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ANALYSIS OF POSTULATED CORE MELTDOWