PSIG. Using K_{IC} instead of K_{IR}, with a factor of 2 margin on pressure stress, and a 1/4T flaw, a temperature of 185°F would be required. With no factor of 2 margin on pressure stresses a temperature of 140°F could be tolerated by the vessel.

Therefore, the staff's conclusion is that there is a very low probability of vessel failure under conditions postulated to occur during the planned cooldown.

6.4 Solid Conditions in Secondary System Piping

The piping systems affected by operation with water-solid conditions on the steam generator secondary side, out to the first isolation valve, are the main steam line, main feedwater line, and the auxiliary feedwater line. The design of both feedwater lines is predicated upon being filled with water during operation and therefore, normal code allowable stresses will not be exceeded. Although the main steam line is not filled with water normally, the entire secondary system was filled with water during pre-operatonal hydro-static testing. The additional dead weight contribution to the piping stresses for the water-solid condition is accommodated within normal code limits for that portion inside of containment (based upon oral input from Burns and Roe, the architect engineer for TMI-2.) The spring hangers (one on one main steam line and three on the other) will bottom out and act as rigid restraints. For the main steam piping in the Turbine building the spring hangers will be pinned so as to carry the additional dead weight load of the water in the piping within normal code limits.

It should be noted that all ASME Section III CL.2 components used in the cooling system were designed for seismic Category I service. However all of these components, both those that are part of the original TMI-2 Main Steam and Feedwater piping system and those obtained from other nuclear sites to be incorporated into the OTSG cooling system, are being utilized in a system with different response characteristics from that for which they were initially designed or are operating with a fluid media different from that for which they were seismicially qualified, i.e., some components designed for operation on steam during a seismic event as opposed to water filled as in the present system. Thus because of these differences from the original seismic design requirements, which can affect seismic response, these components should not be considered seismically qualified as installed as a part of the OTSG cooling system, solely on the basis of their original qualifications.

Seismic capability of these system modifications is not an acceptance criterion; therefore, no additional seismic evaluation of this system is planned.

6.5 Steam Generator Tube Integrity

Steam generator tubes are required to maintain their integrity during postulated design basis accidents including a loss-of-coolant accident (LOCA) or a main steam line break (MSLB) in combination with a safe shutdown earthquake. The design basis LOCA results in a calculated 925 psia secondary-to-primary pressure differential and the design basis MSLB results in a calculated 2200 psia primary-to-secondary differential pressure at approximately 600°F. The required margins of safety against tube failure during these postulated accidents are consistent with the margins of safety determined by the stress limits specified in NM-3225 of Section III of the ASME Code. Furthermore, a factor of safety of three against tube rupture is required during normal operating conditions which corresponds to a 1250 psia primary to secondary pressure differential at approximately 600°F.

Babcock and Wilcox has provided results of laboratory tube burst and collapse tests. The burst tests conducted on specimens with defects up to 70% through-wall resulted in no tube failures at pressures less than 3900 psi and the collapse tests on similarly defected tubes resulted in no tube failures below 3500 psi.

Three Mile Island Unit 2 conducted a baseline inspection of 100% of the tubes in both steam generators in November 1977 following the hot functional tests. Tubes with imperfections of 40% or greater were plugged which is consistent with the basis delineated in Regulatory Guide 1.121, to maintain the factors of safety described above and provide an additional margin for possible operational degradation.

There has been no indication of primary coolant leakage through the "A" steam generator tubes since March 28, 1979. Therefore, based on the above design bases and the steam generator inspections and tube plugging which was conducted prior to that date, there are no further special precautions necessary at this time for operating the 'A' steam generator at the proposed conditions of 700 psid primary to secondary pressure differential and at temperatures below 200°F. However, the development of a leak, although considered to be highly unlikely due to the small primary to secondary system pressure differentials, should be considered and some precautions similar to those described in Section 8.3 for the 'B' steam generator should be implemented.

Under the proposed operating conditions the secondary coolant pressure on the 'B' steam generator will be maintained at a level greater than the primary system pressure such that any tube leakage in the 'B' steam generator would not permit highly contaminated primary coolant to enter the secondary coolant.

6.6 Condenser Flooding

Potential safety concerns associated with flooding of the condenser were considered. Since condenser integrity is not normally included in NRC's safety review, little information was immediately available to determine the safety margins for static or dynamic flooding forces. However, operation of the condenser in the spraying mode, with the hot well at normal operating water levels is anticipated, and potential design limitations of operating the condenser in a flooded mode were not relevant.

6.7 Containment

The containment internal pressure has been slightly lower than atmospheric pressure since a few hours after the accident. Current operating procedures indicate that the water flow to the fan coolers should be terminated if the reverse pressure differential reaches 2.0 psi. This action would effectively terminate further cooldown of the containment atmosphere thereby terminating the transient. In any case, this would be a rather slow transient allowing sufficient time for proper action. We believe, however, a more severe transient should also be considered. This transient is the inadvertent operation of the containment sprays. Initiation of the sprays would result in rapid cooling of the containment atmosphere causing a corresponding rapid decrease in containment pressure. The magnitude of the pressure decrease will depend upon the inlet spray water temperature (BWST water temperature). To assure that the containment does not exceed the design reverse differential pressure of 2.5 psi, the containment parameters should be maintained above minimum values as shown in Figure 6.1. The figure indicates that for a given inlet spray water temperature, the containment temperatures as well as containment pressure should be maintained above minimum values. The pressure could be controlled by the addition of a noncondensible gas such as nitrogen or dry air. Control of containment temperature could also be achieved by terminating the water to the fan coolers. Fan operation should continue, to assure proper mixing of the atmosphere while eliminating the heat removal mechanism. Since the consequences of exceeding the reverse design pressure differential are unknown, we conclude that containment conditions should be maintained in the conditions described above to allow for inadvertent spray operation.

6.8 Criticality

From an examination of the physics startup tests (boron concentration and reactivity worth, control rod worths, reactivity coefficients, etc.) for TMI-2 and Rancho Seco (approximately the same physics parameters), reactivity states may be estimated for TMI-2. A brief summary of the results of such estimates, results from some B&W calculations on fuel redistribution, and some details of these estimates and calculations follow.

From the startup tests, if the fuel pellet configuration is essentially undisturbed from its normal condition, it is estimated that the boron concentration for a just critical cold reactor (k=1.0) ranges from about 500 ppm to 1500 ppm, depending on the presence or absence of the fuel cladding, burnable poisons and control rods. The highest required boron concentration is for no control rod or burnable poison in the core but with fuel cladding in place. The lowest boron need is for a normal, rods in, undamaged geometry. If there were no cladding, substituting borated water for clad raises the low boron requirements and lowers the high requirements. Local rearrangement of the fuel, i.e., from a normal pellet lattice to a more homogeneous water-fuel mixture, reduces reactivity. Based on B&W calculations for gross rearrangements of fuel and borated water into optimized geometries and fuel-water ratios (with no structural material or control rods present), it is estimated that at 1500 ppm boron, criticality could occur (k=1.0) with an optimized compaction of about 40% of the fuel, and at 2200 ppm with about 60% compaction of the fuel. At a boron concentration of about 3000 ppm 100% of the fuel compacted would not be critical.

These results for the basically intact fuel geometries are based on the following information and extrapolation of data for the TMI-2 startup reports and the Rancho Seco report.

1. Base. TMI-2 was initially critical at 530°F, zero power, with all control rods out (ARO), at boron concentration of 1500 ppm.

2. Burnup effect. Reactivity data indicate that during the approximately 95 full power days of operation the critical boron concentration had dropped about 100 ppm.

3. Temperature effect. From Rancho Seco data the effect of going to 300°F was to decrease the critical boron concentration by about 60 ppm. Thus for TMI-2 at the time of the accident the 300°F critical boron concentration was about 1340 ppm with ARO.

4. Moderator reactivity coefficient. At 300°F the moderator (only) temperature coefficient is about +5 x 10^{-5}/°F (rods out) at 1400 ppm boron and about -5 x 10^{-5}/°F (rods in) at 700 ppm boron. The total (including fuel) coefficient at about 1300 ppm is about +2 x 10^{-5}/°F.

5. Temperature effect. From the coefficient results the reactivity is not very sensitive to the temperature in the 100-300°F, 1000-1400 ppm boron range. The critical boron concentration with 100°F, with ARO, is about 130 ppm.

6. Control rods in. The control rods were measured to be worth about 10% Δk. The boron worth in the 100-300°F, 500-1300 ppm, rods in and out range is about 1.1 to 1.3% Δk/100 ppm. Thus the all rods in (ARI), cold, critical boron level is about 500 ppm. This is approximate because of the large extrapolation in the estimate.

7. Burnable poison. The initial burnable poison reactivity worth at full power was about 4.5% Δk. At cold conditions and at the approximately 95 full power day burnup preceding the accident, this reactivity is about 3% Δk, and is equivalent to about 250 ppm boron. Thus, if the burnable poison is no longer in the core the cold, ARI, critical boron level is about 750 ppm with ARI and about 1550 ppm with ARO.

8. Loss of cladding. Loss of cladding and replacement by the borated water is similar to a moderator density increase (small effect from Zr). Replacement of the cladding would be approximately equivalent

to a 20% increase in moderator density. At a 5 x $10^{-5}/^{o}F$ moderator temperature coefficient state, this density change would be about a 3.7% Δk reactivity change.

The average temperature coefficient over the range of boron levels produced by clad loss for (1) ARI and (2) ARO, no burnable poison states, is about -3 x $10^{-5}/^{o}F$ and +3 x $10^{-5}/^{o}F$ respectively. Thus the boron changes are about $\pm$ 150 ppm and the cold, no burnable poison, no clad critical boron concentrations are thus about 900 ppm for ARI and 1400 ppm for ARO.

9. Local fuel-water mixing. Based on B&W criticality calculations and on open literature criticality measurements and calculations, a change in local pellet configuration from the normal pellet and lattice structure to a more broken up and homogeneous fuel-water mixture would result in reactivity loss and a lowered boron critical level for either the ARO or ARI configurations.

Thus it is estimated that if the fuel is not grossly redistributed the required boron concentration is probably under about 1550 ppm, and if the control rod material is still reasonably well distributed in the core, under about 900 ppm.

If fuel is grossly redistributed the resulting system could, under conditions of optimum moderation, go critical at 1500 ppm boron or above. (Note: moderation is required since solid UO^2 spheres require enrichment over 5% to be critical.)

The B&W naval criticality group has calculated the effects of gross redistribution of the fuel. They used Monte Carlo calculations (KENO) and cross sections tested in many critical configuration calculations. The calculations used pellet nuclear parameters (because these maximize reactivity) and fuel-water (boron) ratios which had been optimized by sensitivity studies. The calculations assumed only fuel (core average enrichment) and moderator mixtures with no structure or control material (except for the boron in the moderator) or burnup. They did a series of calculations in which it was assumed that the fuel between grids (seven regions of about 21 inch thickness) stayed between the grids but were rearranged into optimum fuel-water ratio "layers"~~, and~~ in ~~which (in~~ successive calculations) top grids disappeared permitting the combination of the several "layers" of fuel. Calculations were done with boron concentrations of 2100, 3000 and 4000 ppm. Estimating leakages for these calculations and extrapolating to some other boron levels, the following results can be estimated.

1. For no collapsing of "layers" the system is well subcritical at a boron concentration of 1500 ppm.

2. For a collapse of 3 "layers", giving a combination of about 42% of the reactor fuel, criticality would be approached at 1500 ppm but it would be about 4% subcritical at 2200 ppm.

3. For a collapse of 5 "layers", giving a combination of about 71% of the reactor fuel, the system would be several percent supercritical at 2200 ppm but several percent subcritical at 3000 ppm.

4. For a complete combination of all fuel, either in a cylinder or sphere the system would be slightly subcritical at about 3000 ppm.

This last result is the basis for B&W advocating a boron level of 3000 ppm to cover the most extreme configuration.

BNL (NRC physics consultants) have also performed some calculations of upper limit, optimized fuel-borated water configurations (reflected spheres for gross geometry). A preliminary examination of these calculations shows results similar to the B&W results.

In summary, calculations indicate that some postulated gross rearrangements of the TMI-2 fuel could conceivably be critical at boron concentrations of less than 3000 ppm.

These configurations appear highly unlikely, however. With more likely configurations the required concentration would appear to be about 1500 ppm or less, with the requirement for expected configurations under 1000 ppm.

As of 4/7/79 the boron concentration in TMI-2 coolant was believed to be about 2200 ppm and B&W was advocating an increase to 3000 ppm. As of 4/19/79, based on an evaluation of the results from the second primary water sample (4/11/79), the boron concentration was above 3000 ppm (probably about 3400 ppm) and boron feed concentrations were adjusted to maintain the concentration above 3000 ppm.

For the present expected core configuration and boron concentration the core k_{eff} is well under 0.9. At this multiplication, changes in configuration or boron concentration producing reactivity changes of about 1% generally would not be unambiguously indicated by the usual (excore) nuclear instrumentation. Reactivity changes of about 1% would normally be indicated once K_{eff} is above about 0.95. However, in the case of TMI-2 such changes could be partially or highly masked by possibly destroyed neutron sources, gamma backgrounds, or disturbed or changing geometries or water densities.

Thus the prediction of the ability to see an approach to critical would be highly speculative. The excore startup range nuclear detectors have been reading higher than normal (compared to post cold start ups) for the expected reactivity state of the core. While there have been no apparent correlations of count rate with boron concentration changes in the reactor, thus indicating the higher counting level is not indicative of high multiplication in the core (which should be significantly affected by boron changes which have occurred), this does, at least, present some evidence of the uncertainty of monitoring approach to critical.

Because of the possible difficulties on monitoring an approach to critical and because there apparently are possible (even though unlikely) critical configurations of the fuel material (and water) with boron concentrations under about 3000 ppm, it is recommended that the boron concentration not be allowed to fall below 3000 ppm. If, during any operation, there are steaming processes which may concentrate boron to greater than 3000 ppm such that plugging of some part of the system may become a significant possibility, the changes should be monitored as closely as possible and the boron feed level adjusted accordingly.

Minimum Allowable Containment Pressure vs BWST H_2O tempterature - TMI-2

Design External Pressure = -2.5 psi

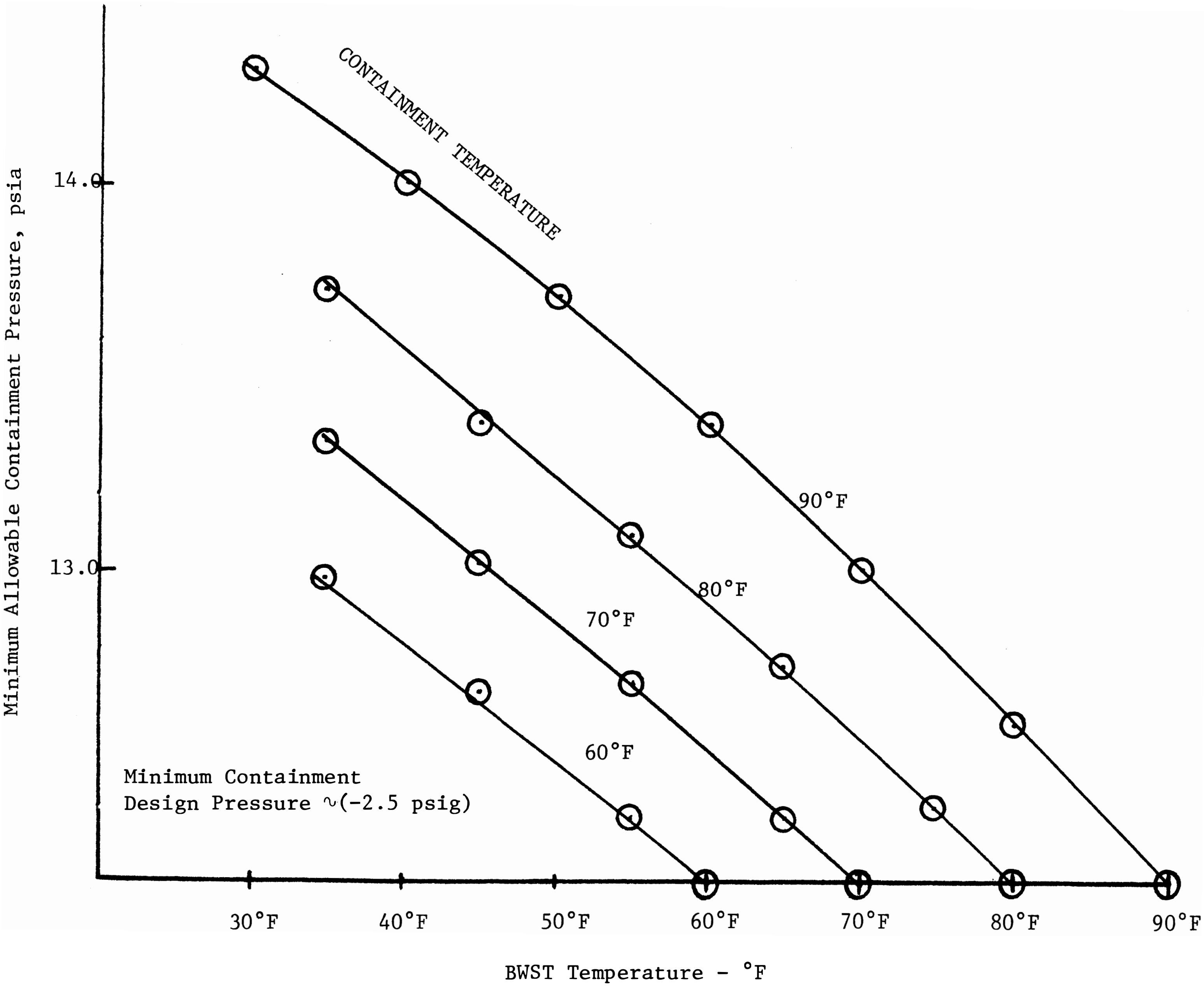


Figure 6.1

Percent of Full Power (2772 MWt)

Use Nominal Values for ANS-5.1 (1971) - infinite operation

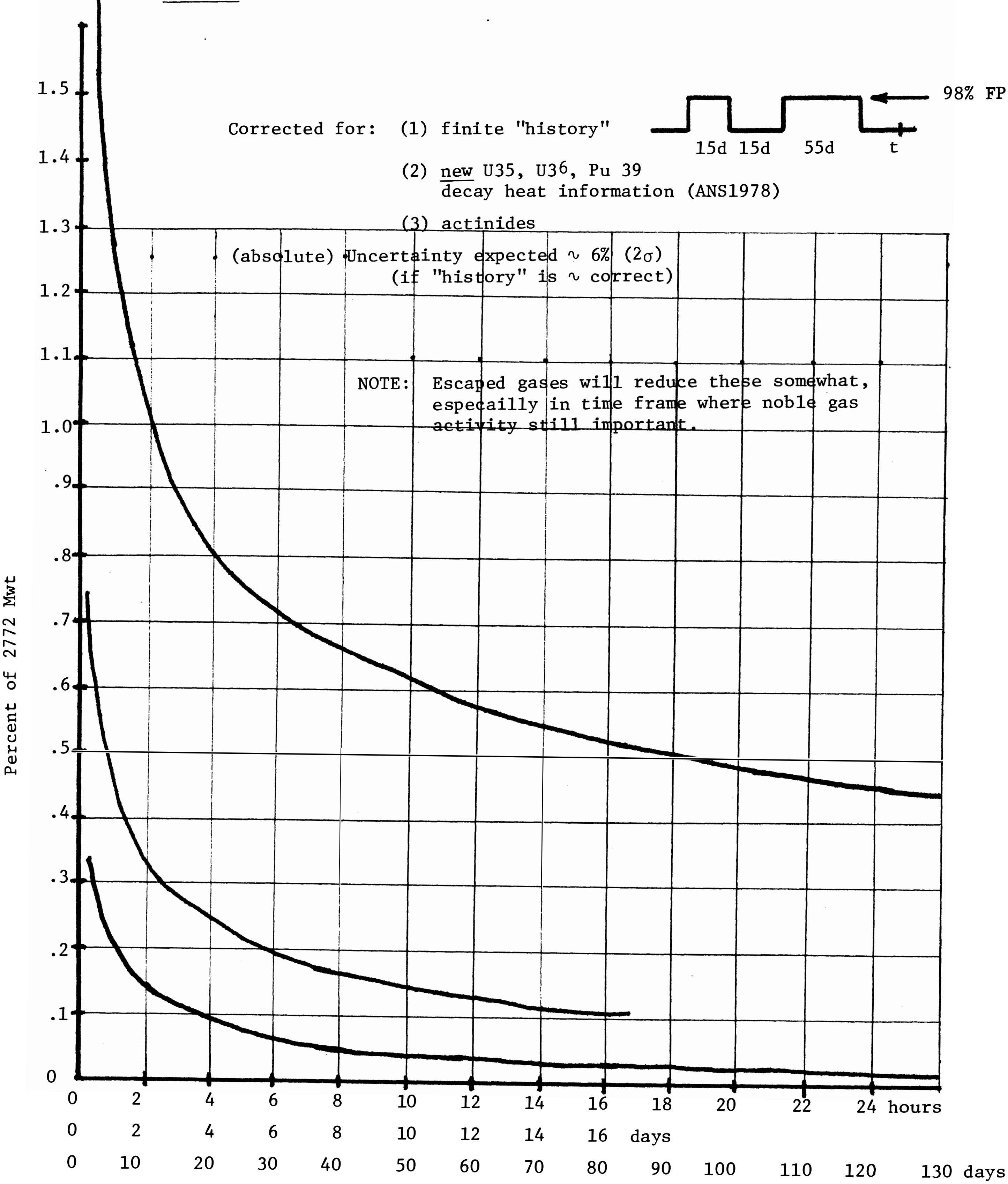


Figure 6.2

7.0 Contingency for Loss of Natural Convection

By the time the natural circulation cooling mode is achieved at TMI-2, decay heat levels will be sufficiently low that time will be available to consider alternative ways of cooling the core if natural circulation cooling is lost (on the order of hours would be required to reach saturation temperature in the RCS, and several more hours would be required to boil away water above the core elevation). Some of these alternatives are discussed in the following subsections.

7.1 High Reactor Pressure

If reactor pressure is high enough to provide adequate net positive suction heat (NPSH) for the reator coolant pump (RCP), the quickest and most obvious alternative would be to jog and/or run one RCP. Since the most likely cause of natural circulation stoppage is some type of flow blockage (e.g., gas collection at some vital location such as the upper loop of the "candy cane") this action might sweep the bubble or other blockage to a less critical location permitting natural circulation to resume.

7.2 Low Reactor Pressure

If reactor pressure is not high enough for the alternative above, then an attempt should be made to increase pressure. Pressurizer heaters and/or an increase in flow from a pressure controlled charging pump should be attempted. The pressure increase by itself might cause resumption of natural recirculation flow by decreasing the size of any collection of gas that might be blocking natural circulation and allowing it to resume. If not, when sufficiently high primary system pressure is reached the first alternative above can be attempted.

7.3 Reflux Boiling

Reflux boiling is the production of saturated or slightly superheated steam in the core (thereby cooling the core), circulation of that steam to the steam generator by a combination of diffusion and pressure differentials in the system, and condensation of the steam in the steam generator tubes by transfer of heat to the secondary water through the tube walls.

If a gas bubble large enough to stop natural recirculation is formed, that bubble is also large enough to greatly inhibit condensation efficiency in the steam generator(s). Thus the core temperature/ pressure needed to achieve a stable "reflux" cooling cycle might be unacceptably high (an analytical basis should be provided before this cooling mode is attempted). The only obvious ways to improve the condensation efficiency are: 1) cool the steam generator - if in the steaming mode, possibly the mode could be changed to the flooded mode; 2) lower primary system water level to increase the condensation surface available to the primary steam. (This would be of questionable value since the steam would have to penetrate a considerable distance down very small tubes to utilize the added surface area.) It is not clear how effective this penetration would be, but again analyses should be provided before this mode is tried. Lowering water level would be an unlikely manuever to try under the poorly instrumented conditions likely by the time this situation might arise. In summary, core cooling by refluxing might be possible, although it seems unlikely and we do not recommend wasting time trying to optimize conditions to achieve this mode.

7.4 RHR System

Core cooling utilizing the existing RHR or utilizing the skid mounted RHR as a backup could be attempted in the event natural circulation is not available. Pressure should be increased as much as possible within the RHR range to minimize potential RHR pump NPSH problems and gas evolution problems. Pressure could be slowly decreased after initiation of RHR to minimize leakage to the environment. Since RHR suction from and discharge to the primary system is at a lower elevation than the top of the pressure vessel and upper loop of the "candy cane" where gas bubbles are most likely to collect, those cooling modes should still be possible even after a bubble has stopped natural recirculation flow. In any event, water returned to the vessel should be hot enough to avoid a thermal shock problem at the vessel welds (temperature can be controlled by controlling cooling in the RHR heat exchanger, i.e., reduce secondary water flow to the exchangers.)

7.5 Contingencies

If primary system pressure and level indication is available, and if controlled venting capability is retained, HPI can be used intermittently to provide makeup flow for boiloff cooling for operation over a wide pressure range. (Heated suction flow should be provided to avoid NDT problems on the vessel welds). Flow to containment would be small by the time this situation arises - on 04/12/79 calculated boiloff flow is only $\approx$25 gpm to remove all decay heat and will decrease to half that value by about 05/04/79. This amount of water could be allowed to accumulate for an extended period in containment, or could be removed and possibly recycled through HPI (if leak tight equipment can be made available).

If controlled venting is not possible but level indication is available, intermittent makeup flow (as above) can be provided to keep the core covered, but system pressure will go to the pressurizer safety valve setpoint where venting will occur. Primary system pressure indication would be desirable, but not necessary, as it could be inferred from HPI injection flow and pressure as discussed immediately below.

For the more degraded conditions discussed below, it would be desirable to provide HPI with a throttled output, bypass return line, or a surge tank controlled pressure so that injection flow and pressure can be carefully controlled and monitored. Rate of pressure change versus flow can be used to calculate steam and gas voids in the system. If experience has shown no gas evolution, then this method can be used in a closed system to determine volume that is not filled, i.e., a level indication. Injection flow rates would have to be low so that pressure measurement of the injection flow would closely approximate primary system pressure, i.e., little pressure loss in the injection piping inside containment where pressure losses cannot be measured.

If level indication is lost and controlled venting is possible, HPI flow can be adjusted along with venting rate to keep the primary system nearly full at any desired pressure. HPI flow rate vs. pressure change can be used to calculate "steam" (non filled) volume, and control can be based on this calculated volume, as discussed above.

If controlled venting is not possible, the above cooling mode can still be accomplished but the pressure would have to be at the safety valve set pressure. This could be accomplished by slowly increasing HPI flow rate while monitoring HPI discharge pressure vs flow rate until calculations as discussed above show the system to be nearly solid (i.e., full). Periodic repetition of the slow HPI flow increase procedure could be used to "benchmark" times when the system was known to be full. At 20 gpm or less boiloff rate, these exercises could be very infrequent - it would take a full day (24 hours) to boil away 1/2 the primary system inventory.

8.0 Radiological Considerations

The potential radiological consequences of loss of let-down flow, use of the RHR system, and steam generator leakage are consideration in this section.

8.1 Purification Demineralizer

Substantial radioactivity may have built up on the purification demineralizer such that if the flow is stopped, the bed will heat up due to decay heat. Rough calculations indicate that the relief valve will lift and discharge small amounts of water and possibly traces of steam to the Reactor Coolant Holdup Bleed Tanks (RCHBT) if the system is isolated. As long as some flow is maintained, there should not be any steam. If water and traces of steam are relieved to the RCHBT, the offsite consequences should be nil because these tanks vent to the waste gas vent header which can be placed at a negative pressure by venting back to containment. Procedures should exist for venting the waste gas vent header back to containment should this become a problem.

Heat in combination with radiation damage could result in degradation of the demineralizer resin. Radiation degradation that would lead to physical property changes should not occur within the next few weeks. If there has been more fuel degradation than the 0700 3/30 primary coolant sample indicated, it is possible that the resins could physically break down. This could lead to plugging of the demineralizer lower retention screens, thus blocking flow. It is our understanding that the valve operator for the inlet to the purification demineralizer has failed thus making realignment of letdown flow difficult. We recommend that procedures be considered for flow blockage in the purification system.

The radiation exposure for the demineralizer resins will also decrease their ability to ion-exchange. It is expected that decreased ion-exchange is now taking place and that radioactivity could leach off of the resins in the future. This should not be a significant concern because downstream components are heavily shielded; however, local radiation levels in this area of the plant could increase.

8.2 RHR System

If it is necessary to use the RHR system, iodine releases from leakage could occur. A method to minimize radioiodine releases would be to install a skid mounted charcoal filter system in the RHR room. Such units already exist and could be lowered through the RHR pump room equipment hatch. This has been considered and installation prior to reactor systems operation which could lead to a likelihood of RHR system operation is being pursued.

The design flow rate of air from the RHR pump rooms is only 350 SCFM. This is a small flow and a small charcoal filter system could be installed in the exhaust ducting if room exists. This would supplement the large Auxiliary Building Filter Units which may become degraded with time. A small fresh charcoal filter would reduce iodine releases by at least a factor of 100 if the RHR had to be used.

8.3 Steam Generator Leak

Consideration should be given to methods of detecting "A" steam generator leakage with a flooded secondary side condition. Procedures should exist similar to those which follow relative to the operations of the "B" steam generator for minimizing releases should leakage occur--e.g., use of condensate polishers on recirculation to the hotwell and maintaining the condenser at a pressure negative to the condenser circulating cooling water.

The secondary coolant presently in the "B" steam generator is contaminated due to the initial primary to secondary leakage which occurred on March 28, 1979. Under the proposed operating conditions the secondary coolant pressure will be maintained at a value greater than the primary system pressure such that if steam generators leakage flow paths are available, the highly contaminated primary coolant will not enter the secondary coolant. However, it is expected that transients of short duration may occur such that a reverse pressure gradient could introduce additional radioactivity into the secondary coolant. To alert the system operators to such a condition, indicators and alarms for pressure and radioactivity in the secondary coolant have been provided. These indicators should alert the operator to such an adverse condition so that prompt corrective action could be taken prior to significant additional contamination of the secondary coolant.

The steam generator secondary coolant system and secondary services closed cooling water system should be periodically sampled and analyzed to determine if heat exchangers are starting to leak. Since laboratory analysis provides a higher level of accuracy than the continuous radiation monitors in terms of sensing increases in radioactivity concentrations, samples should be taken at a routine frequency of at least weekly or at any time there are indications that possible leakage may be occurring, e.g., increase in the steam generator closed cooling loop surge tank levels.

Design features should be provided to monitor and control radioactive effluents from the secondary coolant system to other less contaminated portions of the facility, e.g., secondary services closed cooling water system (SSCCWS) and turbine building floor drains. A radiation monitor should be installed on the SSCCWS to alert the operator to a leak from the secondary cooling system into the SSCCWS.

Potential leakage paths from the secondary cooling system to the turbine building environment must be monitored and controlled. The relief overflow from the surge tank is a relatively high probability source for a spill from the secondary cooling system. To prevent the uncontrolled overflow from the system the licensee has provided a holding tank to contain an overflow. Indication and alarm of high surge tank level would alert the operator to a potential overflow condition to permit corrective action prior to overflowing the holding tank.

Leakage may also occur at various mechanical connections in the secondary cooling system. To the extent practical, locations where leakage is likely, e.g., drain valves, valve stems in valves with a known leakage history, and turbine stop valves, should be isolated from the turbine floor drain system. Isolation in many areas could be accomplished by plugging local floor drains and/or by the addition of curbing to contain spillage. Contingency plans should be provided to transfer collected contaminated liquids to appropriate waste systems. It should be noted that once contaminated water enters the turbine building floor drain system, it is very difficult to prevent its flow to the river.

Isolation between the secondary cooling system and services systems, e.g., nitrogen and demineralized water supplies to the surge tank, should be provided to prevent back flow of contaminated water. At least a soft seated isolation valve (or equivalent) and a check valve should be provided.

Gaseous effluents from this system should be negligible. The noble gas inventory in the "B" steam generator is negligible because the steam generator was vented. Airborne radioiodine releases should also be negligible because the secondary cooling system is not vented (a nitrogen blanketed surge tank) and the low secondary cooling system temperature (110°F) results in a low air/water partition factor which reduces the volatility of the radioiodine. The pressure regulating valve from the surge tank also vents to atmosphere through a charcoal filter.

The staff has estimated the radiation dose rates due to piping in a system filled with diluted secondary water, e.g., such as would occur when the B steam generator is placed in a water solid condition. Table 8-1 shows the results ORNL obtained for calculated dose rates at various distances from various sizes of pipe. As a check, the staff performed hand calculations of the dose rate at one meter from the various sizes of pipe. The hand calculations gave dose rates of 5-11 mr/hr, which is good agreement with the ORNL results.

The source term in 6.5×10^7 cc of water was:

I-131	1.4 μCi/cc
Cs-134	8.6×10^{-3} μCi/cc
Cs-136	1.1×10^{-2} μCi/cc
Cs-137	3.4×10^{-2} μCi/cc

This source term is diluted to a total water volume of 1.7×10^8 cc. The piping is infinite lengths of Schedule 40 with nominal sizes of 12", 16", and 20". The dose rates are due almost entirely to I-131 (98%).

Table 8-1

DOSE RATES VERSUS DISTANCE FOR
INFINITE LENGTHS OF SCHEDULE 40 PIPE
OF VARIOUS NOMINAL SIZES (mr/hr)*

Distance from Pipe Center (feet)	12"	16"	20"
Contact	53	50	45
2.5	10	12	13-1/2
4.5	6	7	8
10	3	3	4
18	2	2	2

* Radionuclide concentrations in water

I-131	5.3×10^{-1} μCi/cc
Cs-134	3.3×10^{-3} μCi/cc
Cs-136	4.2×10^{-3} μCi/cc
Cs-137	1.3×10^{-2} μCi/cc

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

APPENDIX A

APR 13 1979

MEMORANDUM FOR: Roger J. Mattson, Director, Division of Systems Safety, NRR

FROM: R. O. Meyer, Reactor Fuel Section Leader, Core Performance Branch, Division of Systems Safety, NRR

SUBJECT: CORE DAMAGE ASSESSMENT FOR TMI-2

Attached is our assessment of the core damage at TMI-2 for use in the SER for natural circulation. It represents our independent evaluation of the facts available and of the industry/vendor/licensee analysis, which we have heard in several briefings.

An earlier estimate of fuel damage was made by Rubenstein et al, and a recent meeting was held at NRC with industry experts. Memoranda describing those evaluations are attached to this document.

Ralph O. Meyer

Ralph O. Meyer, Section Leader
Reactor Fuels
Core Performance Branch
Division of Systems Safety

Attachment:
As stated

cc: R. Tedesco
K. Kniel
P. Check
C. Berlinger
D. Crutchfield (FOIA File)
D. Houston
M. Tokar
V. Stello
B. Grimes
G. Knighton
J. Voglewede
D. Powers
PDR

CORE DAMAGE

A. Introduction

For the usual analysis of hypothetical accidents, initial core conditions are assumed and consequences are calculated. This would involve complex thermal-hydraulic calculations and fuel behavior analyses. At Three Mile Island, however, some of the consequences are known (i.e., some information on fission product release, hydrogen generation, and instrument readings is available), so we will use "reverse engineering" as our principal method of backing out an assessment of core damage.

We start with the assumption that the core was uncovered and allowed to heat up for significant periods of time. Figure 1 shows the system pressure history for March 28, which includes three periods of significant uncovery. The periods of uncovery correspond approximately with the major periods when system pressure was below the saturation pressure. We will assume that the first core uncovery began shortly after 92 minutes into the accident at which time excore ion chambers show a response spike corresponding to the loss of water shielding. Although the two later periods of uncovery may have produced additional core damage, we will focus on the first period because decay heat was larger then and because that period produced the large radiation instrument reading (at 150 minutes) in the containment indicating major fuel damage.

Because the fuel damage to be discussed below is so extensive, we will

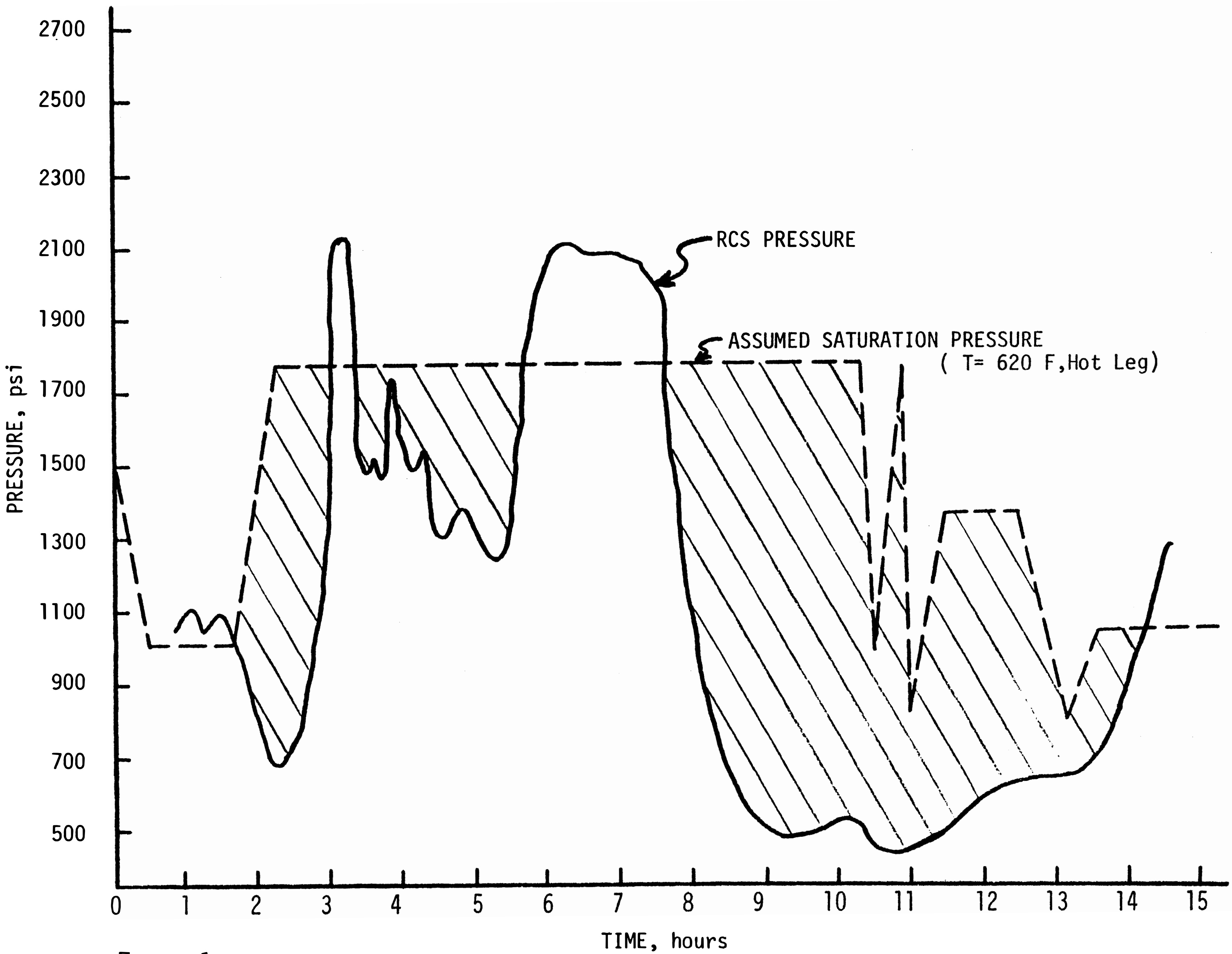

RCS PRESSURE
ASSUMED SATURATION PRESSURE
(T= 620 F,Hot Leg)
PRESSURE, psi
2700
2500
2300
2100
1900
1700
1500
1300
1100
900
700
500
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
TIME, hours

FIGURE 1.

conclude without demonstration that virtually all of the fuel rods in the core failed in the sense of experiencing defects large enough to release gas. Furthermore, the rods probably failed by a LOCA-like ballooning-and-rupture mechanism. Because of the massive oxidation that followed, the mode of failure is probably immaterial.

As a point of reference, Table I lists melting temperatures of the various materials used in the fuel system.

B. Fuel Rods

Fission product and hydrogen measurements at TMI-2 give important clues about the condition of the fuel rods. We will deal with fission product releases first.

Air and water samples containing fission products have been analyzed. While we have analyzed both for indications of fuel conditions, we have concentrated on the Xe-133 concentration in the air sample. This isotope was selected for analysis for several reasons: (a) it is a noble gas and will not react, plate out or condense, (b) it has a relatively long half life (5.29 days) and a high production rate (6.8 atoms per 100 fissions) and therefore will be abundant thus reducing measurement errors, and (c) fission product release correlations are much better established for noble gases than for other fission products.

Bettis (BAPL) has evaluated the Xe-133 activity and concluded it is equivalent to 31% of the total core inventory. We have independently checked this calculation (but, of course, not the sample activity) and agree (31.5%).

TABLE 1. MELTING TEMPERATURES

MATERIAL	TEMPERATURE, ^{o}C	TEMPERATURE, ^{o}F
UO_2	2805	5080
ZIRC-4	1850	3362
ZrO_2	2715	4919
INCONEL 718	1260-1286	2300-2346
304 SS	1399-1421	2550-2590
Al_2O_3-B_4C	2030	3686
Ag-In-Cd	800	1472
UO_2-Gd_2O_3 *	2750	4982

* Two fuel assemblies contained gadolinia test rods.

Fission products including gases are normally retained by the UO_2 pellets. A normal pellet release to the fuel rod internal voidage is only 1 or 2% (even for a successfully terminated LOCA) so that a 30% release indicates additional release from fuel pellets not just a release of the gap activity.

Fuel pellet releases are strongly dependent on temperature, and Figure 2 shows a correlation of release versus temperature for Xe-133 (from a recent ANS-5.4 draft standard). The correlation, however, is for steady-state releases and we are dealing with a transient. Further errors are possible because of kinetics changes due to oxidation to U_4O_9 or U_3O_8. Nevertheless, it is a reasonable approximation and is consistent with recent short-time annealing experiments (private communication 4-10-79, R. A. Lorenz, ORNL) and earlier annealing work (G. W. Parker et al., ORNL-3981 - See attachment A).

Parker heated irradiated samples in a furnace for 5.5 hours. The samples had burnups ranging from trace to 4000 MWd/t (about the same as TMI-2). Parker measured releases of about 5% at 1600^oC, 15% at 1800^oC and 40% at 2000^oC with an uncertainty of about a factor of 2 in release. These experimental releases for conditions roughly similar to TMI-2, but for different isotopes, are in fair agreement with Figure 2.

Using Figure 2 we could conclude that (a) the fuel was uniformly heated (uniform in axial and radial directions) to about 1750^oC, or (b) 30% of the fuel melted while 70% remained below 1200^oC, or (c) any intermediate condition existed. Because of the core uncovery sequence,

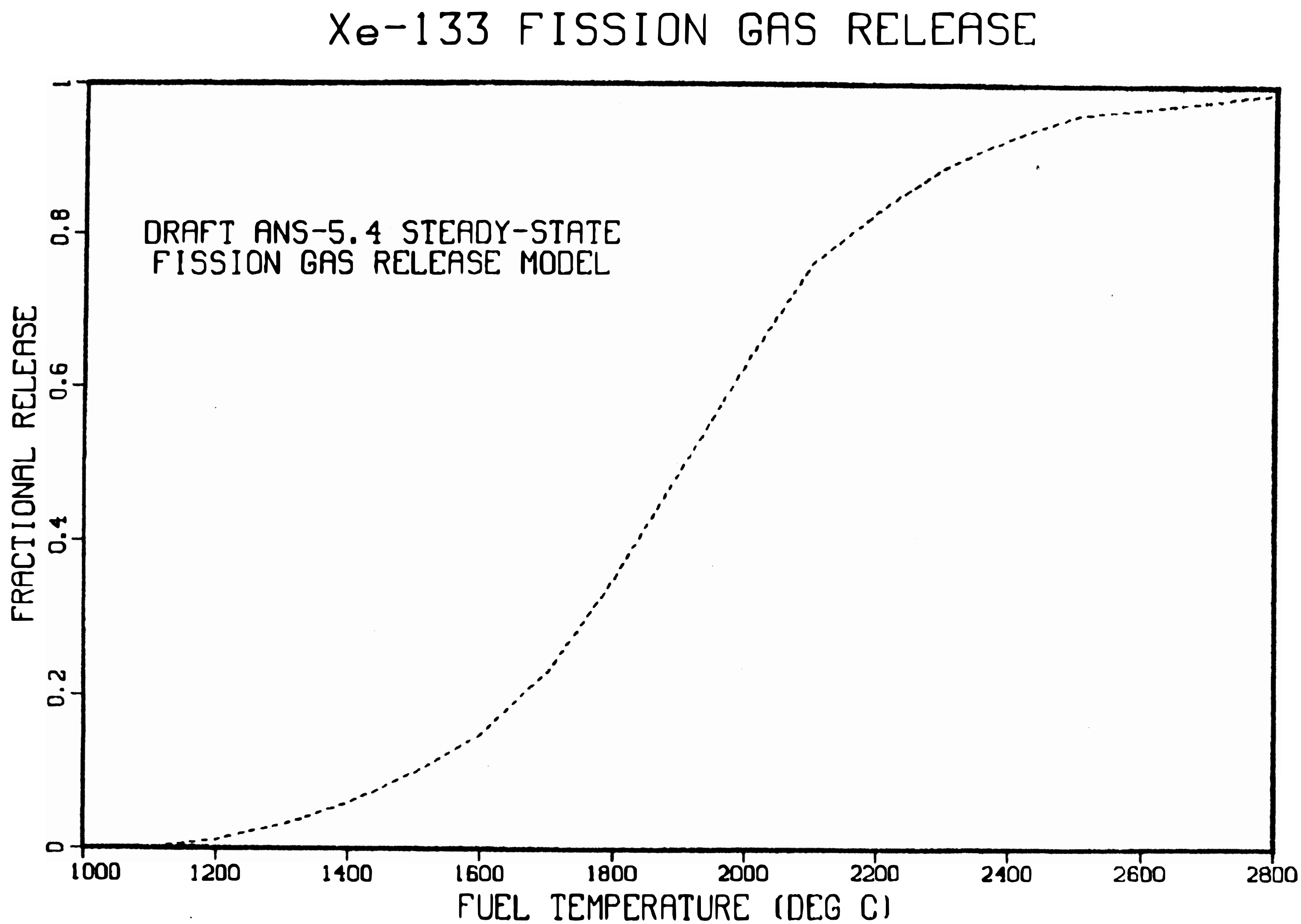

Xe-133 FISSION GAS RELEASE
DRAFT ANS-5.4 STEADY-STATE
FISSION GAS RELEASE MODEL
FRACTIONAL RELEASE
0
0.2
0.4
0.6
0.8
1
1000
1200
1400
1600
1800
2000
2200
2400
2600
2800
FUEL TEMPERATURE (DEG C)

FIGURE 2.

the fuel rods probably did not heat up uniformly in the axial direction. It is reasonable, however, to treat the fuel rods as isothermal in the radial direction because of the low heat flux. Figure 3 illustrates this point with a comparison of a full-power radial temperature profile and a decay-heat-power temperature profile.

There are physical limits on how hot the fuel can get during the periods of core uncovery because the fuel rods have a large heat capacity and a low heat generation rate. If one assumes zero heat removal (this would produce the most rapid heatup rate possible) during the first period of uncovery, the heatup rate is still fairly slow. Figure 4 shows the adiabatic temperature increase with time for the peak-power axial location, for the low-power ends of the rods, and for the average location. Since there must have been some heat removal thus further slowing down the temperature rise, pellet temperatures probably did not reach the melting point. Figure 5 shows the temperature changes with time for a surface heat transfer coefficient of 0.5 BTU/hr-ft^2-^{o}F, which is a very small value.

The results on temperature distribution are, therefore, not conclusive. It is unlikely that fuel temperatures were uniform and no lower than 1750^{o}C, and it is also unlikely that any fuel (UO_2) melting took place. The fuel, however, did get very hot compared with its normal operating temperatures.

Oxidation of Zircaloy by steam and the attendant decomposition of water provided the major source of hydrogen in the TMI-2 vessel and containment.

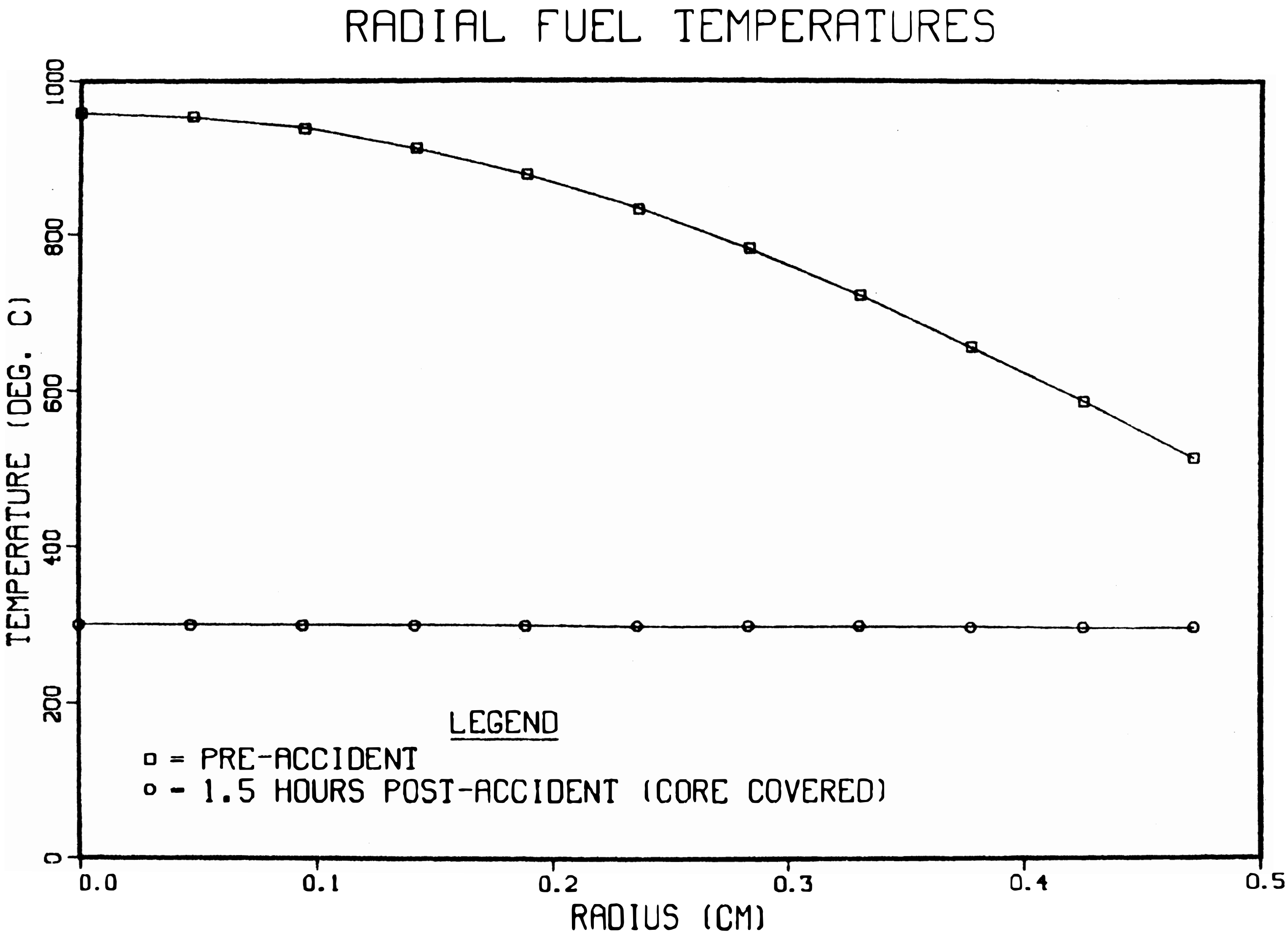

RADIAL FUEL TEMPERATURES
TEMPERATURE (DEG. C)
1000
800
600
400
200
0
0.0
0.1
0.2
0.3
0.4
0.5
RADIUS (CM)
LEGEND
□ = PRE-ACCIDENT
○ - 1.5 HOURS POST-ACCIDENT (CORE COVERED)

FIGURE 3.

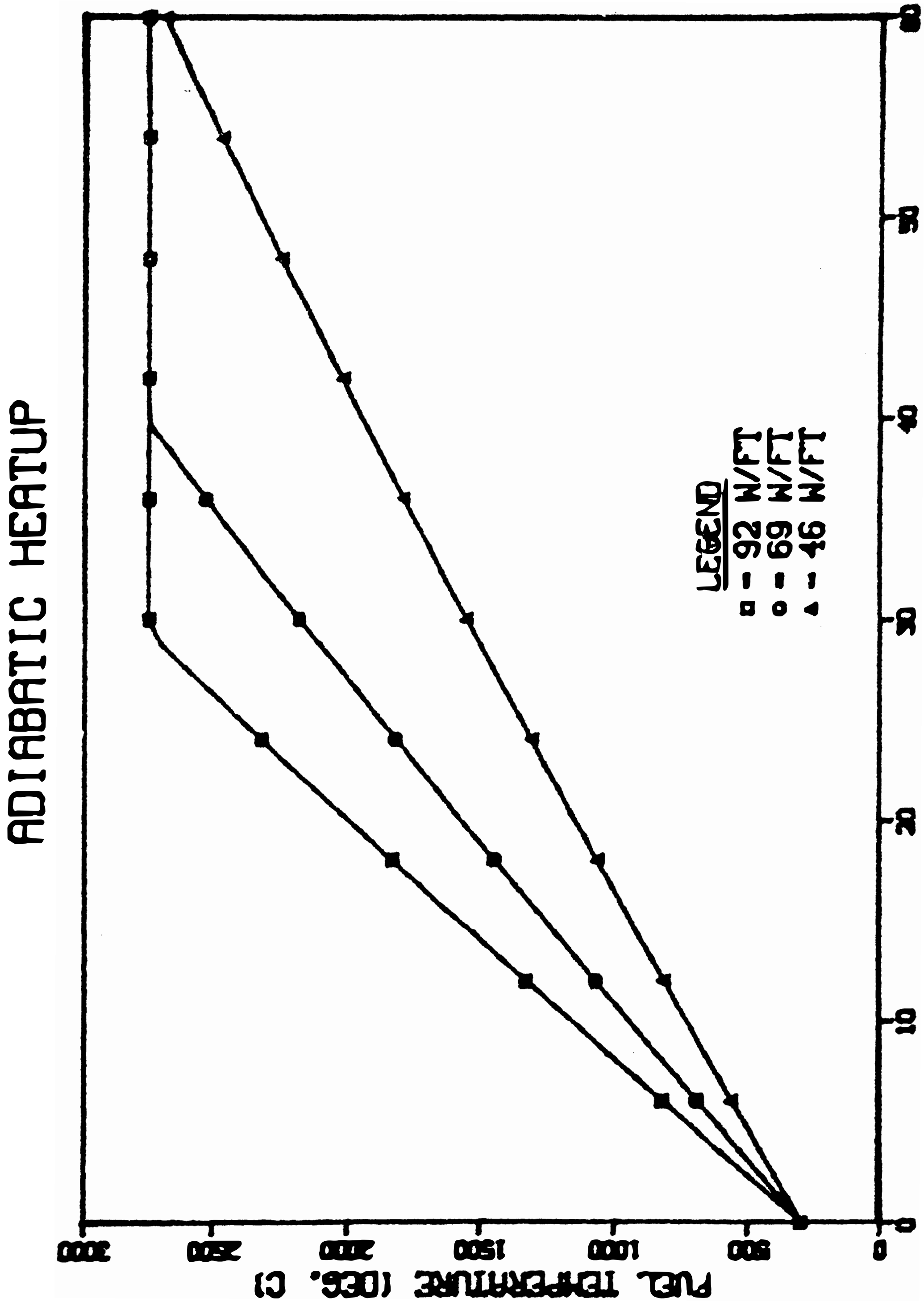
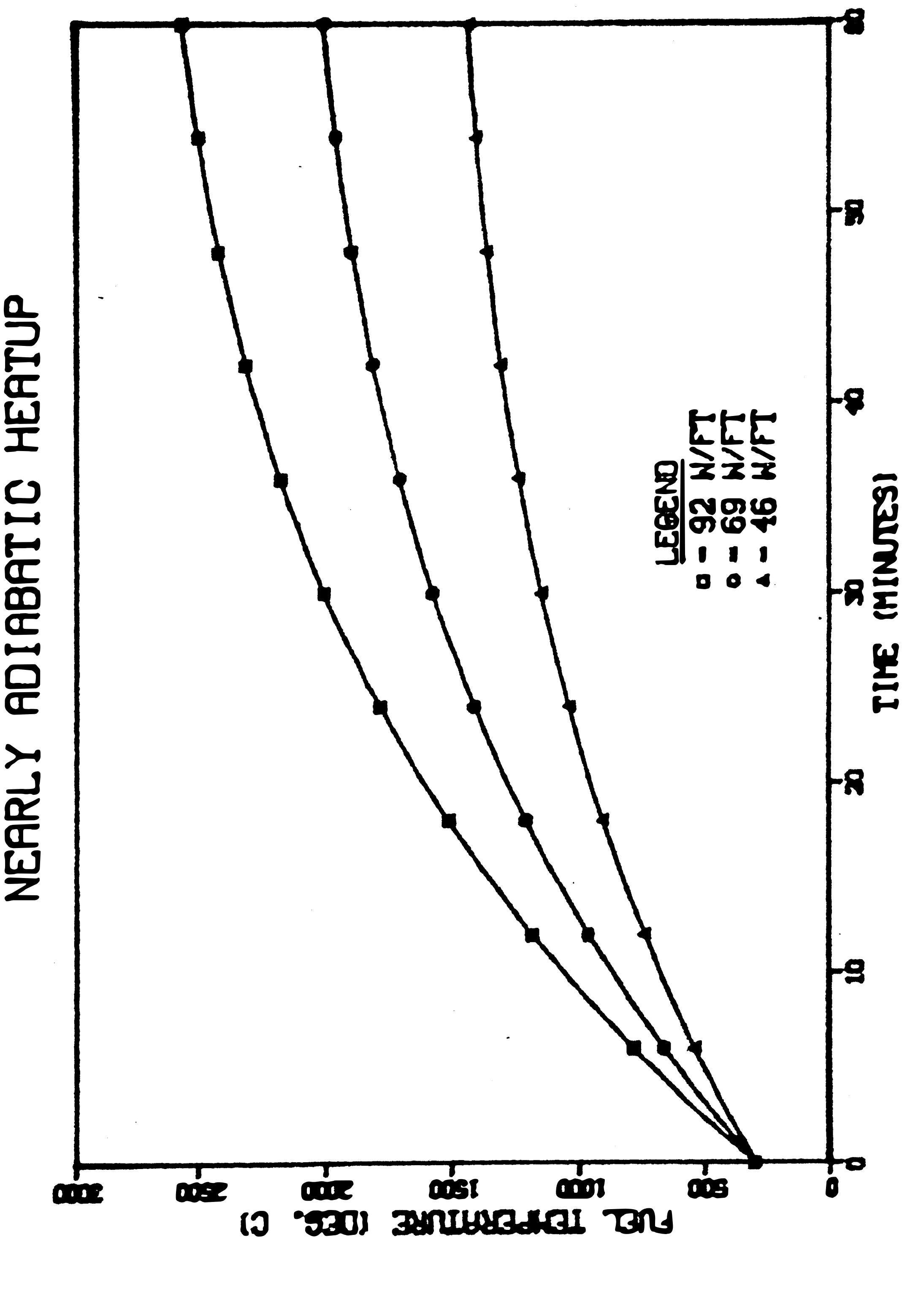

ADIABATIC HEATUP
FUEL TEMPERATURE (DEG. C)
0
500
1000
1500
2000
2500
3000
TIME (MINUTES)
0
10
20
30
40
50
60
LEGEND
□ = 92 W/FT
○ = 69 W/FT
△ = 46 W/FT

FIGURE 4.



NEARLY ADIABATIC HEATUP
LEGEND
□ = 92 W/FT
○ = 69 W/FT
△ = 46 W/FT
TIME (MINUTES)
FUEL TEMPERATURE (DEG. C)
0
500
1000
1500
2000
2500
3000
0
10
20
30
40
50
60

FIGURE 5.

The Containment Systems Branch has estimated the amount of hydrogen present in the plant (Attachment B) after the periods of core uncovery that caused fuel damage. They included amounts (a) consumed by the hydrogen explosion (226 lb mole), (b) remaining in containment after the explosion (80 lb mole), and (c) in the primary system bubble (76 lb mole), which was corrected for radiolysis.

Comparing the above amounts with the total amount of hydrogen that could have been produced if all of the Zircaloy in the fueled region reacted with water, we get 41%. As with the temperatures, an ambiguity exists. This could mean that (a) about 40% of the cladding wall thickness is uniformly oxidized throughout the core, or (b) 40% of the fueled region of the core has fully oxidized cladding, or (c) any intermediate condition exists.

Figure 6 shows the time required for total wall thickness oxidation as a function of temperature (Cathcart-Pawel correlation). It is clear from Figure 6 that complete oxidation is possible in cladding segments that reached temperatures of around 2000^{o}C during the period of core uncovery. It is also clear from Figure 6 that all of the cladding did not experience sustained temperatures of around 1750^{o}C else it would all have oxidized. This is further evidence that fuel temperatures were not uniform throughout the core, and that temperatures locally were very high.

Based on early estimates by the Analysis Branch of core uncovery, we will assume simplified uncovery histories shown in Figures 7 and 8 for the

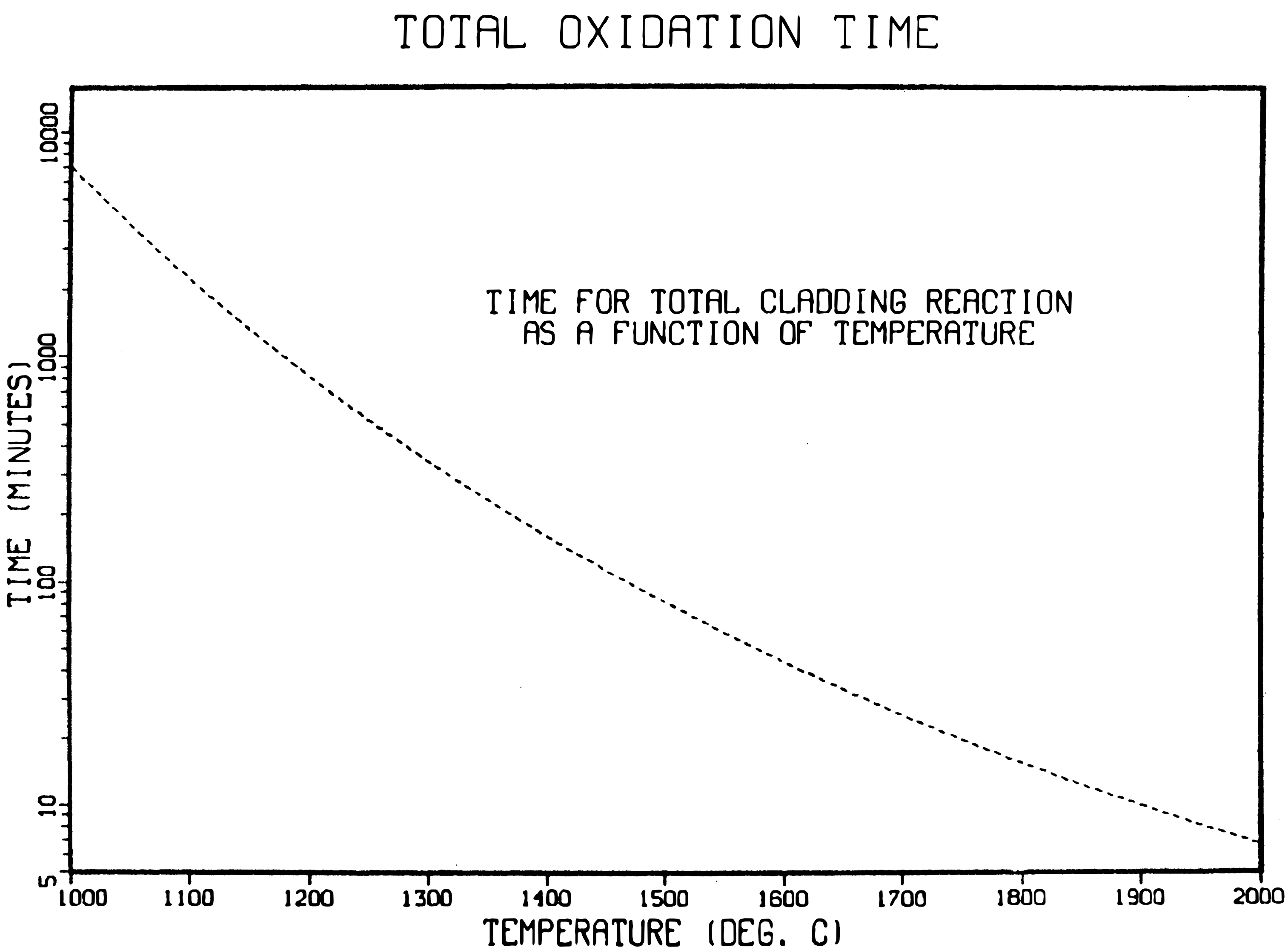

TOTAL OXIDATION TIME
TIME FOR TOTAL CLADDING REACTION
AS A FUNCTION OF TEMPERATURE
TIME (MINUTES)
10000
1000
100
10
5
1000
1100
1200
1300
1400
1500
1600
1700
1800
1900
2000
TEMPERATURE (DEG. C)

FIGURE 6.

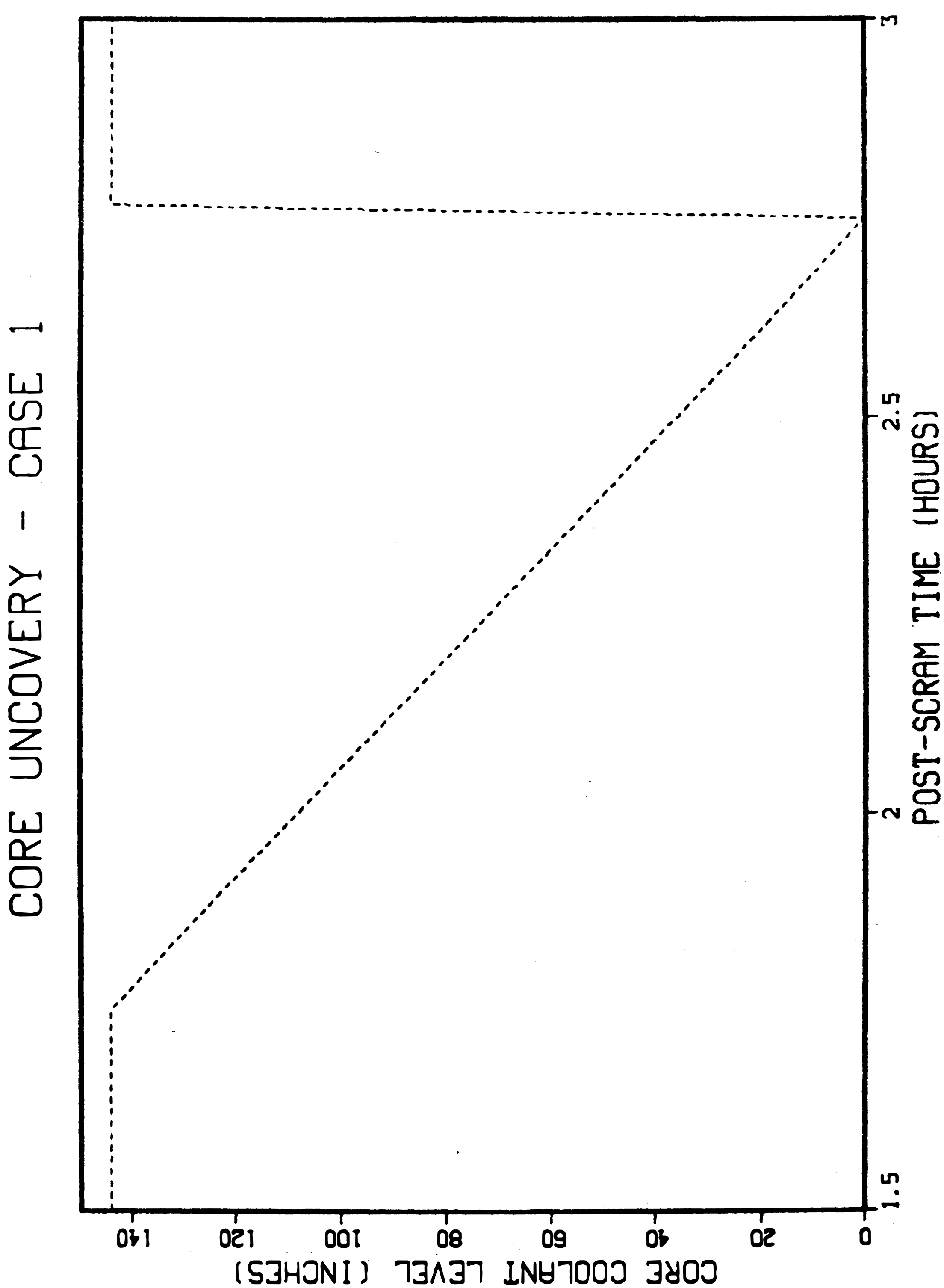

CORE UNCOVERY - CASE 1
CORE COOLANT LEVEL (INCHES)
0
20
40
60
80
100
120
140
POST-SCRAM TIME (HOURS)
1.5
2
2.5
3

FIGURE 7.

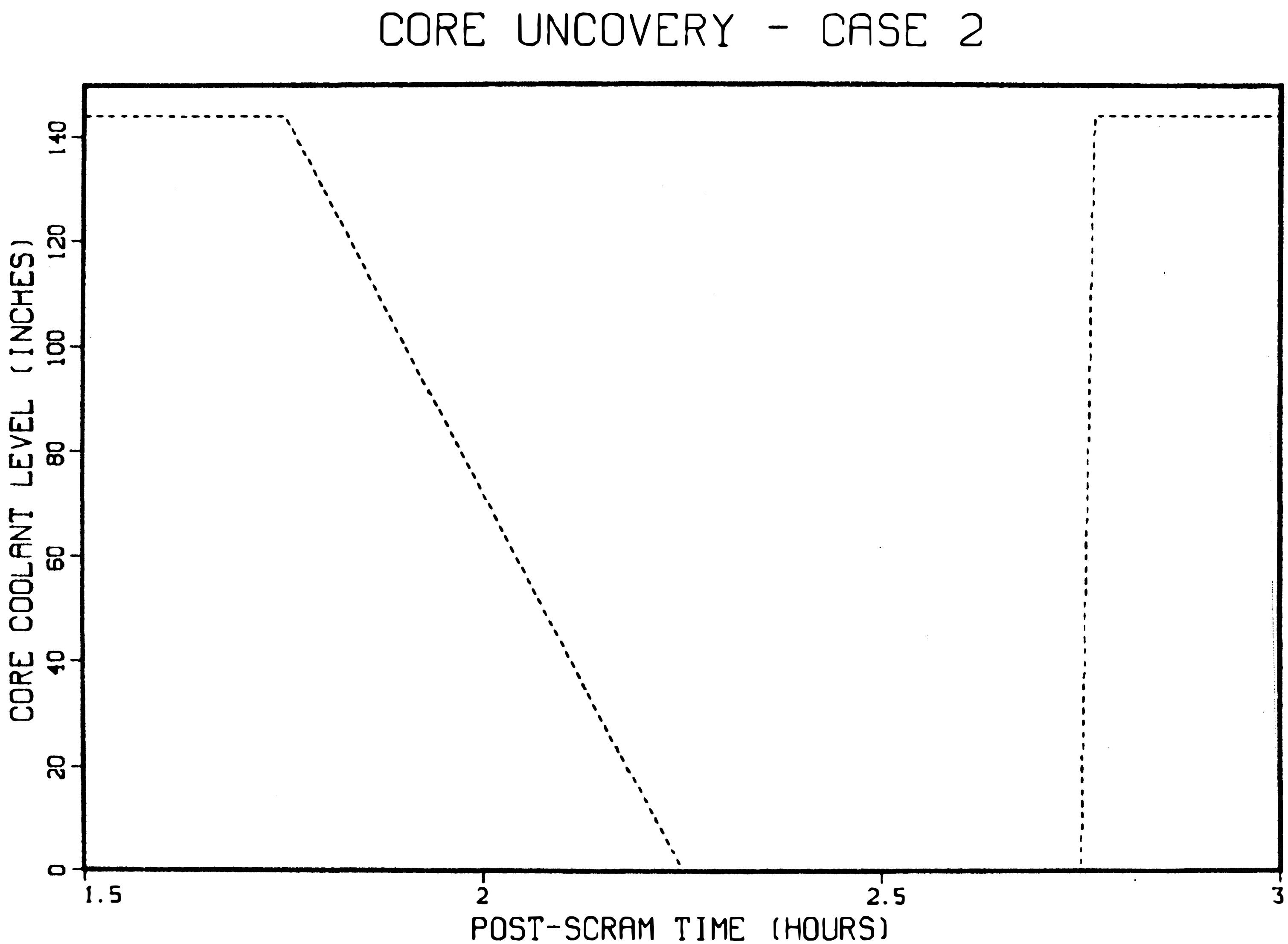

CORE UNCOVERY - CASE 2
CORE COOLANT LEVEL (INCHES)
0
20
40
60
80
100
120
140
1.5
2
2.5
3
POST-SCRAM TIME (HOURS)

FIGURE 8.

following calculation. Fuel that is covered will be considered to be cold (i.e., no cladding oxidation). Fuel that is uncovered will be allowed to heat up; fuel that heats up will be given a heat transfer coefficient that is adjusted such that the total integrated oxidation is 40%. These calculations give the oxidation distributions shown in Figures 9 and 10, and these distributions are insensitive to many of the assumptions that were made. Figures 9 and 10 thus are more probable distributions than 100% oxidation over 40% of the core or 40% oxidation over 100% of the core.

Figure 11 is a recent best-estimate embrittlement correlation (Kassner et al., ANL) that shows high-temperature fragmentation of quenched tubes at about 30% oxidation. Using this correlation, Figures 9 and 10 indicate that a fragmented region of about 5 ft. in height exists near the top of the core. It may well be right at the top of the core as a result of simplifications in our analysis. In any event, at least 4 to 6 ft. of intact (but partially oxidized) fuel rods remain standing at the bottom of the core.

Figure 12 shows fragmented Zircaloy cladding after oxidation in a simulated-LOCA test. Kassner (ANL-78-25 and ANL-78-49 reports that at high temperatures ($> 1250^{o}C$) many fragments are produced whereas at lower temperature the rod may simply break into two pieces. Inasmuch as TMI-2 temperatures were higher than $1250^{o}C$ and oxidation was severe, small fragments of the size shown in Figure 12 should be expected along with larger tube-like pieces.

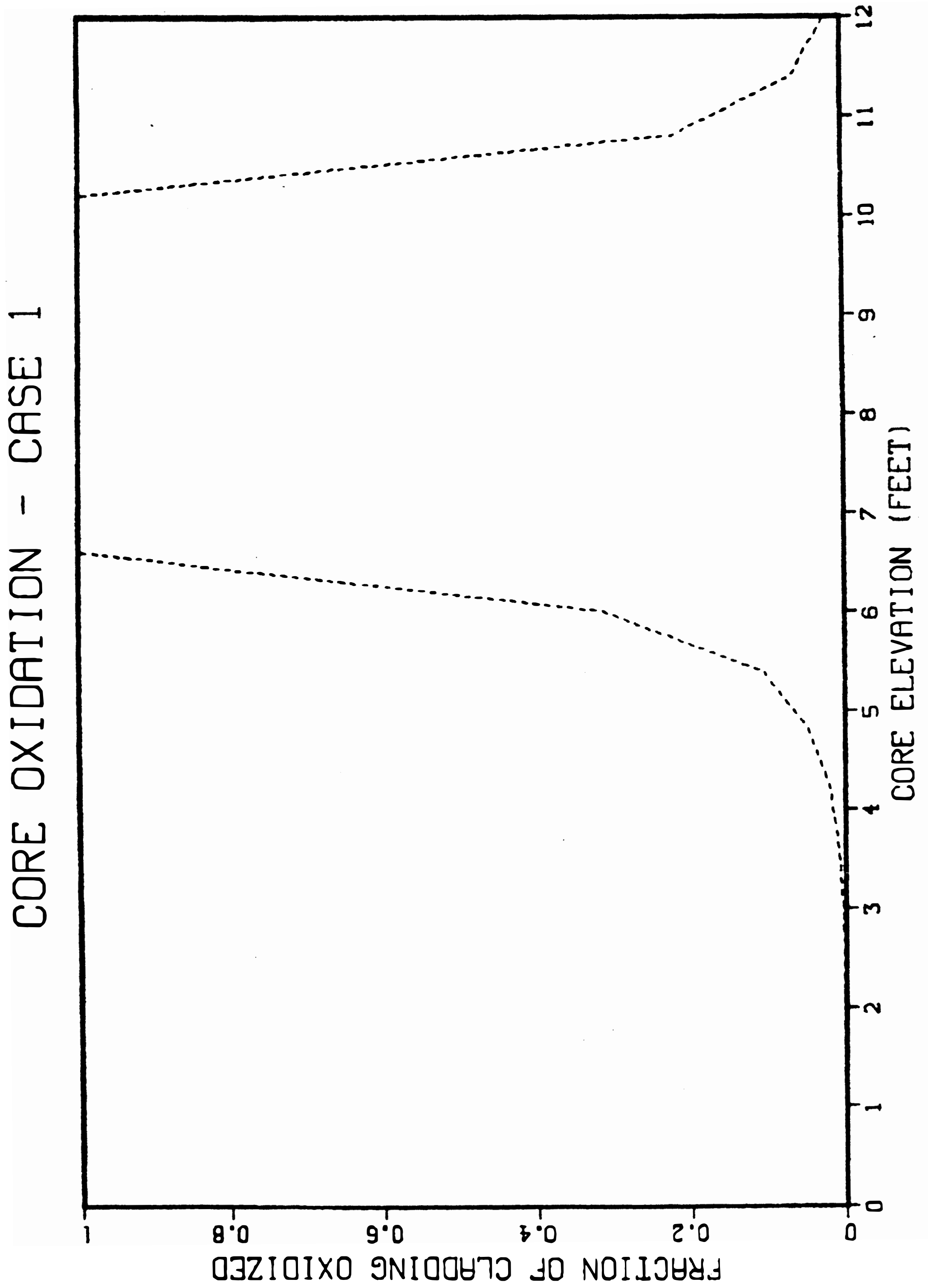

CORE OXIDATION - CASE 1
FRACTION OF CLADDING OXIDIZED
0
0.2
0.4
0.6
0.8
1
CORE ELEVATION (FEET)
0
1
2
3
4
5
6
7
8
9
10
11
12

FIGURE 9.

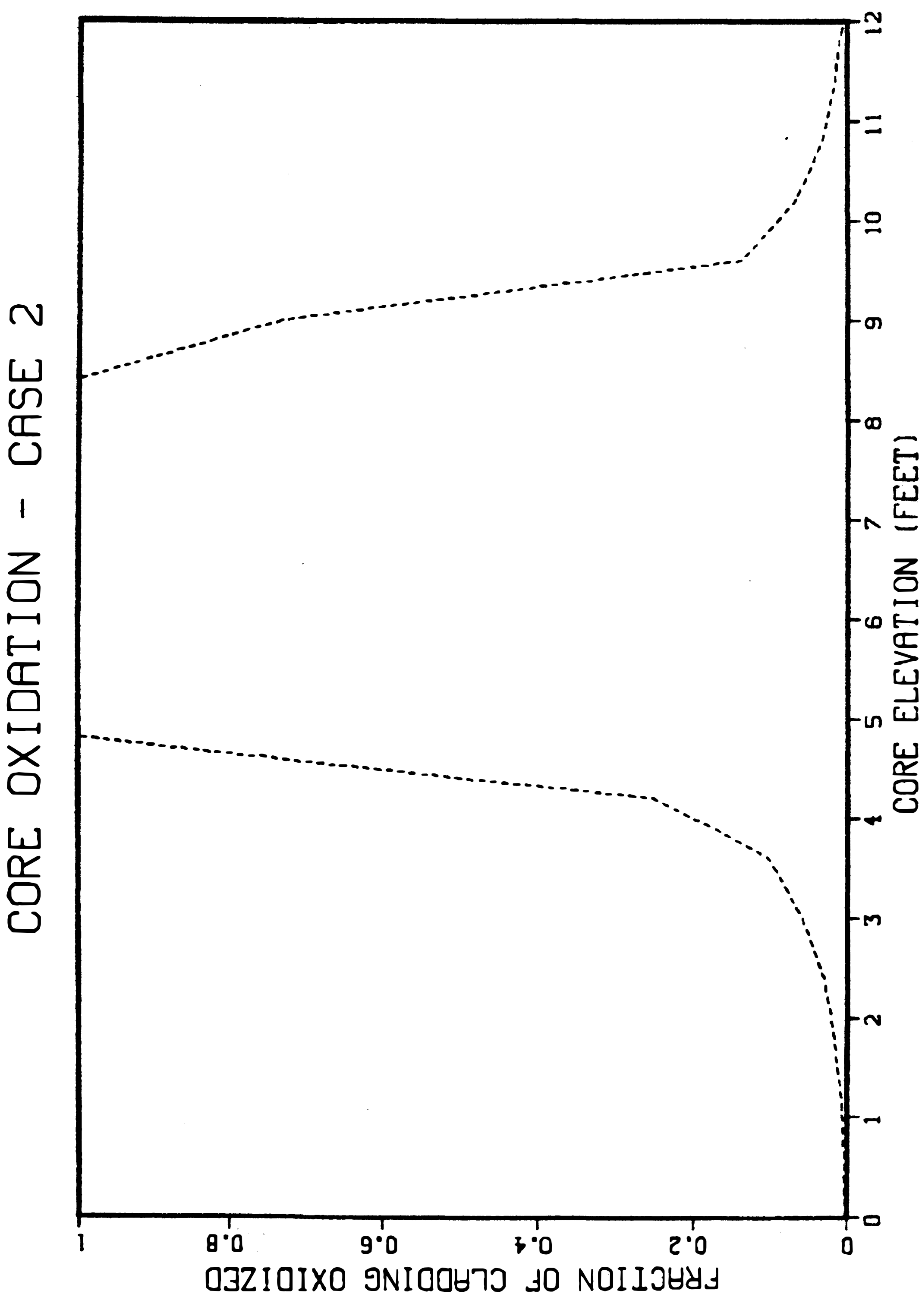
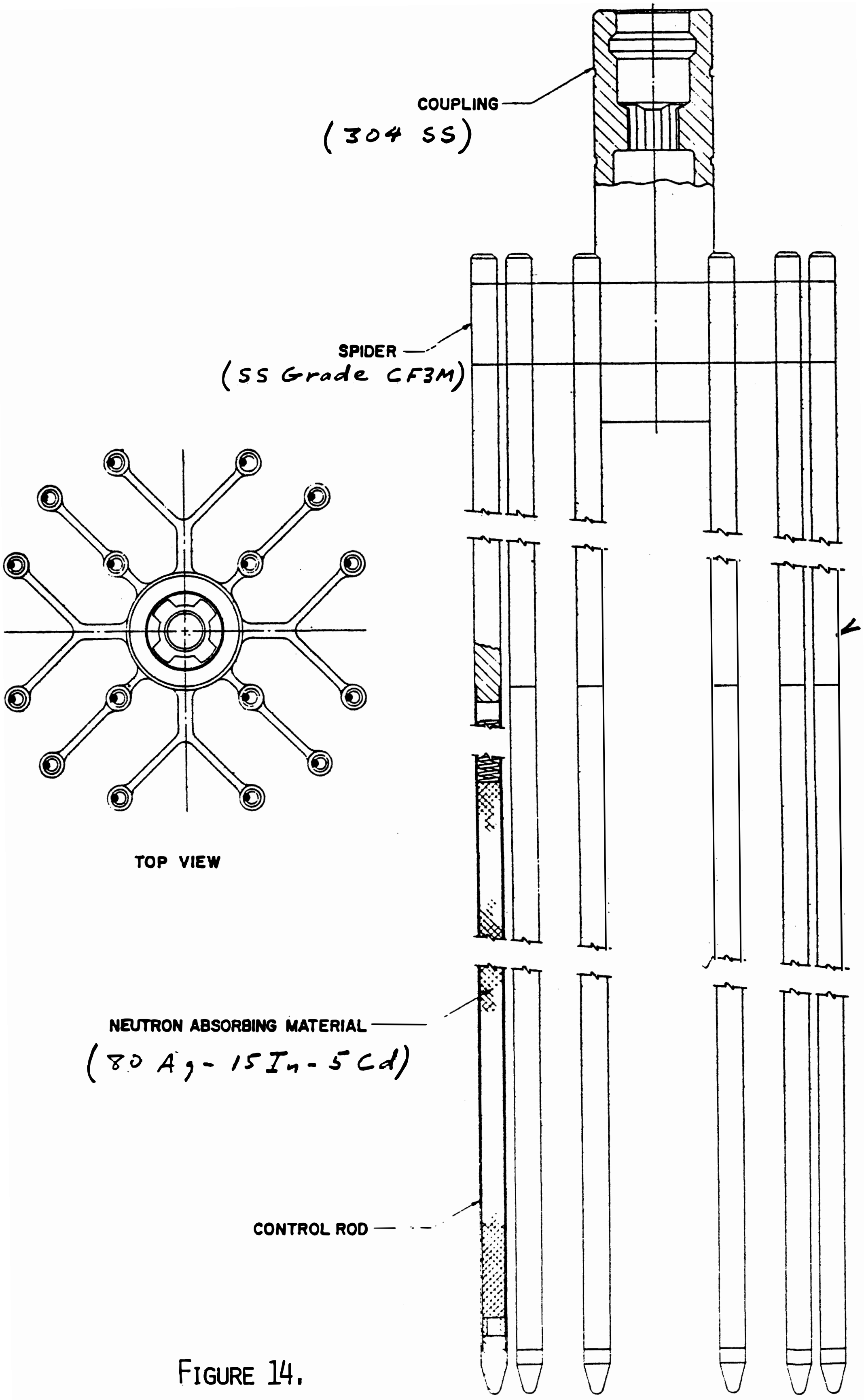

CORE OXIDATION - CASE 2
FRACTION OF CLADDING OXIDIZED
CORE ELEVATION (FEET)
0
0.2
0.4
0.6
0.8
1
0
1
2
3
4
5
6
7
8
9
10
11
12

FIGURE 19.



0.021 in.
304 SS
(0.440 in. O.D.)*

* Guide Tubes are
0.498 I.D. x 0.530 O

FIGURE 14.

Control Rod Assem

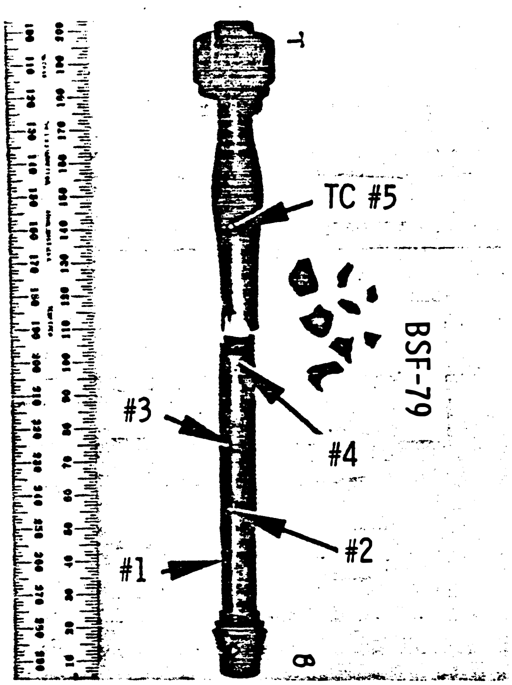


Fig. III.12

Zircaloy–4 Cladding after Thermal-shock Failure Showing Location of Thermocouples That Produced the Temperature–vs–Time Curves in Fig. III.10. ANL Neg. No. 306–78–223.

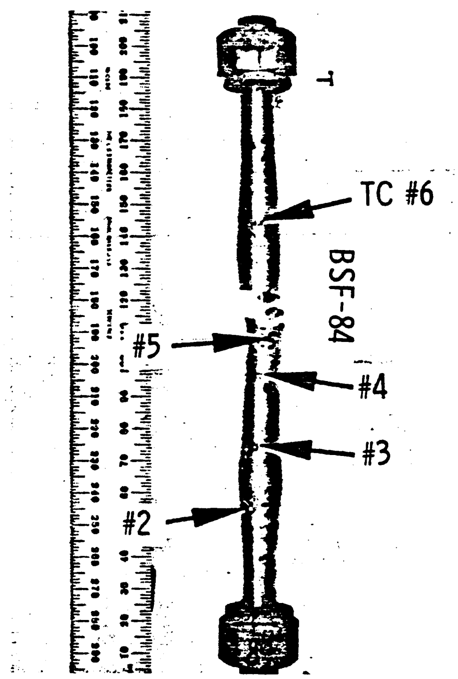


Fig. III.13

Zircaloy–4 Cladding after Thermal-shock Failure Showing Location of Thermocouples That Produced the Temperature–vs–Time Curves in Fig. III.11. ANL Neg. No. 306–78–224.

FIGURE 12.

FAILURE MAP FOR ZIRCALOY-4 CLADDING BY THERMAL SHOCK OR NORMAL HANDLING

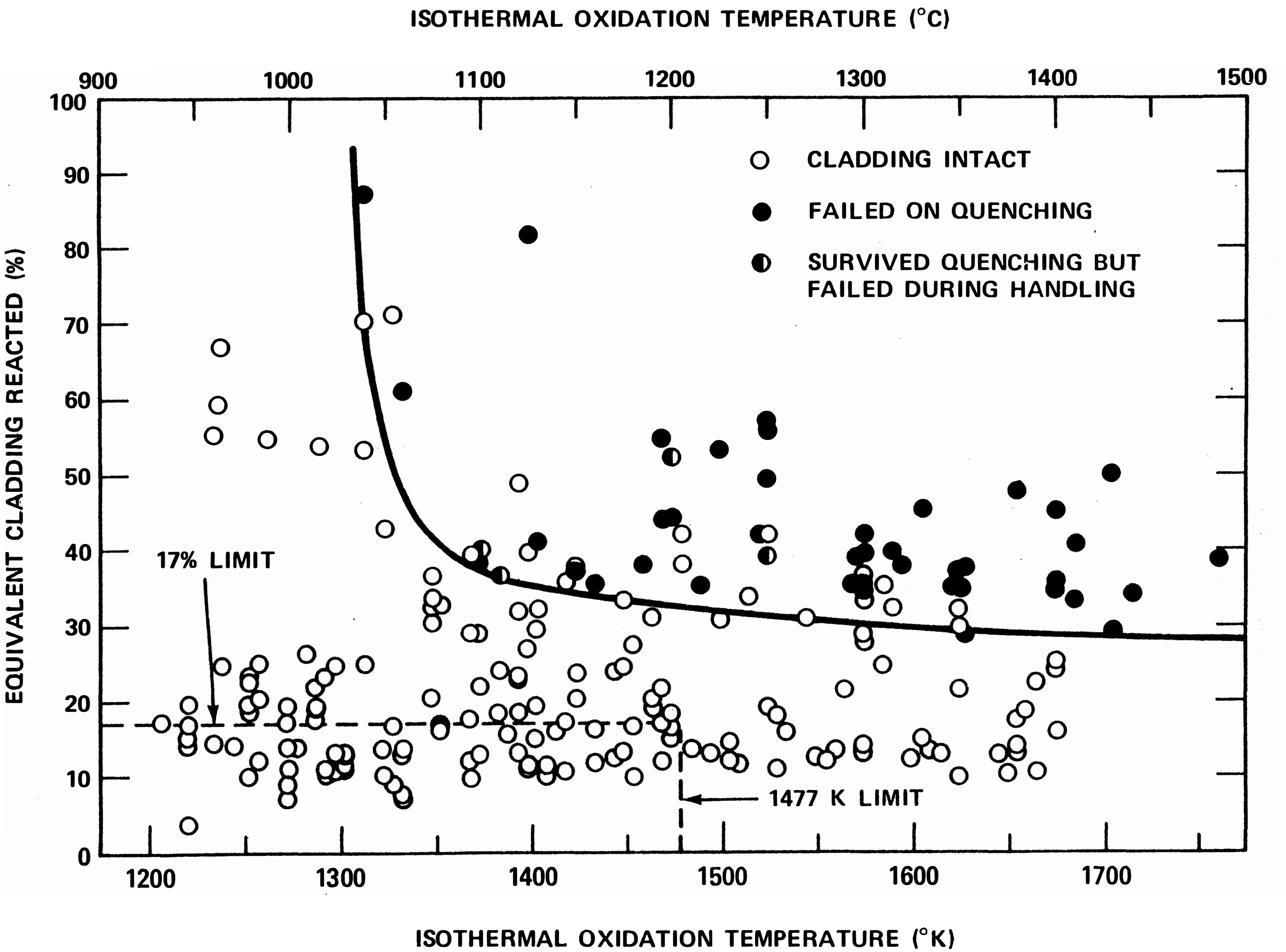


FIGURE 11.

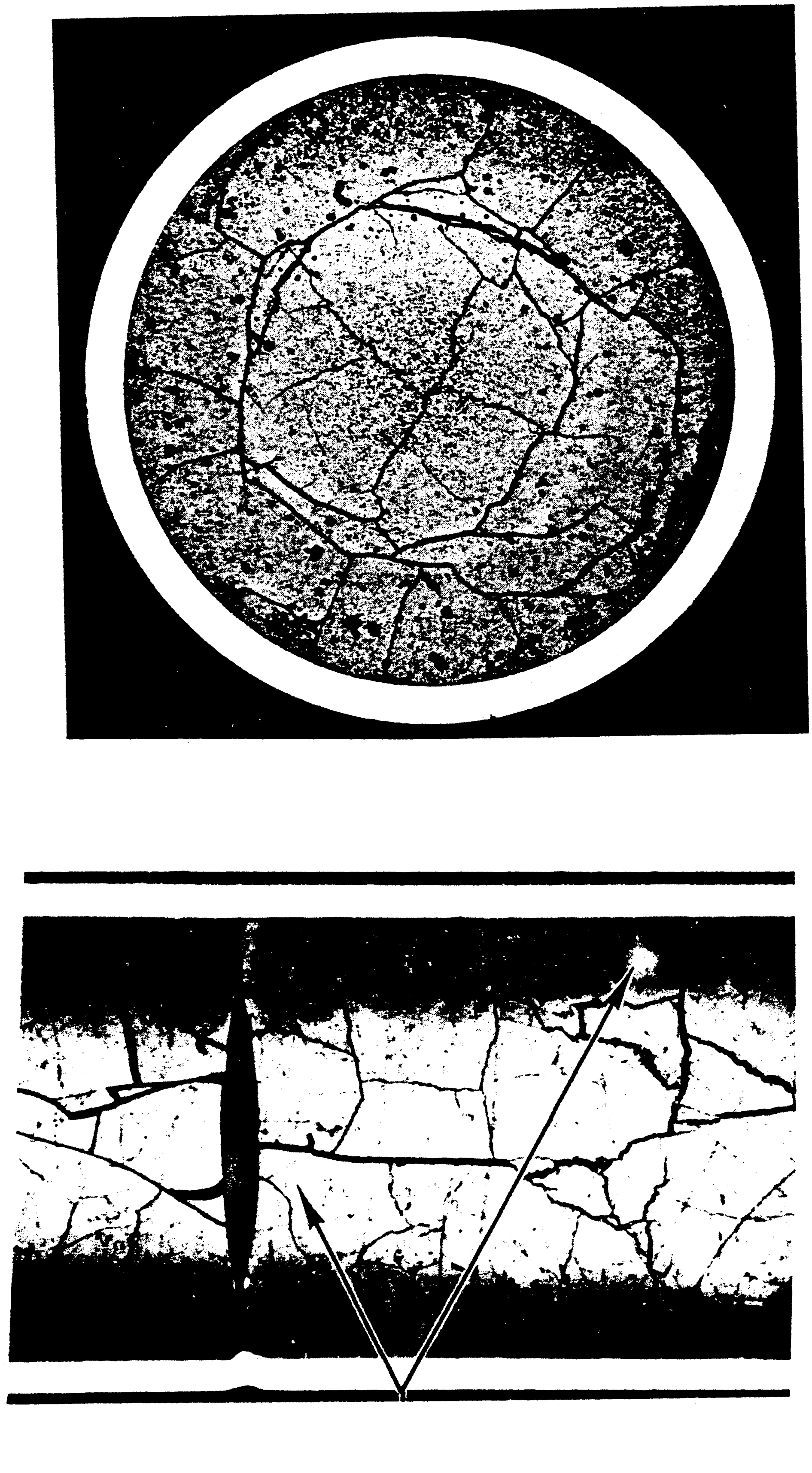


FIGURE 13.

Fuel pellets normally crack during operation and crack healing can occur at power. Figure 13 is a typical example of a cracked pellet. Quenching during core flooding may also promote fragmenting of the pellets. Severely fragmented regions are commonly seen in fuel pellets as a result of extreme temperature conditions in test reactors. Powdered regions in fuel pellets have also been seen in some PBF tests, but these tests are characterized by very high powers ($>$ 20 kw/ft) and very steep temperature gradients unlike the low-power uniform (radial) temperature TMI-2 fuel. Therefore we would expect the TMI-2 fuel to be in millimeter-size granules and larger pieces including whole pellets.

C. Unfueled Components (Control rods, guide tubes, etc.)

Figures 14 through 17 show the control rods, the burnable poison rods, the power shaping rods, and the central instrument tube. All of these rods and the instrument tube are inserted into Zircaloy guide tubes in the fuel assembly. The materials of which these components are made are indicated on the figures.

An important clue about the condition of unfueled components is provided from instrument readings. The fact that all 52 thermocouples worked throughout the accident and most continue to give credible information suggests that a central tubular structural member survivied. It is tempting to conclude that all Zircaloy guide tubes also survived, but this may not be the case since the thermocouple is well protected by multiple barriers.

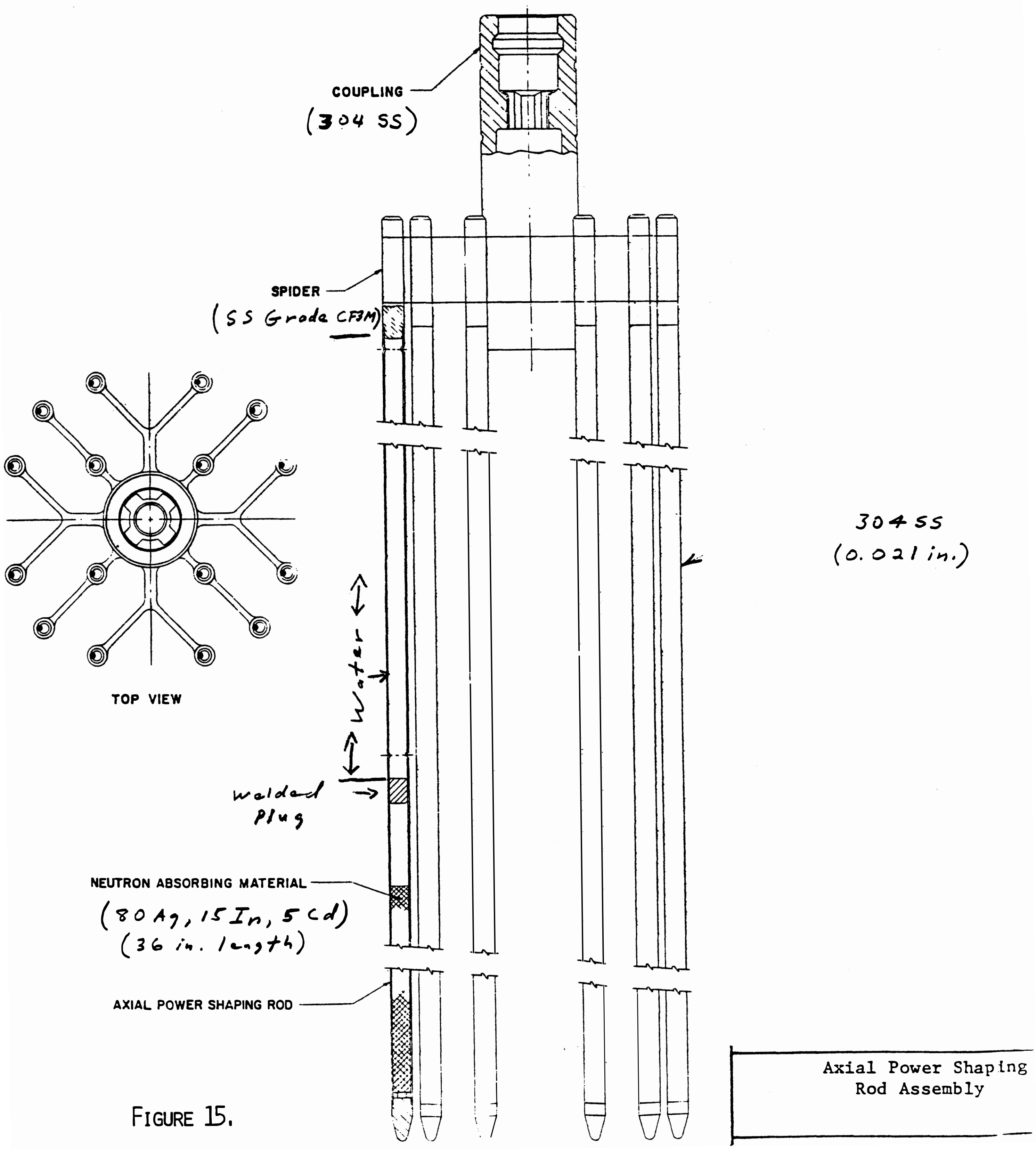


FIGURE 15.

Axial Power Shaping
Rod Assembly

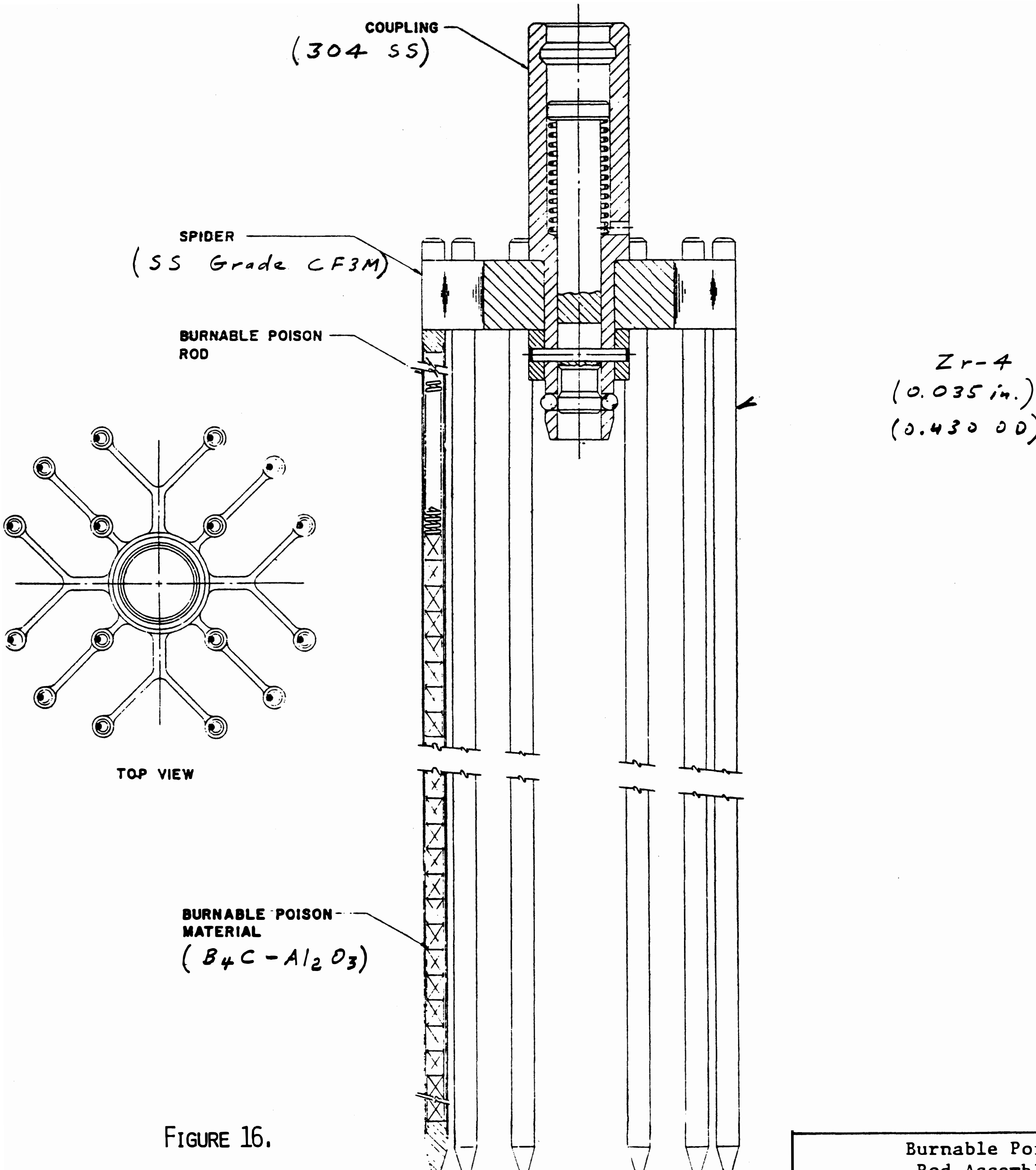


FIGURE 16.

Burnable Poison Rod Assembly

Fixed SPND assembly cross section

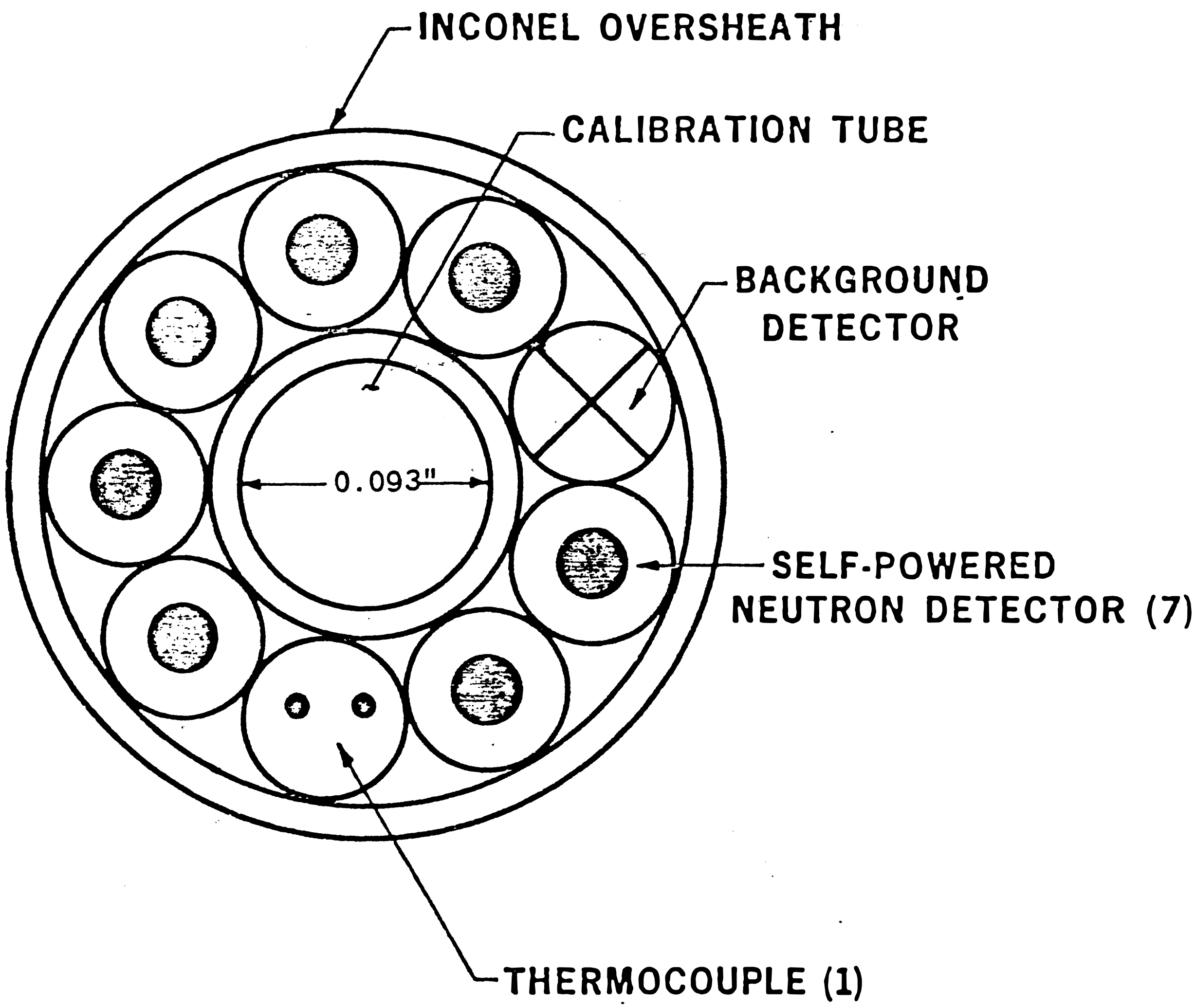


FIGURE 17.

INEL has made calculations of guide tube temperatures by parametrically varying heat transfer conditions (see Attachment C). Their results show that guide tube temperatures lag the fuel rod temperatures by only about 20^{o}F. Babcock & Wilcox has performed similar calculations and concluded that there is a much larger spread in temperatures. We believe the INEL calculations are more nearly correct and that temperatures of unfueled components were close to fuel rod temperatures. Since fuel rod temperatures are believed to have exceeded 1750^{o}C in the hot region of the core, then in that region (a) Ag-In-Cd and its stainless steel cladding would have melted, (b) Inconel spacer grids would have melted, (c) Zircaloy guide tubes would have oxidized, and (d) Zircaloy cladding of the burnable poison rods would have oxidized.

In the cooler parts of the core below about the 4 to 6 ft. elevation, we would expect all unfueled components to be intact, although perhaps damaged, just as the fuel rods are expected to be intact. Control rod segments could have only fallen about 3 inches if severed by melting in the hot region, and the Ag-In-Cd absorber should be in place because it is an insoluble metal. Although the burnable poison rods would also be expected to be in place, their poison is probably lost; boron is known to leach out of B_4C-Al_2O_3 pellets when exposed to water in a radiation environment.

D. SUMMARY

Many or all fuel rods may have ballooned and ruptured, but this mode of initial defecting is probably irrelevant in light of later more extensive damage.

In the hot upper central region of the core, fuel temperatures probably exceeded 1750°C releasing large quantities of fission products; about 30% of the total core inventory of noble gases was released.

About 40% of the Zircaloy cladding reacted with water. This region of severe oxidation was localized above the 4 to 6ft elevation and may not have included peripheral bundles. The severely oxidized fuel probably fragmented into pieces ranging from millimeter size to whole sections of rods.

The temperature of unfueled components lagged the temperature of fuel rods by only about 20°F so that they also experienced temperatures above about 1700°C. Consequently, in the hot region of the core Zircaloy components should have oxidized, and components with Inconel, stainless steel, and Ag-In-Cd should have melted. Because of many layers of protection, the thermocouple tubes have survived even in the damaged core region, although the outer sheath of the instrument tube may be badly damaged.

Nearly all of the broken and oxidized fuel debris should remain trapped in the upper core region because the upper end fittings have a grillage that would act as a screen. Furthermore, the compaction of fuel debris is limited because it is fabricated with a packing fraction of about 46% and the theoretical maximum packing fraction (for a bed of spherical particles) is only about 63%. It is very likely that fuel debris are also trapped in some mixing cups (See Figure 18) contributing to non-uniform thermocouple readings.

An earlier estimate of fuel damage in TMI-2 was made at NRC by Rubenstein, Meyer, Tokar, and Johnston. That estimate is in general agreement with the present estimate although our current evaluation is more refined. A memorandum summarizing the earlier estimate is attached as Attachment D.

E. Recommendations

Reactor fuel is rugged, and it is unlikely that limits for natural circulation conditions will be related to fuel behavior. The general criterion with regard to the fuel should be that additional Zircaloy oxidation and fission gas release should be avoided.

Significant oxidation rates do not occur until 900 or 1000^{o}C (See Figure 6). Significant fission gas releases do not occur until even higher temperatures (See Figure 2). These temperatures should be avoided in the (relatively) undamaged regions of the TMI-2 core, but these temperatures are so high that other limits will probably prevail.

By now the adiabatic heatup rate is low (See Figure 19) and ample time will be provided to detect fission gas or hydrogen releases. Therefore, on-line methods of such detection, if feasible, should give adequate

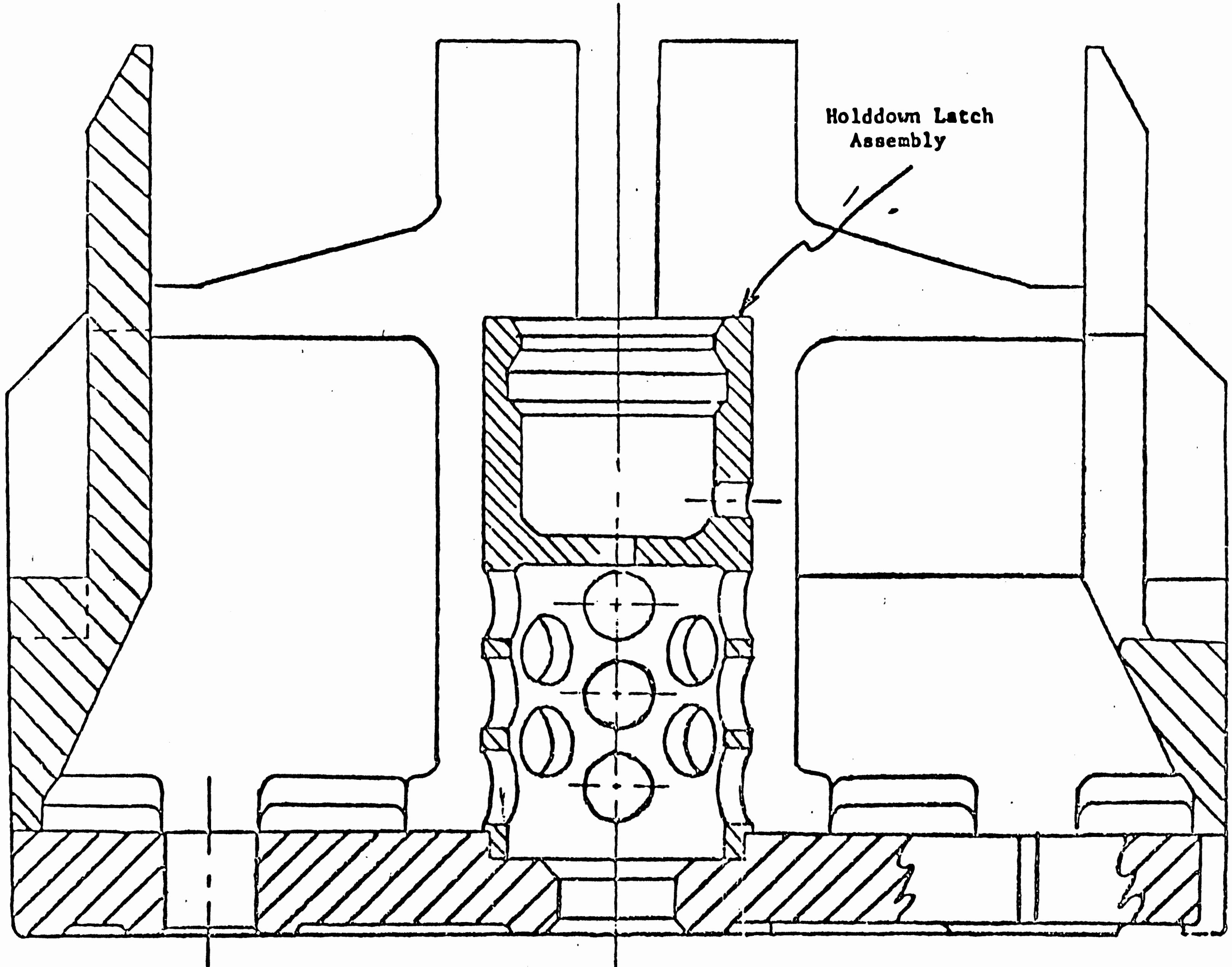


FIGURE 18. UPPER END FITTING AND HOLDDOWN LATCH ASSEMBLY

SUBJECT

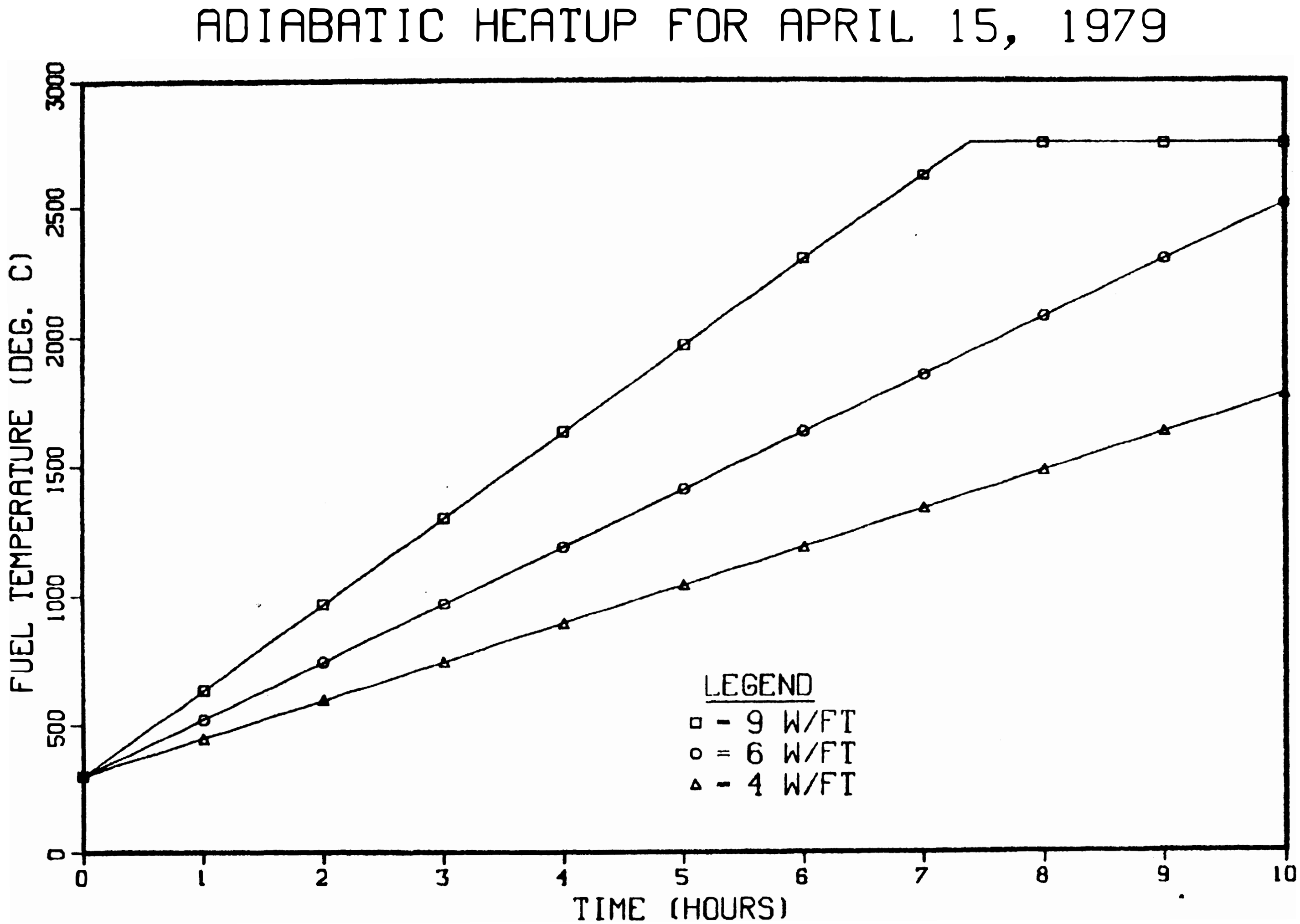

ADIABATIC HEATUP FOR APRIL 15, 1979
FUEL TEMPERATURE (DEG. C)
0
500
1000
1500
2000
2500
3000
TIME (HOURS)
0
1
2
3
4
5
6
7
8
9
10
LEGEND
□ - 9 W/FT
○ = 6 W/FT
△ - 4 W/FT

FIGURE 19.

warning of fuel damaging conditions.

A discussion of instrument responses relevant to fuel behavior was held with a group of fuel experts from across the industry. A summary of those discussions was prepared by W. V. Johnston and is attached as Attachment E. One consensus of that group was that in-core thermocouple readings should be recorded continuously. A recommendation for such data recording was made and is attached as Attachment F.

ATTACHMENT A.

K.O. MEYER
P-1114

VZCZCRNC18S
RATUZYUW RHEGORO2104 2021429-UUUU--RHEGNRC.
ZNR UUUUU
P 120457Z APR 79
FM R A LORENZ OAKRIDGE NAT LAB OAKRIDGE TENN
TO R O MEYER NRR PHONE 2 7603 NRC WASHINGTON DC 20545
AE
BT
UNCLASSIFIED/N O N W D
DATA FROM 53 POST IRRADIATION ANNEALING EXPERIMENTS CONDUCTED BY
G. W. PARKER ET AL. WERE ANALYZED IN ORDER TO OBTAIN ESTIMATED OF
XENON..IODINE AND CESIUM RELEASED FROM UO2. THESE TESTS WERE
CONDUCTED WITH UO2 IRRADIATED TO BURNUPS OF TRACE TO 4000 MWD MT
HEATED AFTER IRRADIATION IN FLOWING INERT ATMOSPHERES FOR
5.5 HR. THE TESTS ARE SUMMARIZED ON P. 80 OF REPORT ORNL 3981.
THE FOLLOWING NUMBERS ARE THE PERCENTAGE RELEASES OF XENON
CORRESPONDING TO MINIMUM PROBABLE..MOST PROBABLE..AND MAXIMUM
PROBABLE. RELEASE.
AT 1600 DEG C THE ESTIMATED RELEASE PERCENTAGES ARE 2.2..5.3.
AND 11.5. AT 1800 DEG THEY ARE 7..15..AND 30. AT 2000 DEG C
THEY ARE 19..37..AND 61. RELEASE PERCENTAGES FOR IODINE AND
CESIUM AVERAGED APPROXIMATELY TWICE THE ABOVE VALUES
BT
#2104

NNNN

Attachment B

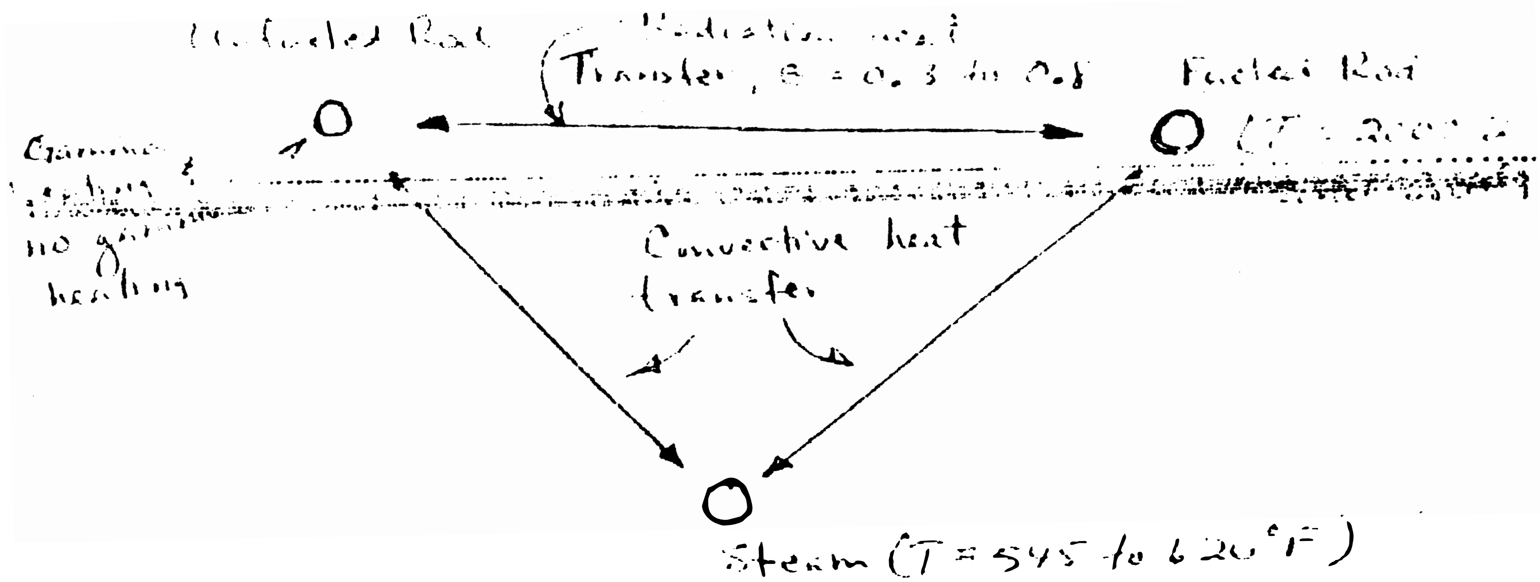
See memorandum from W. Butler, Chief of the Containment Systems Branch, to R. Tedesco, Assistant Director for Reactor Safety, and entitled, "Three Mile Island, Unit 2: Analysis and Evaluation of Selected Containment Related Issues," to be issued on or about April 16, 1979.

4-11-79

To: Ralph Meyer
From: C. Chenhan

ATTACHMENT C.

Ralph - Summarized below are our heat transfer calculations discussed earlier with you.



Results

Heat transfer coeff, h = 1.76 to 2.88 English units

Temp. unfueled rod lags fueled rod by 18°F (essentially same for all cases run)

Assumptions:

1. Power was .(13)(0.011) 2.881 kw/ft.

2. Gamma heating was 11% of decay power.

3. Case's run was four fuel rods and 17 unfueled rods.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
April 6, 1979

Meyer

ATTACHMENT D.

MEMORANDUM FOR: E. G. Case, Deputy Director
Office of Nuclear Reactor Regulation

FROM: TMI Fuel Team

SUBJECT: ESTIMATE OF FUEL DAMAGE IN THREE MILE ISLAND (TMI)

Enclosed is a brief report describing the preliminary conclusions of the team formed to analyze the probable damage to the fuel system at TMI.

L. S. Rubenstein
L. S. Rubenstein, PSS/NRR

Ralph Meyer
R. O. Meyer, DSS/NRR

M. Tokar
M. Tokar, DSS/NRR

W. V. Johnston, RSR/RES
W. V. Johnston

Enclosure -
As stated

On Tuesday, April 3, 1979 a team consisting of

L. S. Rubenstein, PSS/NRR
R. O. Meyer, DSS/NRR
M. Tokar, DSS/NRR
W. V. Johnston, RSR/RES

was formed to survey the fuel groups analyzing the damage to the fuel system of TMI and draw some preliminary conclusions from their deliberations regarding that damage.

The following individuals and organizations were contacted on April 3 and 4, 1979:

E. L. Zebroski (EPRI) -
(representing the Metropolitan Edison Group)
J. Taylor/J. Tulenko, B&W
R. Denning, BCL
D. McCloskey, Sandia
J. Scott, LASL

In addition to the information obtained from conversations with these organizations and the NRC staff, the team obtained a "sequence of events" from B&W (Enclosure 1) a group of curves describing the pressure, temperature changes at TMI-2 during the first 15 hours from D. Eisenhut, and a BAPL radiochemical analysis of the primary coolant taken at 1600 hours March 29, 1979 and decay corrected to 0700 hours March 30, 1979.

The primary information used in our analysis of fuel system damage was obtained from the B&W Company, the Metropolitan Edison Industry Group, and from calculations of the NRC staff (Reactor Fuel Section, CPB; Fuel Behavior Branch, RES).

System Effects

Using the chronology of events obtained from B&W and the control room strip chart tracing of system pressure for the first 15 hours of operation, we were able to determine that there were three periods in which the primary system pressure was below a saturation pressure corresponding to a temperature of 620°F. The system changes which caused these periods are described in the sequence of events provided by B&W enclosed with this report. The details of what occurred to cause the pressure changes in the primary system are not discussed here as these are considered in other staff reports (see e.g., IE Bulletin 79-05A, Nuclear Incident at Three Mile Island) and will be evaluated by others.

Examination of Figure 1 shows that the first period in which the system pressure was substantially below saturation pressure occurred approximately 1.75 to 3 hours after start of the transient. The second period, which was relatively short in duration, occurred in the 4.5 to 5.5 hour time frame and resulted in a small decrease in primary system pressure below saturation pressure. The final period of decreased primary system pressure extended from approximately 8-14 hours after start of the transient. It was during these 3 periods that the core was exposed to extensive amounts of steam cooling and experienced fuel damage. The group was able to infer from examination of these pressure histories, reports of fuel channel temperature changes with time obtained from the incore thermocouples, the behavior of the incore rhodium self-powered neutron detectors (SPND's), and 3'-long Intermediate Range Ex-Core Detectors, and the containment radiation monitors some details of when the fuel pins lost their integrity, the depth of the core which was exposed to steam cooling, the probable time periods of that exposure, and the amount of damage to the fuel.

As previously stated, the evidence for the level of uncovering was obtained from a B&W analysis of the incore SPND's. It can be shown that :

> Above about 700°F, incore SPND's (Rh) act as thermionic elements and generate currents which are correlatable to temperature. Thus, if a discontinuity is observed in current measurement, a transition in temperature may be inferred. It was assumed that this discontinuity represents an elevation at which voiding of the coolant has occurred.

Similarly, the excore Intermediate Range Detectors may be used to provide an indication of voiding.

The information obtained from these detectors was consistant with the results from the Industry Group calculation that, in approximately one hour without introduction of makeup water, the core could boil down to full uncovering.

Fuel System Conditions During Period of 1st Uncovering

During the first period of major uncovering of the core (at least 5 feet of the core was uncovered for about an hour, and perhaps all of the core may have been uncovered for about one-half hour), the uncovered portion reached temperatures high enough to fail fuel rod cladding. At this point, fission products were released into the primary coolant as evidenced by the subsequent alarming of the containment activity monitors. Based on the measured coolant activity and the amount of hydrogen release from reaction of the Zircaloy cladding with water[1], all of the fuel rods probably defected and released fission products.

Fuel temperatures were estimated from calculations based on the fission product analysis of the sample of primary coolant, and also from heat transfer considerations. Based on back-calculations[2] that accounted for temperatures and temperature-dependent release rates that would be required to produce the measured level of activity, fuel temperatures of 1400 to greater than 1600°C were obtained. Estimates by ORNL based on their experiments indicated that the Cs and I releases measured would have required fuel temperatures of at least 1300°C for an hour. The heat transfer calculations indicated, on the other hand, that the fuel temperature may have been only about 1100°C[3]. In either case since the melting point of UO_2 is 2840°C, fuel melting was unlikely. These temperature differences can be rationalized by considering that a small portion of the core may have been at the higher temperatures. There is also a possibility of some eutectic formation between UO_2 and ZrO_2 at temperatures above approximately 1800°C, but no significance was attached to the occurrance of such a eutectic. Later analysis by members of ANS-5.4 fission gas working group (including one of us--ROM) indicates fuel pellet temperatures as high as 2000°C based on Xe^{133} data and the assumption that half of the core remained cool. While noble gas activities lend themselves to smaller analytical uncertainties than iodine or cesium activities, the uncertainty in the core fraction that is responsible for the release still renders this result inconclusive.

Hydrogen balance calculations indicate that from 15 to 30%[3] of the total Zircaloy inventory has been oxidized[4]. Some of the oxidation, however, undoubtedly occurred during the latter uncoverings. The extent of the oxidation probably varies as a function of height in the core, with the greatest amount of oxidation having occurred in the uncovered (upper) portions of the fuel rods. Later calculations accounting for hydrogen in the bubble, in the containment, lost in the hydrogen explosion, and gained by radiolysis suggests that almost 40% of the Zircaloy in the fuel region may have been oxidized.

---

1 CPB Staff Calculation
Industry Group Calculation
B&W Calculation

2 CPB Staff Calculation
3 B&W, Industry Group and NRC Staffs
4 Industry Group & NRC Staffs

As the primary coolant level was restored during the latter portion of the time period of the first uncovering, thermal and mechanical shock loadings of the oxidized and embrittled cladding are believed to have occurred and to have resulted in cladding fragmentation.

At the end of the period of first uncovering, virtually all of the fuel rods had defected and released fission products. Although temperatures had been high enough for a long enough time to have caused severe cladding oxidation, continued operation of incore instruments strongly indicates that fuel assembly structural members such as guide tubes remained intact. Control rod materials are believed to have remained in place, as indicated by the absence of silver in the primary coolant.

Fuel System Conditions and Effects During Period of 2nd Major Uncovering

At about 4 1/2 hours into the event, the core level again decreased to expose the upper 5 feet of the fuel assemblies. The duration of this additional uncovering was shorter than the first, the system pressure was higher, and the overall temperature effects were less severe, as evidenced by the fact that the thermocouples in the outer periphery of the core remained on-scale. Because of the reduced severity of the core conditions during the second uncovering, as compared with the first uncovering, less damage is believed to have occurred to the fuel system.

Fuel System Conditions During Period of 3rd Uncovering

At about nine hours into the event, the core coolant level again decreased, possiblity down to 7 to 7 1/2 ft. from the top of the active fuel level.* The core remained uncovered at this level for about one to three hours, after which the coolant level was again raised and covered the core. The low system pressure (~450 psi minimum), the rather lengthly period of uncovering, and the additional length of fuel surface uncovered, undoubtedly resulted in additional fuel system damage due to Zircaloy oxidation and embrittlement (followed again by more fragmentation due to thermal shock during the recovering of coolant level), although the amount of additional damage is presently unquantifiable.

Fuel System Damage Summary

The picture of the core that has emerged is that the core configuration currently consists of a basket-like shape of relatively intact assemblies that surround a central region of severely oxidized, and probably fragmented, fuel rods in the upper central part. The fuel

*Based on information received via telecommunication from B&W (April 3) rods are less damaged in the lower central part of the core. Although the fuel rods in the upper central region may be completely fragmented, the guide tubes, grids, and end plates are believed to be intact thus providing a skeletal structure which supports the remaining portions of the damaged assemblies. Partial flow blockage caused by accumulation of fuel debris is thought to be responsible for continuing elevated thermocouple readings. The assymetry of the incore thermocouple readings suggests that a region of the core is more heavily damaged than the average.

PRELIMINARY SEQUENCE OF EVENTS (TMI-2, 3/28/79 INCIDENT)

The following sequence of events for the TMI-2 incident of 3/28/79 has been formulated by B&W engineers using available plant data. This chronology has been constructed from numerous sources and has not been totally confirmed. It may not be precise in either event occurrence or sequence.

Time, Minutes	Event
Prior to turbine trip	The initiating events could have come from numerous postulated causes. For purposes of this sequence, they are relatively unimportant. The prime effect is that it led to a loss of main feedwater (MFW) booster pumps.
0	Main feedwater pumps are tripped. Almost simultaneously, the turbine trip occurs.
0.10	Pressurizer pressure increases to the ERDV setpoint of 2270 psig.
0.15	Secondary side pressure peaks at 1070 psig and is limited by steam relief valves.
0.20	RC pressure trip setpoint reached (2355 psig at hot leg tap) and system pressure peaks at about this value.
	Indications from pump discharge pressure are that auxiliary feedwater pumps (one turbine driven, two electric) are running at this point; however, no level change occurs in steam generators.
0.25	Pressurizer level peaks at 255 inches (indicated) and starts to decrease with system contraction.
0.30	Quench tank pressure is increasing.
0.90	Pressurizer level is at a minimum of 158 inches and starts to increase. Hot leg temperature is at a minimum of 577°F and starts to increase slowly.
1.0	OTSG level indication on the startup range is 10 inches. OTSG pressure holds at about 1025 psig.
2.0	OTSG pressure starts a steady decrease. HPI flow is initiated by ESFAS on low RC pressure (HPI setpoint = 1600 psig).
3.0	The quench tank's increasing pressure levels off at 120 psig. Relief valve setpoint is 150 psig.
4.75	The hot and cold leg temperatures start increasing at a more rapid rate. Analytical simulation indicates that this occurs when the HPI is turned off. Site information notes that operator terminates HPI fully at 5.1 minutes.

Time, Minutes	Event
5.0	Pressurizer level indicates a slowing and then continues to increase as the hot leg temperature if increasing.
6.0	Pressurizer level indicates a full pressurizer and the quench tank pressure increases beyond the relief valve setpoint of 150 psig.
	RC pressure reaches a minimum of 1350 psig with a hot leg tempreature of 584°F. This indicates hot leg is in saturation condition.
8.0	Auxiliary feedwater flow is initiated to both OTSG's. This is indicated by immediate OTSG repressurization to ~1025 psia and OTSG level change.
9.0	RC pressure peaks out at 1500 psig and starts to decrease. Hot leg temperature peaks out at 597°F.
11.0	Pressurizer level indiciation is restored. It stabilizes out at 375 inches at 15 minutes.
16.0	Quench tank pressure drops suddenly, indicating the rupture disk has blown (setpoint = 200 $\pm$ 25 psig).
18.0	The decreasing RC pressure stabilizes at 1115 psig.
22.0	The RCS temperature stabilizes at a hot leg of 553°F and a cold leg of 548°F. The temperature decrease from start of auxiliary feedwater to this stabilization represents a 200°F/hr cooldown. Reactor building pressure is 1.4 psig and increasing. Two feet level is restored in both OTSG's.
50.0	The startup level indication shows OTSG B level increasing and OTSG A level decreasing. Pressure increases in both OTSG's.
60.0	During the 22-60 minute period, the system parameters have stabilized in the saturation condition of a pressure of ~1015 psig, temperature of ~550°F. RC flow indication is decreasing from 60 (initial) to 50 x 10^6 lb/hr. The reactor building pressure is 2.2 psig and increasing.
73.0	Two RC pumps are tripped (in Loop B). Reactor coolant flow rate decreases in Loop B.
78.0	OTSG B pressure drops from 950 psig to 140 psig in 18 minutes.
90.0	T_{hot} follows T_{sat}. ΔT across the core equals about 5°F.
100.0	Both remaining RC pumps are tripped.
114.0-120.0	T_{hot} and T_{cold} diverge rapidly. T_{hot} > 620°F in less than 15 minutes.
132.0	Site information notes that EMOV relief line was isolated initially. RB pressure starts decreasing more rapidly.

Time, Minutes	Event
135.0	RCS has depressurized to 670 psig and RCS hot leg temperature is at maximum scale of 620°F. At 620°F, system would have superheating at upper elevations as long as pressure was below saturation pressure of 1772 psig. RCS shows rapid re-pressurization.
150.0	OTSG B level ramped up from 5% to 65% in 48 minutes.
180.0	OTSG B main steam isolation valves and turbine bypass valves are closed. RCS pressure peaks at 2120 psig.
180.0-204.0	Regulation by EMOV block valve reduces RCS pressure.
204.0	HPI comes on (1600 psig signal).
216.0m	HPI pump 1c to Loop A turned off. RC pressure decreases stepwise. RB pressure increases stepwise.
290.0m (4.83 hr)	RB pressure hits 4 psig. Building fan cooler comes on.
318.0m (5.3 hr)	RCS pressure increases rapidly from 1250 to 2120 psig in 35 minutes. The EMOV block valve is closed, one HPI (1A) is on.
354.0 (5.9 hr)	OTSG A level is ramped up from 50% to 95% on operating range in 1 hour and to 100% in 1.5 hour. OTSG A pressure starts to decrease toward zero.
450.0 (7.5 hr)	The EMOV block valve is opened. RCS pressure starts to decrease (2050 psig to 480 psig in 1 hr, 45 min).
519.0 (8.65 hr)	RC system pressure reaches 600 psig, core flood tank setpoint.
588.0 (9.8 hr)	RB pressure spike to 28 psig occurs.
630.0 (10.5 hr)	T_{hot} Loop A reappears on scale, decreases to 525°F in 1/2 hr.
678.0 (11.3 hr)	T_{cold} Loop A increases in about 5 minutes from 190°F to 400°F.
750.0 (12.5 hr)	HPI flow increased to 400 gpm. T_{hot} in Loop A decreases.
810.0 (13.5 hr)	T_{cold} Loop A decreases.
948.0 (15.8 hr)	Pump 1A is started.

Time, Minutes	Event
Thereafter	Condenser vacuum re-established. SG A begins steaming to condenser. RCS cooled to approximately 300°F, 1000 psi. Letdown line ceased to permit flow and relief valve being used (estimated 14-16 gpm flow). Some fuel incore thermocouples reading about 600°F. RB pressure below 1 psi. High radiation in reactor containment and auxiliary building.

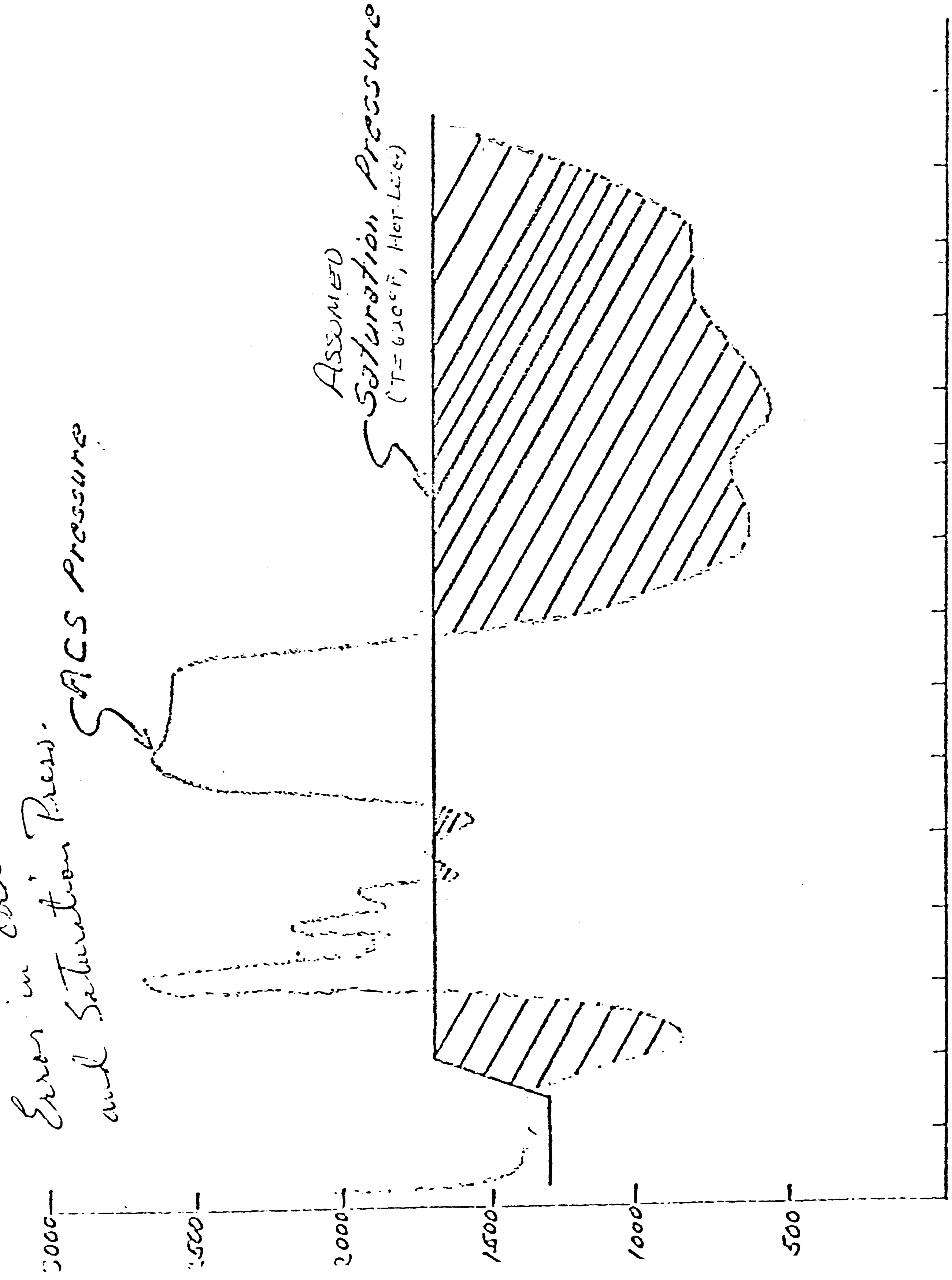


FIGURE 1
Error in Evaluate and Saturation Press.
RCS Pressure
Assumed Saturation Pressure (T=620°F, Hot Leg)
Time, hrs.
0 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15
500 1000 1500 2000 2500 3000

Prepared for Eisenhut's April 4
Commission Briefing

Core Coolant Conditions

- At 2 hours after turbine trip the core had become partly uncovered and remained uncovered for about one hour.
- During this period activity alarms came on indicating significant fuel failure.
- Core was recovered when high pressure injection pump came on.
- Two additional periods of extensive core uncovering followed at about 5 and again at 9 through 12 hours after turbine trip.

Number of Fuel Rods with Defects

- Based on measured coolant activity, all of the fuel rods probably released fission products.

- Amount of hydrogen released from oxidation of cladding (metal/water reactions) also indicates all fuel rods are damaged.

Maximum Fuel Temperatures

- Calculations based on Fission product analysis indicate fuel temperatures of 1400 to 1600°C.

- Heat transfer calculations indicate temperature of about 1100°C.

- The melting point of UO_2 fuel is 2840°C so that core meltdown was not approached.

- The absence of Sr and Ba activity in the coolant confirm the avoidance of fuel melting.

Extent of Fuel Damage

- Hydrogen balance calculations indicate from 15 to 41 ~~30~~% of the Zircaloy cladding has been oxidized.

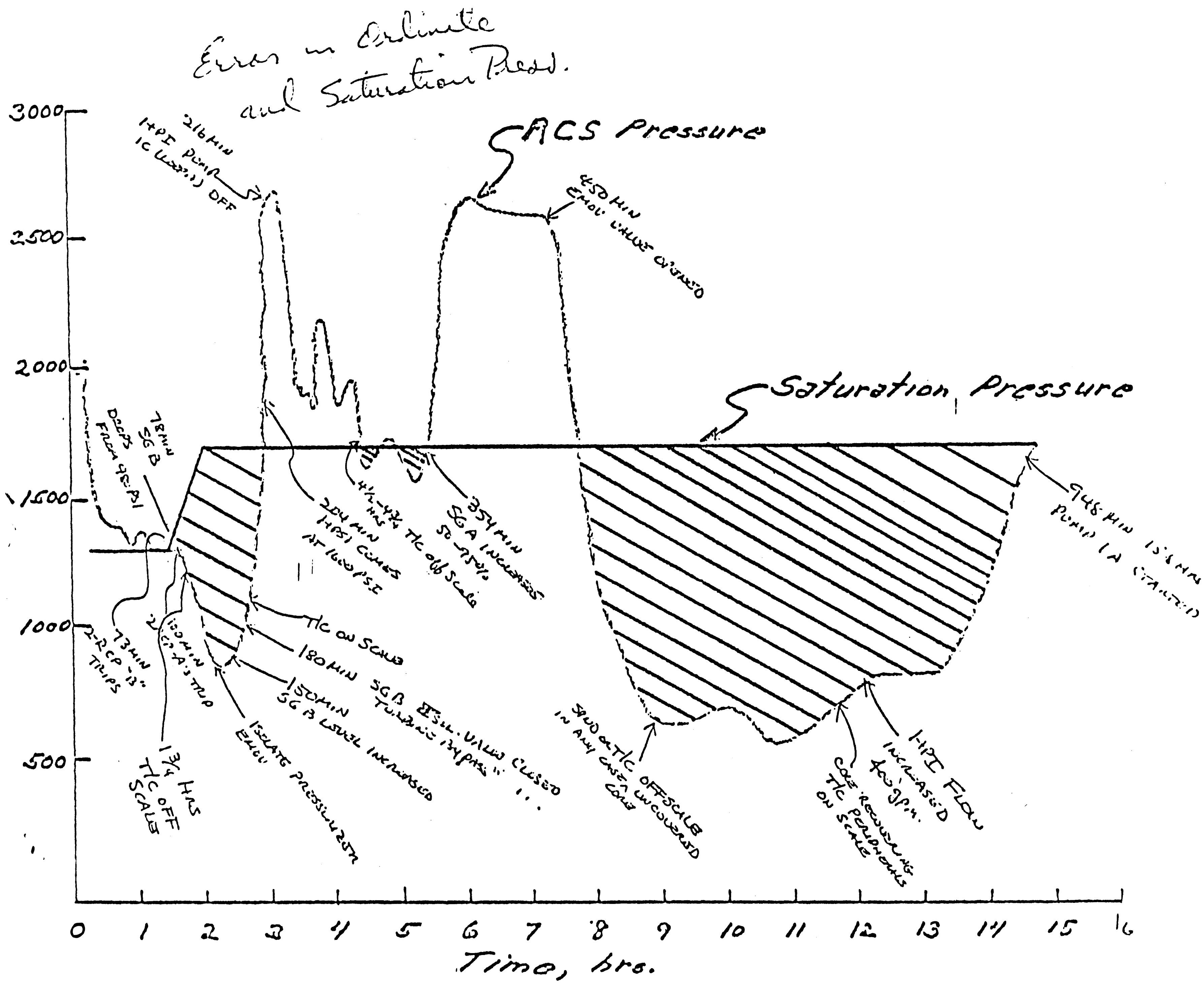
- Continued operation of incore instruments indicates that fuel assembly structural members remain intact.

- Absence of silver in coolant suggests that control rod materials remain in place.

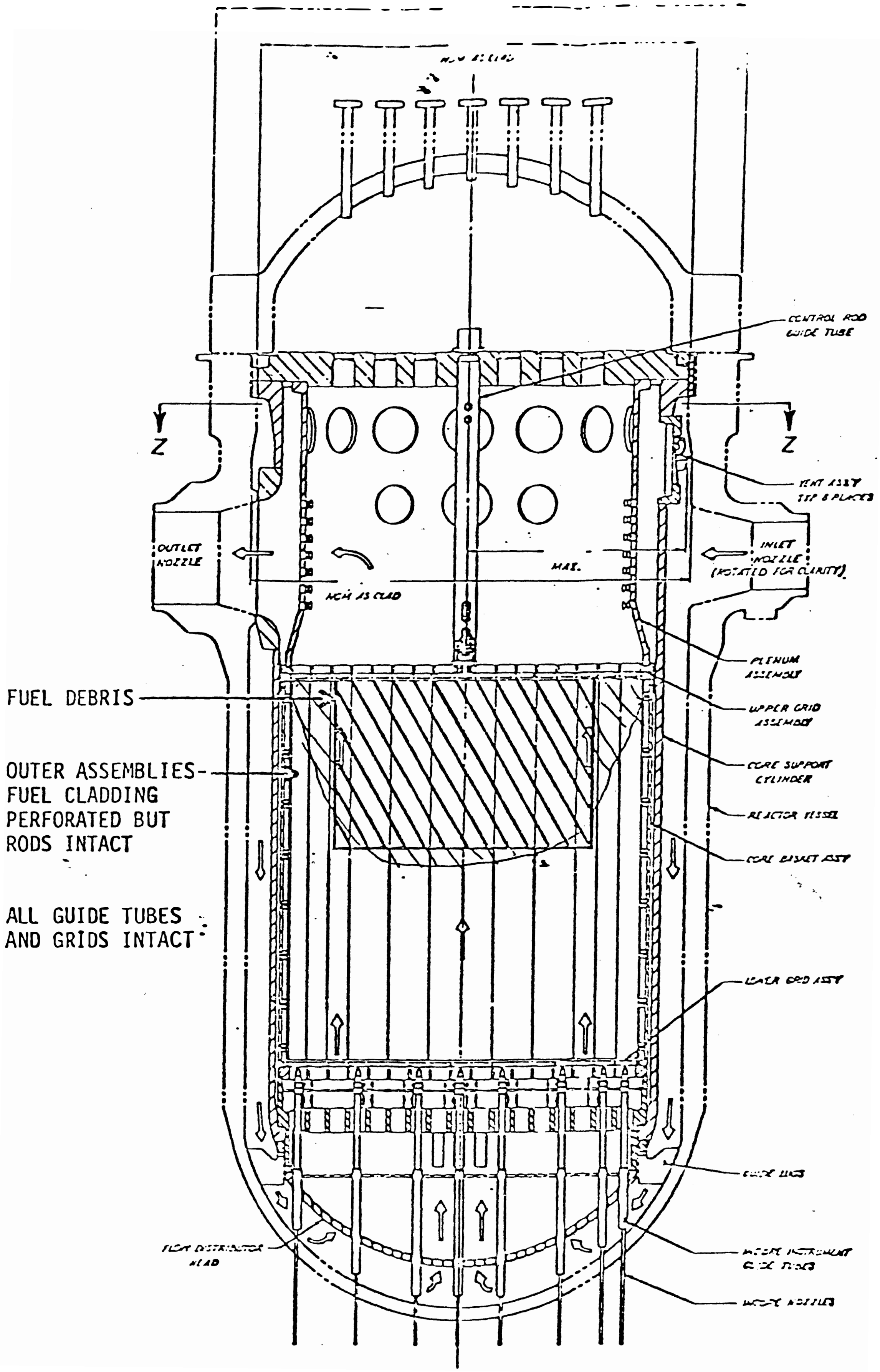
- Continued low thermocouple readings at periphery suggest that peripheral fuel assemblies retained much of their original geam.

- The picture that emerges is that the upper central part of the core is severly oxidized; probably fragmented, and largely confined to the core region (based on loose parts monitoring data).

- Partial flow blockage caused by accumulation of fuel debris has probably occurred and is responsible for elevated thermocouple readings.



Saturation Pressure
3000
2500
2000
1500
1000
500
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
Time, hrs.
216 MIN HPI PUMP 1C (WATER) OFF
450 MIN
78 MIN SG B
204 MIN HPSI COMES AT 1600 PSI
354 MIN SG A INCREASED
T/C ON SCALE
180 MIN
150 MIN SG B LEVEL INCREASED
ISOLATE PRESSURIZER EMOV
73 MIN 2-RCP "B" TRIPS
1 3/4 HRS T/C OFF SCALE
HPI FLOW INCREASED
948 MIN



REACTOR VESSEL & INTERNALS ASSEMBLY

ATTACHMENT E.

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

APR 13 1979

MEMORANDUM FOR: D. Ross, Deputy Director
Division of Project Management
Office of Nuclear Reactor Regulation

FROM: W. V. Johnston, Chief
Fuel Behavior Research Branch
Division of Reactor Safety Research
Office of Nuclear Regulatory Research

SUBJECT: FUEL EXPERTS MEETING ON CONDITION OF THE TMI CORE

A meeting of nuclear fuel experts was held on April 12 to update the estimates of the damage to the TMI core and to consider its effect on the desirability of moving to natural convection cooling of the core. The Experts Group consisted of the following persons: J. S. Tulenko, B&W; R. DeMars, B&W; T. Kassner, ANL; R. A. Proebstle, GE; K. A. Jordan, W; R. Duncan, CE; T. Fernandez, EPRI; T. Buhl, NRC; R. Meyer, NRC; W. Johnston, NRC, Chairman. Additional attendees included L. Rubenstein, NRC; C. Berlinger, NRC; M. Tokar, NRC; R. Majors, ACRS Staff and T. Mott, TEC.

Summary

The group concluded that although the core is badly damaged, essentially all of the fuel has remained in the core and that the overall packing density of the settled portion is not expected to exceed 70%. Therefore, shutting off the RC pumps should not seriously threaten further damage to the reactor. It was further concluded that the thermocouples (Tc's) located in the upper end fittings are the most important indicator of core condition during transition to natural convection cooling. If feasible, the addition of a γ spectrometer to monitor the activity of the loop coolant for new fission products released during the transition to natural convection cooling will provide an independent alert to possible difficulties.

Two questions were considered: Is a pump trip likely to lead to an unsafe condition? and What signals will indicate undesirable conditions in the core?

Summary of Core Status

Summaries of amount of damage to the core based upon measurements or calculations of fission gas release, hydrogen produced by zircaloy oxidation, coolant analysis, coolant boil-off rates, incore and excore instrumentation were presented by J. Tulenko and R. Meyer. A relatively large pressure drop across the core is inferred by TH calculations. If the pressure drop is real, blockage must also exist in the peripheral assemblies (perhaps by ballooning). The shift in location and magnitude of the high reading core Tc's following the pump trip on April 6 was believed to indicate either a change in the core flow path through more heavily damaged sections of the core to a redistribution of debris surrounding some of the thermocouple beads. An alternate explanation for the change in Tc temperature distribution patterns was presented by T. Mott of TEC. He suggested that the Tc temperature difference may be due to non uniform flow distributions caused by operation of a single pump rather than non uniformity within the damaged region of the core. Due to this non uniform flow distribution portions of the core may already be experiencing similar cooling to that expected during natural convection. Mott estimates smaller core pressure drop and suggests the B&W estimates may include substantial external pressure drops.

The group visualizes the core as consisting of a heavily damaged region resembling an inverted bell extending across nearly the full width of the top of the core and reaching down about five - six feet into the core at the center and a less damaged remainder of the core. In the heavily damaged region, 100% oxidation of the zircaloy and less of a regular geometry is expected. The guide tubes and poison rods are damaged similarly to the cladding. Spacer grids should be located at or near their original locations. The important coolability conclusions are that although some settling may have taken place, the overall packing density of the settled portion is not expected to be greater than 70% and that 85% to 98% of the fuel and cladding from this region is believed to have remained in the "core" region including the upper end fitting. The remainder of the core is less damaged although considerably oxidized. The original flow geometry is probably retained although the rods may be twisted or warped and broken in a few places and the spacer grids may have collected some loose debris.

The above conditions should not preclude satisfactory achievement of natural convection flows.

What should be monitored to determine undesirable changes in the core during the transition to natural convection cooling? The temperature distribution of the core exit thermocouples are the most important condition monitoring signals. The group believes that all exit Tc's should be continuously tracked and recorded. B&W suggested the following criteria for remaining in natural convection cooling: No more than 2 Tc's above 800°F and at least 10 Tc's below Tsat. There were some reservations among the group about allowing so many Tc's to read above Tsat (as many as 39) and radiolysis was an expressed concern. There was a lot of discussion but no concensus on how many interior Tc's should be permitted to exceed Tsat. Tc's in peripheral assemblies should not exceed Tsat.

The following table summarizes the available instrumentation and its possible application to monitoring core condition.

Detector	Event-Core Overheating Criteria	Basis
1. Exit Tc's	Limit no. in film boiling Limit no. above 800°F	Not to exceed previous core damage reverse procedure.
2. RTD Hot leg Cold leg	Maintain positive ΔT across core.	No flow reversal permitted.
3. Ion chambers δ and N	Void formation If +, record for future interpretation, watch Tc's.	Ambiguous signal since some local superheat may be permitted.
4. Noise detection	If + indicates bubbles in core or loop, check Tc's, SG.	Same as above.
5. System Pressure	If increasing system effects Branch should review this.	Not direct indication of core condition, but for gas bubble formation detection.
6. Pressurizer Level	Same as above.	Same as above.

Additional Detection - Feasibility needs to be established.

γ spectroscopy of coolant via sampling line	Increasing activity of Xe, I_2	Overheated core alert for major error in procedure.
H_2O analysis on line monitor	Boron, O_2, H_2	Core criticality and chemistry control radiolysis and H_2 content control.

William V Johnston

W. V. Johnston, Chief
Fuel Behavior Research Branch
Division of Reactor Safety Research, RES

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

April 13, 1979

ATTACHMENT F.

NRC ADVISORY -- THERMOCOUPLE READINGS

We recommend that all incore thermocouples be read in a continuous manner with provisions for rapid retrieval and permanent storage. It is clear that a continuous reading of thermocouples will be needed during the transition to natural circulation. It is also clear that temperature trends, which were not recorded during the pump changeover on April 6, would have given additional clues to core behavior in natural circulation. Since future flow transients cannot be ruled out and since the transition to natural circulation could occur involuntarily, we recommend that the continuous recording of all thermocouple readings be initiated as soon as possible.

Ralph O. Meyer

Ralph O. Meyer, Leader
Reactor Fuels Section
Core Performance Branch

APPENDIX B

TMI-2 PLANT MODIFICATIONS FOR ACHIEVING COLD SHUTDOWN*

MAY 1979

*This report was prepared based on information available to the staff prior to the plant being placed in a natural circulation mode of cooling on April 27, 1979. It is expected that certain plant changes different than those described herein may result and will be evaluated in a subsequent report.

TMI-2 PLANT MODIFICATIONS

FOR ACHIEVING COLD SHUTDOWN

Table of Contents

1. Background

In order to bring TMI-2 to a cold shutdown condition, modifications will be made to various plant systems in phases to be carried out over a period of the next few weeks. These modifications will permit a gradual transition from the current plant operating mode to one which provides a stable long term cooldown mode of operation.

The first planned modifications will be made in conjunction with the transition from forced primary coolant circulation (by the reactor coolant pump) to natural circulation through the core. To accomplish natural circulation, the secondary side (shell) of the steam generators will be operated water solid. Water will enter through the main feedwater ring and exit through the main steam line.

In order to provide for water solid operation, certain modifications to each steam generator secondary flow loop will be required. The design of modifications to steam generator "B" has accounted for possible contamination as a result of suspected tube leakage in the steam generator.

To backup pressure control capability of the primary system during natural circulation, a new pressure control and makeup system will be provided. This system is essential in the event of loss of pressurizer heaters and level indication. Criteria and procedures for letdown and overpressure control of the primary system will be established prior to going into this mode of operation.

Because of suspected leakage in the existing plant decay heat removal system, a program will be conducted to identify and correct leaks to provide as leak tight a system as possible. Also, an additional skid mounted decay heat removal train will be connected into the existing system as a backup. Connections will be provided to the new train for a possible addition of a dedicated decay heat removal and cleanup system located in its own permanent structure.

To facilitiate early completion of design and installation of these system modifications, system functional capability following a seismic event has not been a design requirement. If a seismic event should occur and damage the modified systems, the seismic Category I TMI-2 Decay Heat Removal System and Reactor Coolant Makeup System could be used to remove core decay heat and control primary system pressure as necessary.

More detailed descriptions of these modifications are included in the following pages.

All piping connections will be welded. These modifications are expected to be completed and the system ready for operation by the middle of May.

3) Systems Evaluations

The system design as proposed will meet the necessary secondary side requirements for decay heat removal with either forced or natural circulation through the reactor core. All required modifications will be made to accomplish this purpose. This system is completely independent and separate from steam generator "B" during all intended modes of operation with the exception that the Nuclear Services River Water System and Secondary Services River Water System will be shared by both loops of steam generator cooling.

The system will not be provided with redundant active components. However, a single active failure within the system will not compromise natural circulation of the primary system in that the secondary cooling loop through steam generator "B" will continue to operate (see staff evaluation of TMI-2 natural circulation performance).

The flowrates predicted through each of the heat exchangers will provide adequate cooling based on an assumed heat load of 30×10^6 Btu/hr (RCP operating + 3 MWDH). Operating performance and design parameters for the system are as follows:

Steam Generator A Modified System

System Location	Operating Conditions Pressure (psig)	Temp. (°F)	Flow (gpm)	Design Pressure (psig)
New Pump Discharge	670	100	3000-5000	800
New Pump Suction (New Heat Exch. Disch. Shell Side)	500	100	3000-5000	600
New Pump Recirc.	670	100	Pump min. flow	800
New Heat Exchanger Supply (Shell Side)	670	120	3000-5000	800
NSRW Supply to New Heat Exchanger (Tube Side)	100	85	6000	150
NSRW Return from New Heat Exchanger (Tube Side)	100	95	6000	150

The system flow arrangement has been selected to minimize fouling effects by maintaining Nuclear Services River Water on the tube side of the new heat exchanger. System operating temperature indicated for NSRW supply and return are design values.

b. Modifications to Steam Generator "B" for Water Solid Operations

1) Design Concept

The short and long term concept for water solid operation of Steam Generator B have been consolidated. One concept will be utilized. Water will be circulated by the new pump through the tube side of the new heat exchanger and into the secondary side of steam generator B in a closed loop to remove heat from the steam generator. The secondary

2. Steam Generator Modifications

a. Modifications to Steam Generator "A" for Water Solid Operations

1) Design Concept

The short and long term concept for water solid operation of Steam Generator A have been consolidated. One concept will now be utilized. Water will be circulated by the new pump through the shell side of a new heat exchanger and into the secondary side of the steam generator A in a closed loop to remove heat from the steam generator. The tube side of the new heat exchanger will be cooled by the existing Nuclear Services River Water System (NSRWS) which supplies water from the river and returns it to the mechanical draft cooling tower. Refer to Figure 1 for a schematic of this flow path. Provisions will be made for system pressure and expansion control by utilizing the existing 3rd stage feedwater heater shell and its nitrogen supply as a pressurized surge tank. The design also includes provisions for sampling, demineralization and chemical addition capability. For the initial phase of operation, all valves, will be manually operated and instrumentation will provide local read out.

The new loop of heat removal equipment has been designed to operate at a pressure higher than the expected reactor coolant system pressure thus assuring in-leakage of secondav system liquid into the primary system in the event of steam generator tube leakage.

2) Modification

This scheme will involve installation of a new high pressure train consisting of a pump, heat exchanger, valves and piping located in the turbine building basement. The loop will be connected to the main steam turbine bypass line between the connection to the main steam lines and the condenser, and to the main feedwater line between the feedwater pump and 3rd stage feedwater heater FW-J-6A. Horizontal runs of piping will be supported for static loads and secured to supporting structures to prevent lateral motion. Vertical runs of piping will be secured to permanent structural members as required. Additional piping will be required for the surge tank (3rd stage feedwater heater FW-J=6A), chemical addition tank and demineralizer. In addition, the interconnections between the A and B feedwater heater trains will be broken and capped off.

Jumper pipes will be installed between the existing Nuclear Services River Water System (NSRWS) the existing Secondary Services River Water System (SSRWS) to provide cooling water to the tube side of the new heat exchanger. The safety classification of the Nuclear Services River Water System will be maintained by providing double isolation valves.

side (shell side) of this new heat exchanger will be cooled by the Secondary Services Closed Cooling Water (SSCW) system which is, in turn, cooled by the Nuclear Services River Water System, by way of the mechanical draft cooling tower. Refer to Figure 2 for a schematic of this flow path. Provisions will be made for system pressure and expansion control by utilizing the shell side of the existing 3rd stage feedwater heater with its existing nitrogen supply and thus will operate as a pressurized surge tank. The design also includes provisions for sampling, demineralization, and chemical addition capability. For the initial phase of operation, all valves will be manually operated and instrumentation will provide local read out.

The first intermediate loop of this scheme (that portion of the secondary system which removes heat directly from the steam generator) has been designed to permit normal operation at a pressure higher than the reactor coolant system pressure, thus assuring inleakage of secondary system liquid into the primary system in the event of steam generator tube leakage. However, the expected mode of operation from which to initiate natural circulation would involve higher primary side pressures.

2) <u>Modification</u>

This scheme will involve installation of a new high pressure train consisting of a pump (LTB-P-1), heat exchanger (LTB-C-1), valves and piping located in the turbine building basement. The loop will be connected to the main steam line at a 10" drain pot between the main steam isolation valve and the stop valve, and to the main feedwater line between the feedwater pump and 3rd stage feedwater heater FW-J-6B. Piping supports will be similar to those provided for Steam Generator A modifications. Additional piping will be required for the surge tank (3rd stage feedwater heater FW-J-6B), chemical addition tank (LTB-T-2), and demineralizer. In addition, the inter-connections between the A and B feedwater heater trains will be broken and capped off.

Connections will be made to the existing secondary services closed cooling water system supply and return lines for cooling the shell side of heat exchanger LTB-C-1.

Jumper pipes will be installed between the existing Nuclear Services River Water System and the existing Secondary Services River Water System to cool the tube side of the Secondary Services Closed Cooling Water System. The safety classification of the Nuclear Services River Water System will be maintained by providing double isolation valves.

All piping connections will be welded. These modifications are expected to be completed and the system ready for operation by May 7, 1979.

3) Systems Evaluation

The system design as proposed will meet the necessary secondary side requirements for decay heat removal with either forced or natural circulation through the reactor core. All required modifications will be made to accomplish this purpose. This system is independent and separate from steam generator "A" during all intended modes of operation with the exception that the Secondary Services River Water System will be shared by both loops of steam generator cooling.

The system will not be provided with redundant active components. However, a single active failure within the system will not compromise natural circulation of the primary system in that the secondary cooling loop through steam generator "A" will continue to operate (see staff evaluation concerning TMI-2 natural circulation performance).

The flowrates predicted through each of the heat exchangers will provide adequate cooling based on an assumed heat load of 30×10^6 Btu/hr (RCP operating + 3 MWDH). Operating performance and design parameters for the system are as follows:

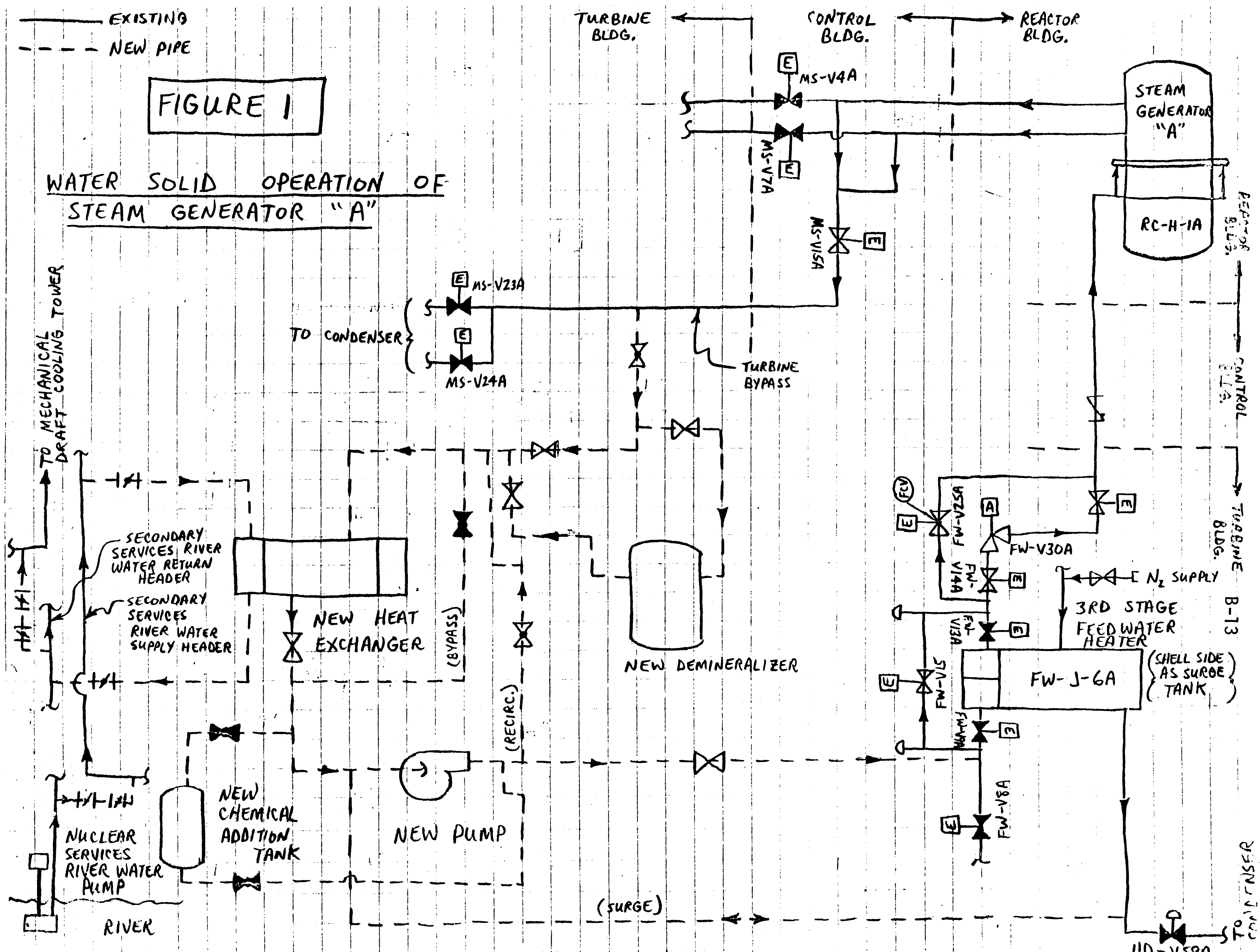
Steam Generator B Modified System

System Location	Operating Conditions Pressure (psig)	Temp (°F)	Flow (gpm)	Design Pressure (psig)
New Pump Discharge	670	100	3000-5000	800
New Pump Suction (New Heat Exch. Disch., Tube Side)	500	100	3000-5000	600
New Pump Recirc.	670	100	Pump min. flow	800
New Heat Exchanger Supply (Tube Side)	670	120	3000-5000	800
SSCW Supply to New Heat Exchanger (Shell Side)	150	72	4000	
SSCW Return from New Heat Exchanger (Tube Side)	100	59	5000	
NSRW Return from SSCW Heat Exchanger (Tube Side)	100	71	5000	

The system flow arrangement has been selected to minimize fouling effects by maintaining Nuclear Services River Water on the tube side of the Secondary Services Closed Cooling Water heat exchanger.

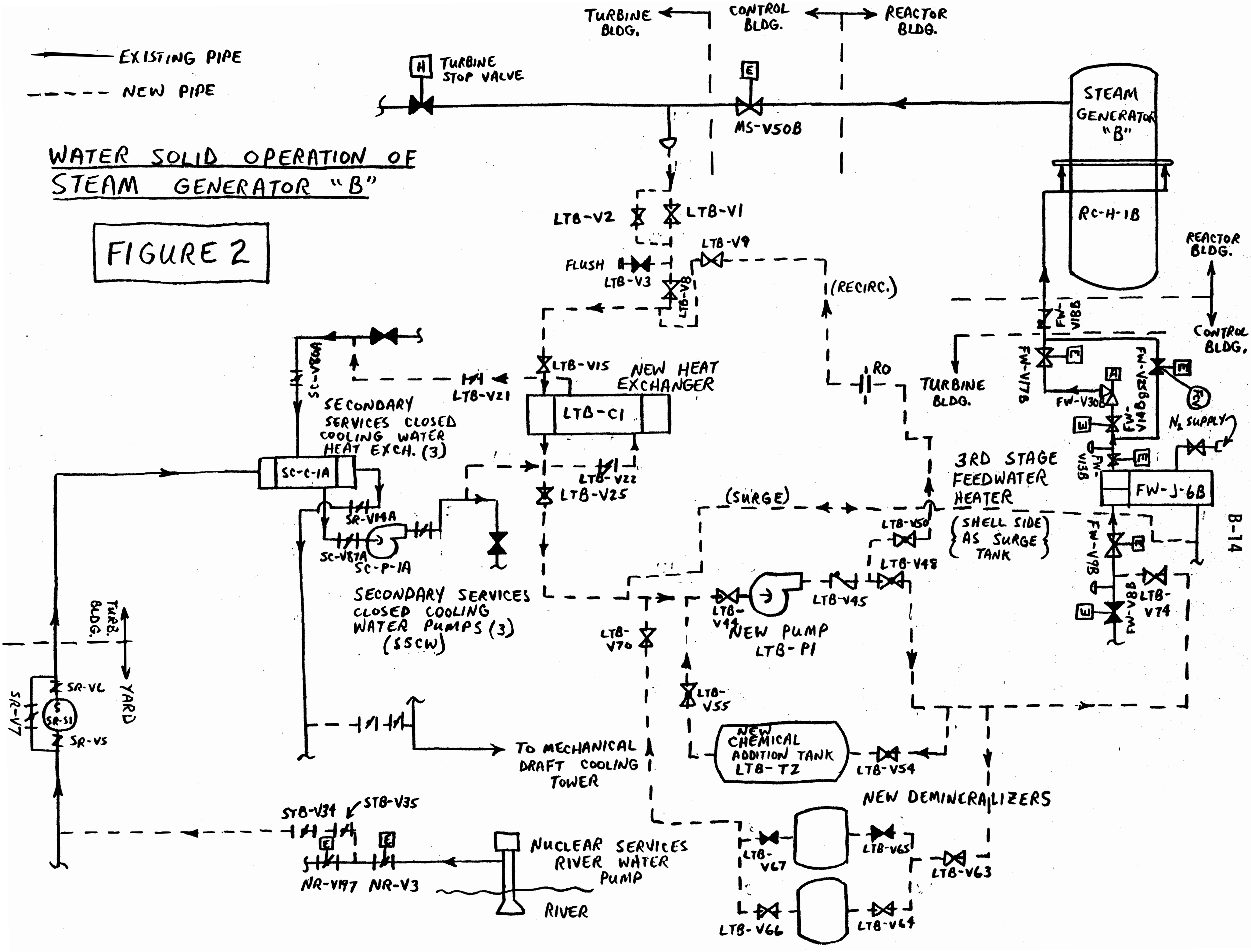
C. Mechanical System Design - Steam Generator "A" and "B" Modifications

All components and supports of both nuclear class and non-nuclear will be designed or verified to have been originally designed for the maximum loads that they could be exposed to during testing startup, and expected operation of the system, i.e., pressure, temperature deadweight, pump vibration, etc. The component design structural information is listed in Table 1.


EXISTING
NEW PIPE
FIGURE 1
WATER SOLID OPERATION OF STEAM GENERATOR "A"
TURBINE BLDG.
CONTROL BLDG.
REACTOR BLDG.
STEAM GENERATOR "A"
RC-H-1A
MS-V4A
MS-V7A
MS-V15A
MS-V23A
MS-V24A
TO CONDENSER
TURBINE BYPASS
TO MECHANICAL DRAFT COOLING TOWER
SECONDARY SERVICES RIVER WATER RETURN HEADER
SECONDARY SERVICES RIVER WATER SUPPLY HEADER
NEW HEAT EXCHANGER
(BYPASS)
(RECIRC.)
NEW DEMINERALIZER
FCV
FW-V25A
FW-V30A
FW-V14A
FW-V13A
FW-V15
FW-V9A
FW-V8A
N2 SUPPLY
3RD STAGE FEEDWATER HEATER
FW-J-6A
(SHELL SIDE AS SURGE TANK)
NEW CHEMICAL ADDITION TANK
NEW PUMP
NUCLEAR SERVICES RIVER WATER PUMP
RIVER
(SURGE)
TO CONDENSER
B-13

WATER SOLID OPERATION OF STEAM GENERATOR "B"

FIGURE 2



A specific concern that we have addressed in our review is the structural adequacy of that portion of the Main Steam piping system which as incorporated into the OTSG cooling system will contain solid water in lieu of the pressurized steam for which it was designed. Components in the system will not experience any significant dynamic loads. Special precautions will be taken during initial filling and startup of the system to minimize the potential for water hammer. Loads that will be experienced include pressure, deadweight of water, and thermal expansion. Since the system will be operated at a maximum pressure of about half the design pressure of the piping and its maximum operating temperatures will be considerably lower than the design temperature of the main steam piping; stresses resulting from these loads will be minimal.

After assembly and prior to initial operation of the plant, the existing piping was hydrostatic tested and at that time was water filled. Thus the piping and its supports have been demonstrated to be adequate for the weight of the water.

In order to minimize piping deflection, the licensee has specified that selected spring hangers be pinned. We concur with this requirement.

Design Bases Loads

It should be noted that all ASME Section III CL. 2 components used in the cooling system were designed for seismic Category I service. However all of these components, both those that are part of the original TMI-2 Main Steam and Feedwater piping system and those obtained from other nuclear sites to be incorporated into the OTSG cooling system, are being utilized in a system with different response characteristics from that for which they were initially designed or are operating with a fluid media different from that for which they were seismically qualified, i.e., some components designed for operation on steam during a seismic event as opposed to water filled as in the present system. Thus because of these differences from the original seismic design requirements, which can affect seismic response, these components should not be considered seismically qualified as installed in the proposed cooling system, solely on the basis of their original qualifications. Additional work would be required to evaluate the seismic capability of these components for this application. However, seismic capability of these system modifications is not a necessary acceptance criterion; therefore, no additional seismic evaluation of this system is planned.

Evaluation Conclusion

We have concluded that the Licensee has specified components designed and fabricated in accordance with acceptable industry codes or standards and will take into account the loads associated with startup, testing, and the planned system operation.

The use of components that are in conformance with these criteria provides adequate assurance that structural integrity of the OTSG "A" and "B" cooling system will be maintained.

Table 1

Component Design Structural Information

Pumps A & B System

ASME Section III CL. 2

Design Temperature - 350°F
Design Pressure - 700 psig.

Heat Exchangers

A System

ASME Section VIII
Design Pressure: 150 psig (tube side)
600 psig (shell side)

B System

ASME Section III CL. 2

Design Temperature: 350°F (Both Shell and Tube Sides)
Design Pressure: 675 psig (ube side)
200 psig (shell side)

3rd Stage Feedwater Heater (shell side as surge tank)

ASME Section VIII

Design Pressure: 1000 psig

Steam Generator

Secondary Side - ASME Section III CL. 2

Piping, Valves and Mis. Tanks (minimum requirements)

Piping - ANSI B31.1
Misc. Tanks - ASME Section VIII Div. 1
Valves - ANSI B16.5 and B16.34

From feedwater check valves outside containment to the main steam isolation valves

ASME Section III CL.2

Supports

ANSI B31.1 or as described in text.

d. <u>Structural Design - Steam Generator "A" and "B" Modifications</u>

Modification of Steam Generator "A" and Steam Generator "B" involves installation of one pump and one heat exchanger for each steam generator. The new equipment will be installed in the north-west end of turbine building at elevation 281'-8", between column lines TJ and TK in north-south direction and T41 and T44 in east-west direction. The enclosed Figure 1 shows the general area of location of the above equipment.

The details of the equipment is as follows:

The heat exchanger for steam generator "B" weighs 36 kips wet and is supported on two saddles, 2'-0" by 4'-0" each. The pump for this steam generator weighs 13.7 kips.

The heat exchanger for steam generator "A" weighs 135.8 kips wet. It is approximately 33 ft. long and is supported on two saddles 8 in. wide and 4'-8" long located 13'-0" apart.

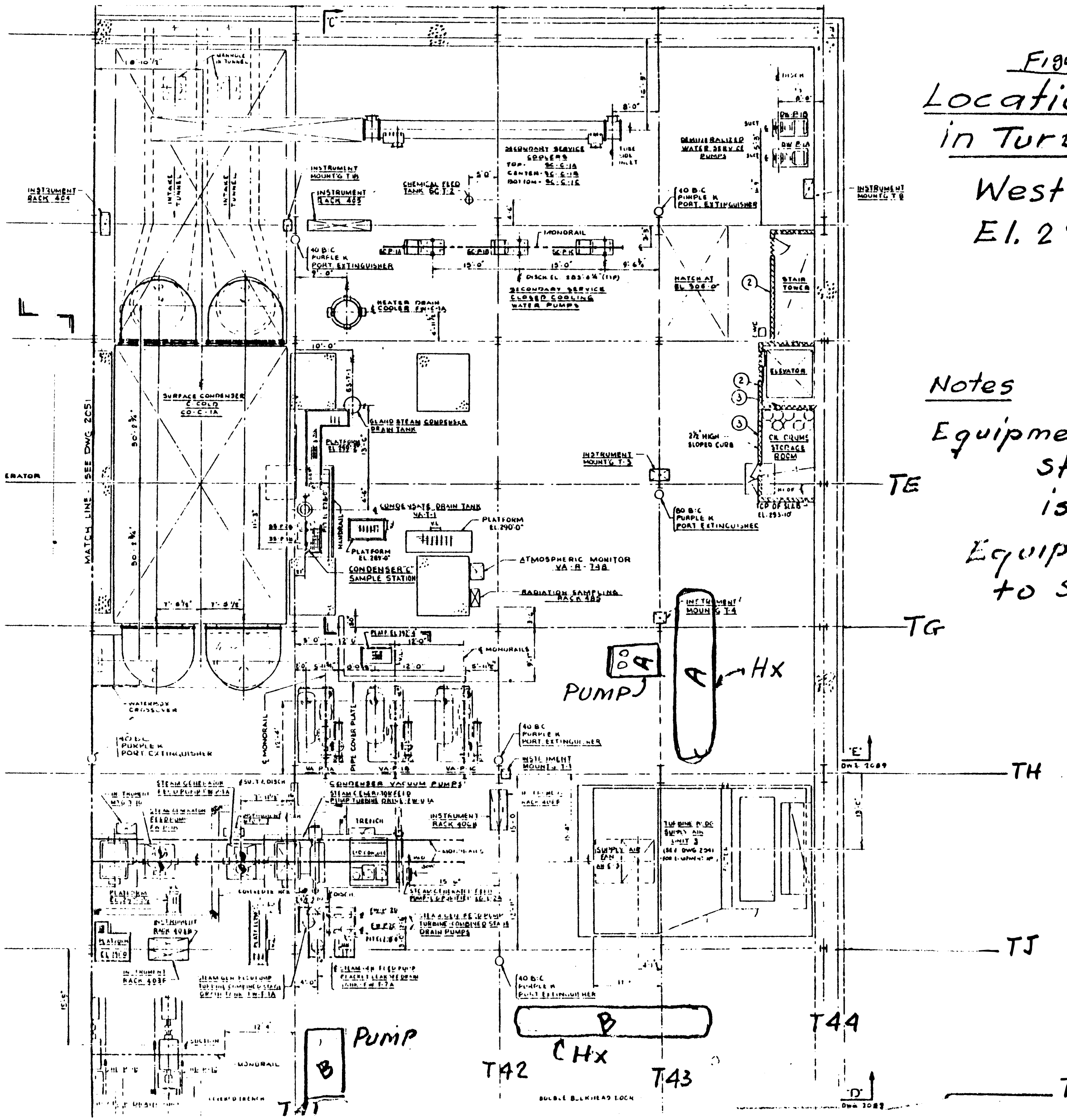
Examination of the existing structural drawings of the turbine building area, where this equipment is to be installed, reveals that the base slab is four feet thick, with #11 rebars at 12 in. spacing, each way, top and bottom. Top of the structural concrete base mat is at the elv. 278'-0". The base mat is covered with a 3'0" layer of lean concrete fill plus approximately 8 in. wearing surface, bringing the top elev. to 281'-8". Assuming 45 degree distribution of the load, and on the basis of the allowable concrete bearing pressure, the

Figure 1

Location of Equipm. in Turbine Bldg

West side

El. 281'-8"



Notes

Equipment pertaining to Steam Gen. "A" mod. is marked "A"

Equipment pertaining to Steam Gen. "B" mod. is marked "B"

LEGEND

= FIRE R.

— HOURLY F.

KEY PLA

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BURNS AND ROE ENGINEERS AND CONS.

licensee estimated that the maximum load that can be placed over one square foot of the area is 90 kips. Since the heaviest piece of equipment to be installed in this area is the heat exchanger, 135.8 kips, which when distributed over the area of the supporting saddles (2 x 4.67' x .67' = 6.1 sq. ft) delivers the head to the floor of 22 kips, the licensee concluded that the slab is capable to support the load. The analysis was performed for the dead and live loads only, under static conditions.

We have concluded that the licensee has performed his analysis in accordance with the methods and procedures which are specified by the appropriate codes and standards. The use of these methods provide a reasonable assurance that the structural elements affected by this modification will perform their intended function.

e. <u>Instrumentation and Control (OTSG "A" and "B" Cooling)</u>

The following new instrumentation (listed below) is being provided on the OTSG "A" and "B" Long Term Cooling Systems.

1. Radiation Monitor (* 'B' only)
2. Loop flow Indication*
3. Steam Generator Heat exchanger, Shell Side Temperature - T_{in} (*A only), T_{out}
4. Heat exchanger, Tube Side Temperature - T_{in}, T_{out}
5. Pump Pressure - Suction and Discharge
6. Loop Temperature*
7. Loop Pressure

*Redundant sets of instrumentation are marked with an asterisk.

Control room alarms will be provided for radiation as a result of high S/G out-leakage. For initial installation and system operation, only local indication will be available mounted near the sensors in the turbine building. Similarly, all controls will be local.

S/G leakage into the cooling loops is to be sensed by victoreen model 855 area monitors strapped to the piping just down stream of the tie-in to the main steam line. Loop flow rate is to be sensed by 2 (two) barton model 200 differential pressure mechanical indicators across a single permutit orifice plate. All new temperature sensors/indicators are specified to be ashcroft (5 - inch code 50 El) bi-metal, liquid-filled thermometers. Loop pressures are to be sensed by ashcroft 1279 (bourdon system) mechanical pressure gauges. Table 1 provides additional instrumentation characteristics.

The system design criteria include the requirement to provide control and sensor read-out to the main control room on an expedited basis. Due to the time constraints placed upon initial system operation, we find the above design criteria to be acceptable. **The specific details of the design associated with providing control and sensor read-out to the control room have not been developed at this time.**

OTSG "B" COOLING LOOP - TABLE 1

VARIABLE	SENSOR TYPE	LOCATION
S/G "B" Leakage (primary to secondary)	Strap-on gamma Detectors Victoreen	1 monitor on existing main steam turbine bypass header ("A" loop only)
		2 monitors on new pipe just downstream of existing main steam header ("B" loop only)
Loop Flow		
Total Flow	Orifice Plate-Permutit	2 indicators in new pipe just downstream of existing main steam header ("A" loop)
		2 indicators in new pipe just upstream of existing eedwater header ("B" loop)
	Bimetallic Thermometer-Ashcroft	In new piping just off mainsteam header
Heat Exchanger		
Shell Side, Temperature	Bimetallic Thermometer-Ashcroft	At heat exchanger
T_{in}, T_{out} Tube Side, Temperature T_{in}, T_{out}	Bimetallic Thermometer-Ashcroft	At heat exchanger
Pump Pressure		
Suction and Discharge	Pressure Gauges-Ashcroft	At discharge and suction of pump
Loop Temperature	Bimetallic Thermometers-Ashcroft	Downstream of pump recirc.
Loop Pressure	Pressure Gauge-Ashcroft	Upstream of heat exchanger

f. Radiological Evaluation (Steam Generator A/B)

The radiological considerations which are integral to the water solid steam generator secondary side cooling method will assure that radioactive effluents from contaminated systems will be controlled and minimized. Additional precuations will also be made to minimize occupational radiation exposures to operating personnel.

The secondary coolant presently in the "B" steam generator is contaminated due to the initial primary to secondary leakage which occurred on March 28, 1979. The measured radioactivity concentration and curie content is estimated in Table 1. Under normal conditions the secondary coolant pressure will be maintained at a value greater than the primary system pressure such that if steam generator leakage flow paths are available, the highly contaminated primary coolant will not enter the secondary coolant. However, it is expected that transients of short duration may occur such that a reverse pressure gradient could introduce additional radioactivity into the secondary coolant. To alert the system operators of such a condition, indicators and alarms for pressure and radioactivity in the secondary coolant have been provided. These indicators will alert the operator of an adverse condition so that corrective action can be taken prior to significant additional contamination of the secondary coolant.

The steam generator secondary coolant system and secondary services closed cooling water system will be periodically sampled and analyzed to determine if heat exchangers are starting to leak. Samples will be taken at a frequency of at least weekly or at any time there are indications that possible leakage may be occurring, e.g., increase in the steam generator closed cooling loop surge tank levels.

Leakage may also occur at various mechanical connections in the secondary cooling system. To the extent practical, locations where leakage is likely, e.g., valves with a known leakage history will be evaluated and a program to minimize leakage implemented. Even with a leak minimization program leakage of contaminated liquids to the floor drain system may still occur. This leakage should flow into the floor drain system and be collected in the turbine building sump, the turbine building sump pumps will be operated in a manual mode with an analysis of the sump water radioactivity content being made prior to sump pump operation. If the radioactivity content is low, the water can be discharged through its normal flow path to the river. If the radioactivity content is high, such that discharge would exceed technical specification limits, the sump water will be pumped to an appropriate radwaste system for treatment. Adequate free volume in the radwaste system will be provided for such contingencies.

Isolation between the secondary cooling system and services systems, e.g., nitrogen and demineralized water supplies to the surge tank, will be provided to prevent back flow of contaminated water.

Gaseous effluents from this system should be negligible. The noble gas inventory in the "B" steam generator is negligible because the steam generator was vented. Airborne radioiodine releases should also be negligible because the secondary cooling system is not vented (a nitrogen blanketed surge tank) and the low secondary cooling system temperature (100°F) results in a low air/water partition factor which reduces the volatility of the radioiodine.

The licensee is providing a demineralizer system to maintain secondary water chemistry and to reduce radioactive contamination. Details on this system are not available at this time. The demineralizers will be shielded and designed to permit resin bed disposal as radioactive waste. The shielding and spent resin handling system will also be designed to minimize occupational radiation exposures, e.g., the use of disposable demineralizers.

Following the TMI-2 incident there has been no indication of primary coolant leakage past the "A" steam generator tubes. Since the primary to secondary system pressure differential will result in in-leakage only, precautions similar to those evaluated above for the "B" steam generator modification are not required.

Table 1

"B" Steam Generator Radioactivity Content
(Activities based on 4/19/79 sample reported by B&W)

	April 19, 1979			May 9, 1979	
	Half-Life	Radioactivity concentration (uCi)/gm	System Inventory (Ci) (1)	Radioactivity concentration (uCi)/gm (3)	System Inventory (Ci) (2)
I-131	8.05d	.93	60.8	8.9×10^{-2}	9.9
Cs-134	2.1y	9×10^{-3}	.59	5.3×10^{-3}	0.59
Cs-136	13d	$8.x \times 10^{-3}$	.54	1.6×10^{-3}	0.18
Cs-137	30y	3.4×10^{-2}	2.2	2.0×10^{-2}	2.2

(1) S.G. water level = 368 inches = 6.5×10^7 cc

(2) S.G. full at 625.5 inches = 1.1×10^8 cc

(3) Assumes decay from 4-19 (21 days) plus dilution from filling the steam generator.

3.0 STANDBY REACTOR COOLANT PRESSURE CONTROL SYSTEM

A. System Evaluation

1) Description

A standby reactor coolant pressure control and makeup system has been proposed by the licensee. This system would serve as a backup to the CVCS and maintain reactor coolant system pressure with the pressurizer filled solid with water. Primary coolant system pressure will be maintained even with the loss of pressurizer instrumentation and inoperative pressurizer heaters. Also, the pressure control system will be designed to provide adequate NPSH to the reactor coolant pumps if they are needed.

The stanby reactor coolant pressure control system will consists of passive components (a series of water storage tanks and a surgetank with nitrogen blanket for pressure control) and active components (charging pumps). The system will control reactor coolant pressure over the range of 100 psig to 750 psig.

The passive reactor coolant pressure system which would be operated initially with local control. Additional instrumentation and remote control will be incorporated to permit automatic operation of the system. The active pressure control portion would resupply water to the surge tanks with added capability of providing additional makeup water directly to the RCS if needed.

2) Modifications

The standby reactor coolant system pressure control and makeup system will involve installation of two 900 gallon capacity water tanks and one surge tank (all will be of the Westinghouse Boron Injection Tank design), nitrogen bottles, two 40 gpm positive displacement pumps, a degassed borated water supply tank, valves, and piping. This system will be connected between the discharge side of the high pressure makeup system downstream of valve MU-V-144C and upstream of valve MU-V16C. All the components will be placed in the fuel handling building.

The above modification will establish a flowpath of makeup water and pressure control through the normal makeup lines that connect with the reactor coolant loop cold legs. Chemical control of the degassified borated water used in the pressure control system will be provided by the present chemical addition system. Connections will be provided to accommodate the addition of boric acid, H_2, demineralized water and hydrazine, LiOH and NaOH. The degassed water tank will be replenished via piping connection from borated water transfer pump and boric acid batching tank. The boron concentration will be maintained in the range of 2200 to 4000 ppm. Figure 1 depicts the proposed pressure control system and interface connections to the existing systems.

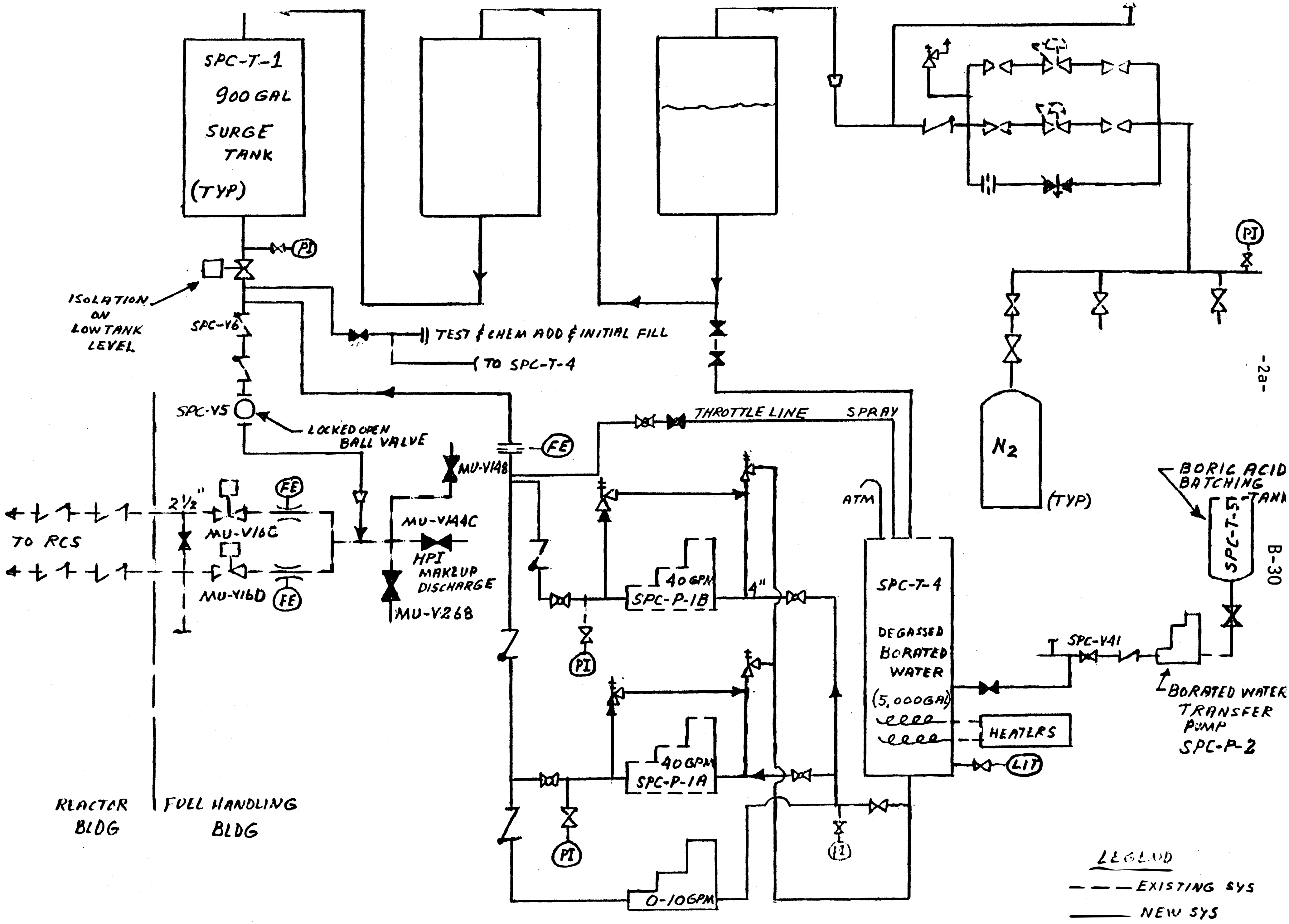

FIGURE 1. STANDBY REACTOR COOLANT PRESSURE CONTROL SYSTEM

3) Evaluation

If required, the passive portion of the standby pressure control and makeup system is designed to provide peak initial 500 gpm injection rate to the reactor coolant system. The 500 gpm injection rate will be adequate to provide primary system makeup for certain transient events that can cause considerable shrinkage in the RCS. Because of the finite inventory (1800-2000 gal) this injection will decrease as the discharge proceeds. Also, the passive portion of the system will be designed to provide continuous makeup of 4 gpm for 8 hours.

Sufficient makeup volume requirement can be met by the proposed RCS pressure control system for moderate system perturbations and for the following postulated transient event: loss of natural circulation cooling due to a loss of all secondary side cooling with restart of one secondary cooling loop following a hot leg temperature rise of 50°F.

For this event the licensee has shown a total volume change of 1900 gallons which can be made up by the proposed system. However, the pressure control system is not designed for makeup requirements of more severe transients such as a sudden complete loss of natural circulation for a period of 2 hours, followed by an RC pump start. Procedures which permit reactor coolant pump restart following loss of natural circulation would require the availability of other makeup systems such as the HPI in addition to the proposed pressure control system.

The reactor coolant pressure control and makeup system will not provide letdown capability of reactor coolant caused by overpressurization events. Reactor coolant fluid expansion will be relieved by one or a combination of the following current components and systems: (1) normal letdown line (through the letdown coolers); (2) maintaining letdown with concurrent termination of makeup/seal injection; (3) continued reactor coolant pump seal return flow; (4) opening of pressurizer vent valve; and (5) lifting of the pressurizer relief valve.

Piping integrity of the reactor coolant pressure control and makeup system has been examined for postulated overpressurization events such as inadvertent startup of an HPI pump. This system will have a design pressure of 1000 psig except for the section from the HPI makeup line back through the second isolation check valve which will have a design pressure of 1500 psig. Overpressurization protection of the latter piping section when the HPI/makeup pump is started will be provided by installation of a relief valve about the HPI/makeup pump set at 1000 psig and a relief flowrate of 525 gpm. Check valves located downstream of the HPI/makeup pumps that are inside the Reactor Building will also provide protection to this system from inadvertent overpressurization of the RCS due to any other causes. The criteria and procedures for letdown and overpressure control of the RCS will be established prior to the operation of this system.

The proposed reactor coolant pressure control and makeup system may not meet makeup demand following a depressurization event of the primary system such as inadvertent opening of the pressurizer PORV (with downstream isolation valve open). If not isolated in time, the system would be drained resulting in entrainment of nitrogen into the reactor coolant system. To prevent such an occurrence the system will automatically isolate on low level in the water tank. In the event of loss of offsite power this valve will fail in the "as is" position. Since this position is the preferred position for normal operation and it would also be the preferred position in the event of loss of offsite power. Also, an alarm will be annunciated locally and in the control room when the isolation valve is not fully open.

We have examined for single failures that can disable the pressure control and makeup system. The discharge valve SPC-V5 is a single failure point, however it will be a manually operated ball valve positioned to an open position, and then locked in that position. For simplicity of design and installation, we have not required redundant valves to meet single criterion to insure system isolation capability. A redundant charging pump will be available to fill the water tanks in response to tank level reduction.

Requirements for preoperational flow test of the RCS pressure control system will be determined pending a review of the margin suggested by analysis for peak makeup flow demand and what the system is designed to provide.

We conclude that the proposed system will maintain reactor coolant pressure control for normal water solid conditions and provide sufficient makeup water for a wide range of expected transient events that would cause shrinkage in the RCS. Emergency procedures will be u ed for additional makeup capability that may be required to mitigate more severe and less probable transient events.

B. Mechanical Design

1) Description

The Reactor Coolant Pressure Control System is madeup of several water supply tanks, positive displacement pumps, valves, and piping. The applicable design codes and standards used for the design of these components are provided in the Table 1.

Table 1

Applicable Design Codes or Standards

Water Supply Tank (Passive System)

ASME Section III CL. 1 - S.S.

Charging Pumps

ASME Section III CL.2

Piping

ANSI B31.1 (minimum requirement)

Degassified Water Supply Tank

ASME Section VIII Div. 1

Component Supports

ANSI B31.1

2) Design Loads

Pressure, pump vibration, component and fluid deadweight, and maximum anticipated pressure surge forces were considered in developing design loads for the RCS makeup system. The staff considers these to be acceptable load considerations for this application. The staff has not required the RCS pressure control and makeup system be evaluated for seismic load capability.

3) Other Considerations

Welded construction will be used wherever possible to minimize the potential for system leakage. Components will be fabricated from stainless steel or carbon steel clad with stainless steel.

4) Evaluation Conclusion

We conclude that the licensee has specified components that will be designed and fabricated in accordance with acceptable industry codes or standards and will taken into account the loads associated with startup, testing, and expected system operation.

The use of components that are in conformance with these cirteria provides adequate assurance that structural integrity of the Reactor Coolant Pressure Control System will be maintained.

C. Structural Design

The system includes the following major equipment:

a) Three water tanks, 900 gal, capacity. Each weigh 20 kips empty and 27.5 kips when full of water. Each tank will be supported on four 12 in. by 12 in. plates.

b Borated water tank weighing 60 kips supported on four 6 in. plates.

c) Two - 100 HP pumps weighing 5.4 kips each.

The enclosed Figure 2 (2 sheets) show the conditions of the original structure. The area is located between columns AP and AT in south-north direction and columns A65 and A67 in east-west direction. The slab is three feet thick and the reinforcing steel is #8 at 9 in. top and bottom in north-south direction and #8 at 6 in. top and bottom in the east-west direction. The compressive strength of the concrete is 3000 psi and the yield stress of the reinforcing steel is 60 ksi. The licensee has analyzed the slab for the additional loads resulting from the new equipment and concluded that the stresses will be within the allowables using generally accepted codes. The analysis was performed for the static conditions only. We conclude that the licensee has performed his analysis in accordance with the methods and procedures which are specified by the appropriate codes and standards. The use of these methods provide reasonable assurance that the structures affected by this modification will continue to perform their intended function.

Fig 2 (Sheet 1)

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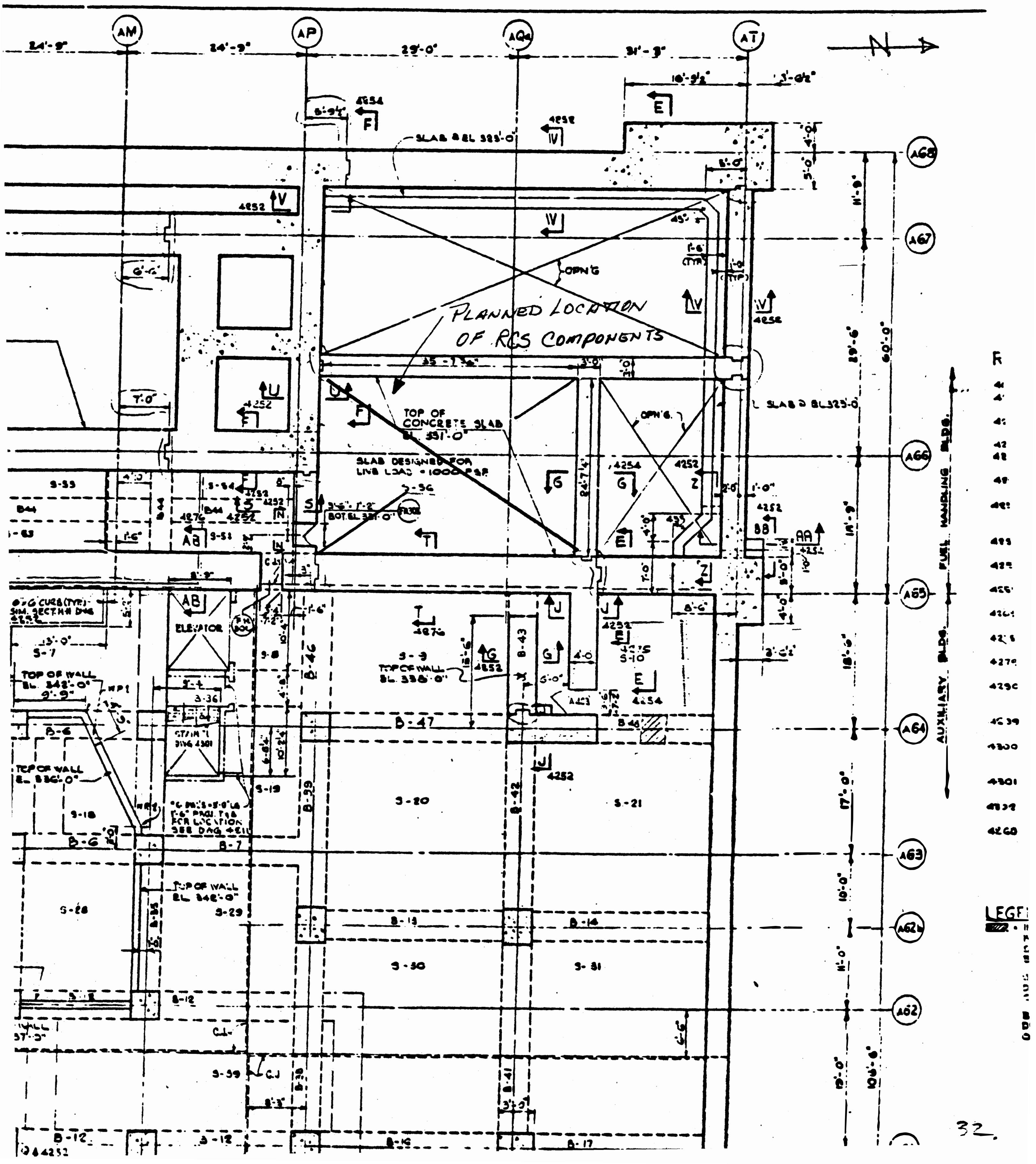
-9a-

AUXILIARY & FUEL HANDLING BLDG
EL 328'-0"

W.O: 2555

RCS PRESSURE CONTROL SYSTEM

BURNS & ROE
DWG #4250



We have required that the design include automatic isolation capability to preclude the addition of nitrogen into the Reactor Coolant System. The licensee has responded by providing an automatic close signal to the motor operated isolation valve initiated by low level in the tank nearest the Reactor Coolant System. The isolation valve will be signaled to close when water in the tank is depleted to 300 gallons. Tank pressure taps will be utilized for level indication. (One pressure tap in the tank and the other in the 6-inch piping upstream of the tank). Because of the location of pressure taps and uncertainty of level control to actuate the isolation valve under transient conditions we require functional testing of this control system that would require tank blowdown prior to installation of the pressure control and makeup system. If the test results turn out to be unsatisfactory other means of preventing N_2 insurge into the RCS such as automatic isolation and venting of N_2 supply will have to be considered. We have further required that an alarm be provided to the operator when the differential pressure of the Reactor Coolant System and the pressure control system exceeds a set value. The licensee has complied and the alarm will be annunciated whenever the differential pressure is greater than 50 psi. We find the instrumentation and control aspects of this system as described above to be acceptable.

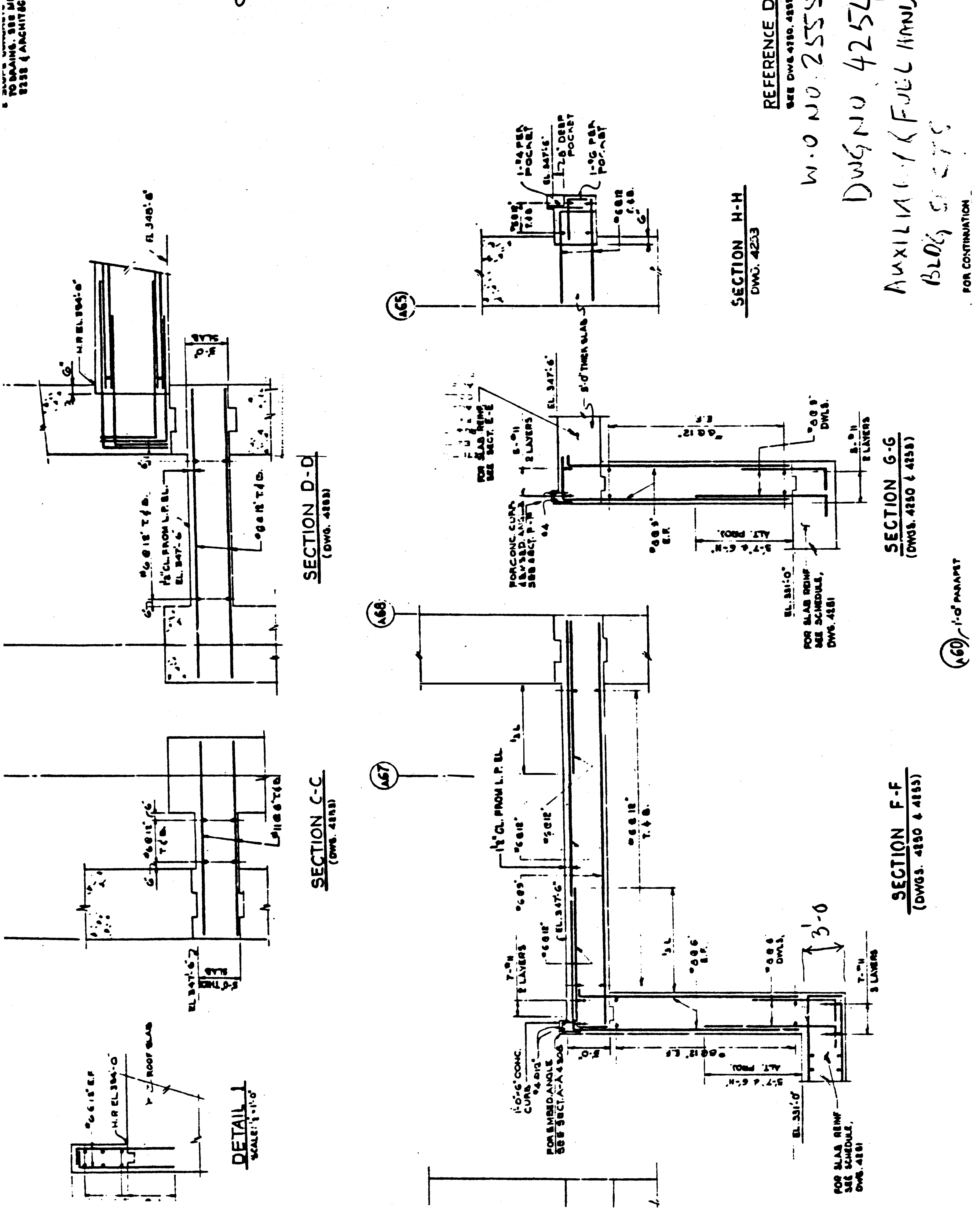
4. Decay Heat Removal System (Existing and Skid-Mounted)

Decay heat removal capability will be provided for the plant for its current operating modes. To accomplish this, the following work and modifications will be undertaken: 1) upgrade the leak tightness of the existing decay heat removal system as required, 2) install an additional skid mounted train of decay heat removal equipment, and 3) consideration of a dedicated long term decay heat removal and primary coolant cleanup system in a permanent structure.

A. Upgrading Existing Decay Heat Removal System

Provisions will be made for conducting a preoperational test of each loop of the existing DHR system. Locations of system leakage will be identified using television cameras installed at key locations. Once leakage paths are identified, they will be corrected if possible, thereby providing as leak tight a system as is practical. Leakage collection capability will also be added to the system where feasible (i.e., collection of leakage around valves). Instrumentation to detect DHR pump vibration will also be installed. The availability of the existing decay heat removal system will depend on the final leak tightness of the system considering the contamination levels present in the primary coolant.

Fig. 2 (Sheet 2)


REFERENCE DWG.
W.O. NO. 2555
DWG NO. 4254
AUXILIARY FUEL HANDLING BLDG
SECTION H-H
DWG. 4253
SECTION G-G
(DWGS. 4250 & 4253)
SECTION F-F
(DWGS. 4250 & 4253)
SECTION D-D
(DWG. 4253)
SECTION C-C
(DWG. 4253)
DETAIL 1
FOR CONTINUATION
1'-0" PARAPET
3'-0

D. Instrumentation and Control

The following instrumentation is to be provided for the Reactor Coolant Pressure Control System:

1. System Water Pressure
2. Water Level in each tank
3. Nitrogen Pressure
4. Make-up Flow
5. Make-up Pump Discharge Pressure
6. Borated Water Storage Tank Level
7. Borated Water Supply Temperature

Redundancy is provided for: Water level (in each tank) and system water flow. Nitrogen pressure is sensed at the cylinders and just downstream of the N_2 pressure regulators.

The design of the system is not predicated on the availability of the pressurizer instrumentation and controls. The ranges of instrumentation have not been provided.

The control aspects of this system are as follows: The first make-up pump will be cut in by a "Low" level signal from the nitrogen pressurized surge tank; the second pump will be cut in by a "Low-Low" level signal; and both pumps will be cut out by a "High" level signal. The controls for the heaters have not been described.

B. Design of New Skid Mounted Decay Heat Removal System

A third train for decay heat removal will be provided. This will involve a tie into the existing decay heat removal system drop line downstream of valve DH-V-3 located in the fuel handling building and ties into the two return lines to the cold legs also located in the fuel handling building (See Figure 1). New lines will be run through the penetration room to an opening cut in the fuel handling building wall and out to a skid located outside the building at grade elevation (304'-6"). This skid will contain a new decay heat removal heat exchanger and two full capacity pumps. The discharge line for the heat exchanger will return through the opening in the fuel handling building to the return line tie-ins. The tie into the decay heat removal drop line will be made by welding an 8 inch weldolet to the pipe with a full penetration weld, dye penetration testing the weld, then cutting the hole in the pipe using a plasma arc cutting process to minimize debris and finally welding the new pipe to the weldolet. A similar procedure would be used for the tie-ins to the two return lines. This procedure should minimize the time that the decay heat removal system will be out of operation. All valves will be electric motor operated and will have two seals with provisions for collecting leakage and directing it to the existing radwaste system. Additional connections will be provided in the new piping outside the fuel handling building for future use in the installation of a dedicated, long life, hardened structure which will contain heat exchangers, pumps, demineralizers and filters for long term decay heat removal and cleanup of primary coolant water.

The secondary side of the new decay heat removal heat exchanger will be cooled by a new separate decay heat closed cooling water system with its own pump, piping and valves. (See Figure 2). This system in turn will be cooled by water from the nuclear services river water system. New connections will be made to this system.

The design of this new decay heat removal system will be compatible with the current primary system decay heat levels and operating parameters. It will require that the pressurizer level be maintained half full at all times to insure adequate NPSH to the DHR pumps. Alternatively, the backup makeup and pressure control system could serve to satisfy this requirement. The new skid mounted DHR system and closed cooling system is completely separate and indepent from the existing DHR but is not designed against single active failure. However, the existing DHR system would be available in this event.

The plan is to have the skid available but not make the connection to the lines penetrating the fuel handling building wall at this time.

In the event this sytem would be required for immediate service, the skid and piping would be shielded to provide all reasonably achievable radiation protection.

NEW WESTINGHOUSE DECAY HEAT REMOVAL SYSTEM

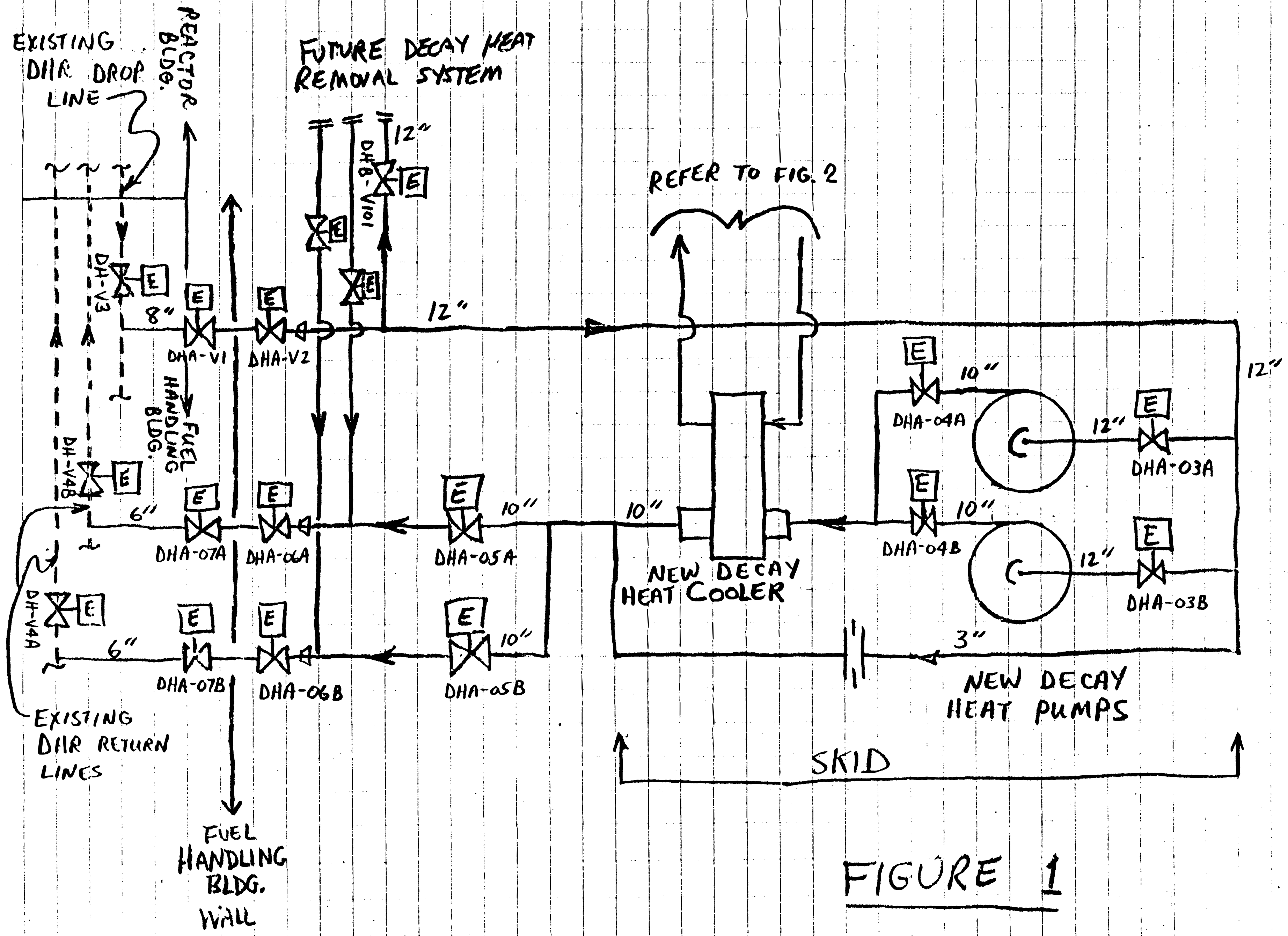
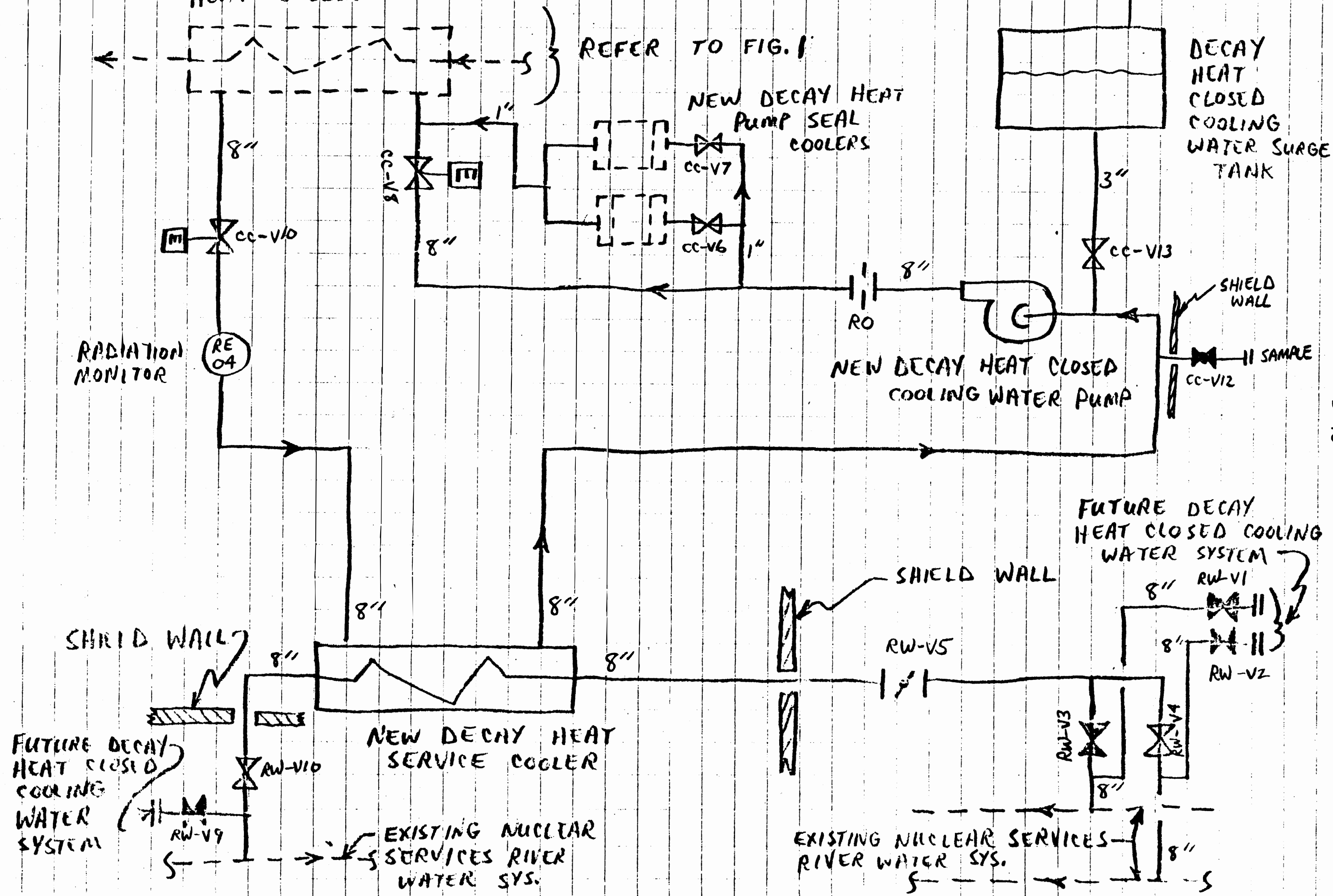


FIGURE 1

Dotted lines - existing system

NEW WESTINGHOUSE DECAY HEAT CLOSED COOLING WATER SYSTEM



Dotted lines - existing system or reference to Fig. 1

FIGURE 2

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The staff has reviewed the expected system performance of this system and concludes that its heat removal capacity is sufficient for decay heat removal from the TMI-2 reactor.

Mechanical Design Considerations

The skid mounted decay heat removal system consists of components such as heat exchangers, pumps, tanks, valves and piping. The applicable design codes and standards used for the design of these components are provided in the table below.

Table 1

APPLICABLE DESIGN CODES OR STANDARDS

Pump and Heat Exchanger

ASME Section III Cl.2

Valves

ASME Section III Cl.1

Piping

Mixture of Type 304 and 316 10" Sch. 40 Stainless Steel sections.

ASTM Material Certification

All welds fully radiographed except for weld-o-let connections to existing DHR piping.

Design Criteria

Loads Considered - Normal operating conditions, - to include operating pressure, dead weight, pump vibration, thermal expansion, and maximum anticipated pressure surges.

Normal stress limits will be met for all piping and components including loads from OBE.

Design Stress Limits Used:

Pumps, Valves, Heat Exchangers - as specified in ASME, Section III for applicable Code Class.

Piping - Stress limits per ANSI B-31.7.

Design Information: Specific for System Tie-in to Existing DHR Piping, Weld-o-let Connection

Reinforcement area of fitting provides a 240 percent margin over the area of the existing DHR piping it replaces.

Pipe supports will be arranged so maximum stress levels at the weld-o-let to DHR pipe interaction will be held to about one third of the B-31.7 allowable stress limit for normal loads.

Weld-o-let to DHR existing pipe welds will be made using a qualified procedure and by welders qualified on weld-o-let to pipe connection mockups.

Because of time constraints, the weld-o-let to pipe welds will not be radiographed; however, the root of the weld will be ground and dye penetrate inspected and the final surface will be dye penetrant inspected. Additionally, the design is being qualified by hydrostatic pressure tests and bending moment tests which apply loads until the simulated existing DHR pipe exceeds its yield strength.

All welds and the cut into the existing DHR pipe will be performed using the plasma arc method. The plasma arc was chosen for the combination of small heat-affected zone and minimum resulting slag which can be cleaned up with relative ease.

Miscellaneous

Valves - Line valves and relief valves will have leakage or discharge fluid piped to a drain tank in the auxiliary building.

Decay Heat Closed Cooling Water System Components

ASME Section III CL.3

For all components

Materials: C.S. piping
S.S. Pump, Valves, Heat Exchanger

All connections welded except for piping to component interfaces which will be flanged.

Evaluation Conclusion:

We have concluded that the Licensee has specified components designed and fabricated in accordance with acceptable industry codes or standards and will take into account the loads associated, with startup testing, and planned system operation.

The use of components that are in conformance with these criteria provides adequate assurance that structural integrity of the Decay Heat Removal System will be maintained.

Structural

Fuel Handling Building Wall Penetration (Aux. Bldg)

In order to provide the skid mounted decay heat removal system a penetration would have to be made through the west wall of the Fuel Handling Bldg., between column lines AC and AF and across column line A68, at elevation 297'-0" (See B&R Drawing 2075). This is approximately seven feet below grade. Excavation outside of the structure will be done by pick and shovel to minimize a possibility of damaging any piping or electrical conduit. The outside wall at that location is reinforced concrete, 5'0" thick. The compressive strength of concrete in this wall is 5000 psig.

The opening is to be made 21" wide 5'1" high on the outside tapering to 4'-1" at the inside face. One reinforcing bar, #18 is to be cut at each face. Figure 3 shows the front view and the cross-section at the opening.

Procedure

At the time of writing of this report various methods of penetration are being studied. One possibility is using the oxygen lance process. A trial penetration is currently being performed by this method. Drilling through the well is also considered. The final method of the wall penetration will be described upon pending results of the test and consultation with the other experts in the field of construction materials.

Instrumentation for New Decay Heat Removal System

The following identifies the instrumentation to be provided for the third DHR System train. A trailer will be used to provide remote control room operation.

The following instrumentation will be provided:

1. DHR pump suction pressure
2. DHR pump suction temperature (DHR cooler inlet temperature)
3. DHR Cooler inlet pressure
4. DHR Cooler discharge pressure
5. DHR Cooler discharge temperature
6. DHR flowrate

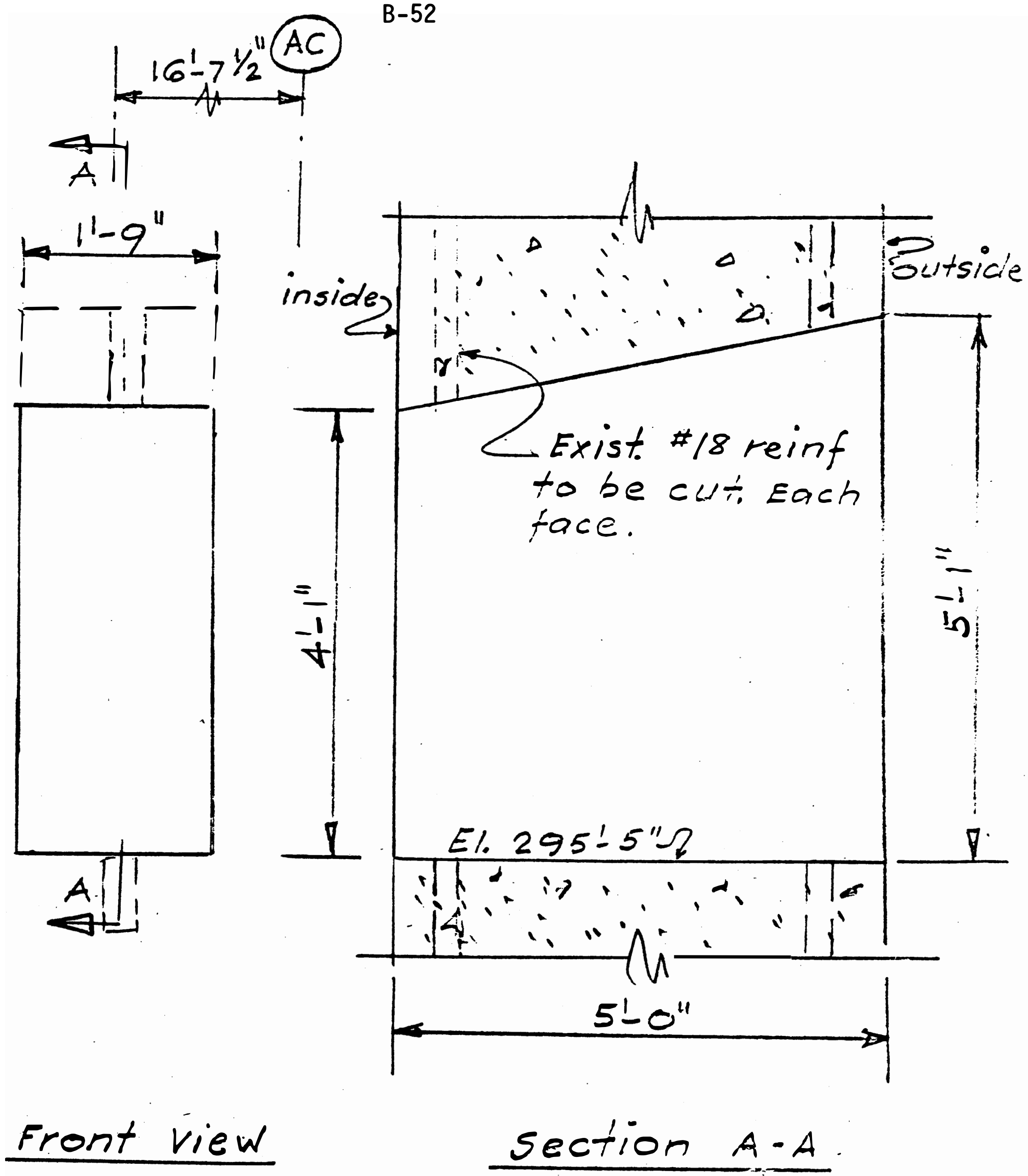

FIGURE 3
Sketch Showing Penetration through Auxiliary Bldg Wall.
4/26/1979.

Redundant measurements and display are provided for each parameter and separated to be consistent with the A or B pump train. Independent power supplies are provided for each train of instrumentation. No automatic control or interlock features are provided.

To the extent practical, the nuclear instrumentation which is safety related has been purchased to Class 1E requirements. The instrumentation selected was based on, to the extent available, the same manufacturer, model, and principle as the instruments used in the existing DHR system.

The staff concludes that the instrumentation and controls to be provided for the skid mounted decay heat system are appropriate for their intended function.

5. Electrical Power System Modifications

A. General Description

The licensee has identified a number of modifications to the Unit 2 power system in order to accommodate a loss of offsite power. Backup power to the two existing safety related load groups (designated Red and Green) is supplied by the two existing Class IE diesel generators. No loading changes have been proposed for these busses and the staff position is that all new loads be powered from other busses leaving the original Class IE system intact. The staff has further stipulated that the loading of these Class IE busses be on a "manual only" basis due to the time available for operator action following any perterbation and the complex nature of trying to coordinate which loads are needed and which would be detrimental if actuated for each potential plant mode of cooling. Approved written emergency procedures are required to cover the above contingencies. We find the above described modifications to the existing Class IE system to be acceptable.

For all electrical loads that previously did not require loss-of-offsite power back-up protection and all newly added loads which now require this protection, the licensee is installing two new diesel generators. These diesel generators will be assigned to busses 2-3 (breaker position 3-3) and 2-4 (breaker position 4-13).

These load groups are now designated "Gray" and "White" respectively. These diesels are self-contained skid-mounted outdoor units rated nominally at 2500 kw each. The convention for bus assignment has been odd numbers for the A loop and even numbers for the B loop and this has been retained for the modifications. We have required that the new load groups be maintained independent and that no interlocks shall be incorporated into the power system design that couples the load groups in any manner.

Each new diesel will have its own switchgear, control battery, and fuel tank installed adjacent to it. The switchgear will be connected (via underground cabling) to busses 2-3 and 2-4 (gray and white respectively) through the existing circuit breakers (previously ysed for condensate booster pumps Co-p-2A and Co-p-2B) with modified relaying to accommodate the power sources. Loss of offsite power will be detected by new undervoltage schemes which will automatically start the diesels. The breakers at busses 2-3 and 2-4 will be normally closed and the new diesel generator circuit breakers will be automatically closed when the units have attained rated speed and voltage. Motor loads are automatically tripped upon loss of power as part of the existing design. The staff requires that these busses be manually loaded by approved written procedure. These actions can be performed from the control room. The basis for this requirement is the time available for operator action and the different loads required for the various potential plant conditions. The licensee's design is in conformance with this requirement and is acceptable.

Busses 2-3 and 2-4 have existing bus transfer schemes that allow continuity of power supply, should one of the two offsite power circuits through the auxiliary transformers be lost, by fast-transfer to the other transformer. This scheme is to be left intact and the new undervoltage detection schemes will have a 10 second delay to accommodate the fast transfer if one circuit remains available. We find the above modifications to be acceptable.

We will require that testing requirement of the Gray and White diesel generators and their associated D.C. control power supplies (batteries) be comparable to those that now apply to the Class IE diesel generators and batteries.

B. Steam Generator A and B Short Term Solid Water Operation

The 900 horsepower condensate pumps are the single largest loads that will receive diesel generator back-up power source protection. The circulating water pumps rated at 2250 horsepower each have placed an additional restraint on the power system. This size load is too large for any of the four diesel generators now on site (red, green, gray and white). The licensee has proposed to provide a 13.2 kv line from the Middletown Substation to accommodate the motor starting requirements of these large loads. This line will have the capacity to start a second circulating water pump while supplying power to the first. Cooling system requirements are fulfilled by one pump operation.

The new 13.2 kv offsite transmission line will originate at the Middletown Junction Substation. This substation has a 230 kv and a 115 kv bus tied together by four auto-transformers. This modification utilizes auto-transformer no. 2 and connects to the 13.2 kv tertiary winding rated at 25 MVA. The group operated air break switches on the 230 kv side will be opened to isolate the transfer from a direct connection to the 230 kv system. There is a spare 230 kv line that terminates on the tower at the entrance to the Middletown switchyard on one end and terminates on the tower at the entrance to the Three Mile Island Switchyard on the other end. This line shares a double-circuit line of transmission towers with an existing 230 kv line to the plant.

At the Middletown Junction Substation, a short line section from the existing capacitor breaker on the 13.2 kv tertiary of the no. 2 transformer back to the spare 230 kv conductors must be constructed. The capacitor bank will be disconnected. The necessary additional relaying will be provided for this capacitor breaker to protect this new feed.

At the Three Mile Island Substation, a 13.2 kv underground cable supply from the spare 230 kv conductors will be run around the southern side of the natural draft cooling tower. This underground portion of the cable run protects the line from any of the other incoming lines falling on it. Once around the cooling tower, the line goes overhead for one span and terminates on a 10 MVA 13.2/4.16 kv transformer. A spare 10 MVA transformer will also be in place with manual switching capability for energization should the first one fail. These transformers are located adjacent to the circulating water pump house and the electrical switchgear bay. The 4 kv portion of the line ties into breaker cubicle 5-4 on bus 2-5 located within the switchgear bay. This breaker was originally used to supply power to circulating water pump CW-P-1E. This breaker will have modified relaying to accommodate the new power source. We have required this alternate power source scheme to be "manual-only" by approved written procedure. The bus must be cleared of all connections following loss of offsite power prior to reenergization and subsequent loading of the circulating water pump. There is a normally open bus tie between busses 2-5 and 2-6. Closing this bus tie gives access to five of the original six pumps.

We will require that the 13.2 kv line be tested weekly by energizing busses 2-5 and 2-7 for a short time interval to assure continued functional capability. There are no phase angle differences between the existing system and this new line so that the connection may be made on a live bus followed by tripping circuit breaker 28-52 to prevent tieing the 115 kv system to the 230 kv system through the plant distribution system. We further require that circuit breakers T-56-2 and T-78-2 be verified open on a daily basis.

1) the application is limited to the circulating water pumps.

2) these pumps are only required for the time window necessary to complete the modifications to steam generators A and B for long term core cooling.

3) the time required to have the 13.2 kv line operational is consistent with the requirement to provide back-up power as soon as practicable.

4) alternative proposals such as the use of diesel generators were not considered feasible due to the size requirements for starting 2250 horsepower loads and indoor service would require a new building that could not be built in time based upon existing schedules.

5) the new line has been isolated to the extent practicable from the 230 kv system and therefore may survive a local 230 kv system disturbance.

6) given a total grid blackout, there are six combustion turbines in the close proximity of Three Mile Island. These units are rated 23 MVA each and have black-start remote supervisory control. The system dispatcher in York has supervisory control of this system with the one exception that the system dispatcher in Lebanon would have to be consulted as to the position of certain intervening circuit breakers.

7) transmission line tower failure could not redner the 13.2 kv and the 230 kv systems inoperable.

We find the addition of the 13.2 kv transmission line as a backup power source to be acceptable given the operational needs and time restraints present at Three Mile Island Unit 2.

C. Steam Generator A Modifications

These modifications provide a new 700 horsepower high pressure pump for circulating water through the secondary side of the A steam generator. This new loop is in cooled by the Nuclear Services River Water System and the Nuclear Services River Water Purges. The new high pressure pump will be powered from bus 2-3 (Gray), the Nuclear Services River Water Pumps are powered from the existing Class IE power system. We find the power source allocations to be consistent with the separation of the A and B steam generator modifications, to capable of supplying the practical requirements of the system and to therefore be acceptable.

D. Steam Generator B Modifications

These modifications provide a new 700 horsepower high pressure pump for circulating cooling water through the secondary side of the B steam generator. This new loop is in turn cooled by the Secondary Services Closed Cooling Water Systems which utilizes the Secondary Closed Cooling Water Pumps. The final cooling loop uses the Nuclear Services River Water Pumps for circulation. The new high pressure pump will be powered from bus 2-4 (white), the secondary service closed cooling water pumps SC-P-1B and SC-P-1C are powered from 480 motor control center (MCC) 2-41B which in turn connects to 480 volt bus 2-41 and through a 4160/480 volt to bus 2-4. This arrangement allows a back-up pump (one pump operation meets cooling system requirements) and is part of the white power system. The Nuclear Services River Water pumps are existing loads on the Class IE diesels. We find the above power assignments to be consistent with the separation requirements of keeping the B steam generator cooling system on the even numbered separate busses, to be capable of supplying the functional requirements of the system and to therefore be acceptable.

E. Skid-Mounted Decay Heat Removal System

This new system will have 480 volt motor operated valves arranged in such a manner that there will be two sets of isolation valves on each of the three DHR lines that will be tapped. These valves will be assigned power sources (Gray and White) in a manner that assures isolation capability given a single power source failure. The remaining 480 volt motor operated valves will be arranged on a "per loop" basis to allow selection of either of the two new 4160 volt 400 horsepower pumps. These valves and the associated pumps will be powered from separate busses (Gray and White) to assure system function given a single power source failure.

Two new 480 volt motor control centers will also be provided. All electric powered equipment (valves, pumps, motor control centers and cabling) will be Class IE system quality. Because of the low decay heat levels, sufficient time is available for manual operator action assuming loss of power or equipment.

The secondary cooling loop for the DHR heat exchanger includes an additional 250 horsepower pump connected to 480 volt bus 2-44. All associated motor operated valves will receive normal power from the white power system. Should the white power system fail, backup power can be provided from the gray system by closing the normally open bus tie breaker between buses 2-44 and 2-34. We require this tie breaker be racked-out at all times and only closed upon the failure of the white power system by approved written procedure.

This system will not be used concurrently with the steam generator cooling modes describes above and therefore does not affect diesel generator capacity. We find the electrical power aspects of this design as described above to be acceptable.

F. Reactor Coolant Pressure Control System

All electrical equipment and instrumentation required to operate the system are powered from the gray and white power systems. The charging pumps (A and B) are rated 100 horsepower and are powered from 480 volt motor control centers 2-32A (gray) and 2-42A (white) respectively. The charging water storage tank heater is rated at 100 kw and will be powered from bus 2-45. There are a number of small loads associated with this system that have not been assigned power sources.

The system will be automated as soon as possible but this may not occur before initial operation. A motor operated isolation valve will be provided to automatically close when the water level in the tank nearest the reactor coolant system is approximately one third full. This is to preclude the introduction of nitrogen into the reactor coolant system.

6. Quality Assurance Program for TMI 2 System Modifications

The NRC Technical Review Group has reviewed and evaluated the Quality Assurance (QA) Program of GPU/MetEd and of their major subcontractors for the TMI-2 system modifications. These QA Programs recognize the uniqueness of the TMI-2 plant condition and balance the schedule urgency for completion of system modifications against the extent of application of traditional QA program practices consistent with maintaining assurance that specified system requirements are met.

The GPU/MetEd QA Program has been specifically tailored for the TMI-2 system modifications. The Program will apply QA criteria of 10 CFR Appendix B commensurate with the specified system requirements and will be compatible with the MetEd Operations QA Program previously accepted by NRC. GPU/MetEd has established a QA organization at the TMI-2 site specifically responsible for the system modification QA activities. This staff is experienced in all QA disciplines and associated technical fields, including mechanical, electrical and civil engineering as w ll as welding and non-desctructive examination. The GPU/MetEd QA Manager and QA engineers were brought in from the Forked River facility. GPU/MetEd is the lead responsible QA organization for the TMI-2 system modification program. Their QA Program will provide surveillance over the activities of their subcontractors, including Westinghouse and Burns & Roe. Westinghouse has established a QA Program to control QA activities associated with design, procurement and vendor component fabrication related to the skid mounted Backup DHR system modification. Westinghouse has organized an experienced QA staff specifically responsible for this task. Westinghouse will also provide system installation and pre-operational testing procedures and on site technical supervision which will be conducted subject to the GPU/MetEd QA Program.

Burns & Roe is responsible for establishing designs for the other TMI-2 system modifications, including the designs for the Reactor Coolant Pressure Control System, the A and B Steam Generator Cooling System.

Burns and Roe is implementing design control QA practices which assure that appropriate quality standards are specified and included in design documents that provide for verifying or checking the adequacy of design and that will control design changes. The GPU/MetEd QA Program will provide QA surveillance for the follow-on activities for these systems, including procurement, fabrication, installation and testing.

The NRC Regional Office of Inspection and Enforcement has available qualified QA staff experienced in mechanical, electrical and civil engineering disciplines and welding and non-destructive examination to provide surveillance of system modification activities at TMI 2 site and at equipment vendor facilities as necessary.

Based on our review and evaluation of the QA practices, controls, and organization of GPU/MetEd and their major subcontractors, we conclude that these QA Programs will assure meeting the criteria of 10 CFR 50 Appendix B commensurate with the TMI-2 system modification requirements and are acceptable.

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On the basis of current understanding of the accident scenario and available data, the staff reports here on its evaluation of the condition of the core and the core flow resistance as it might affect ability to cool the core by natural circulation. The natural circulation cooling capability of TMI-2 for the estimated core flow resistance and a variety of other conditions is evaluated and a comparison of the Base Case and off-nominal plant configurations is presented. The potential for and effects of natural convection core cooling are addressed, and the staff recommendations for reactor performance acceptance criteria upon initiation of natural convection are presented.

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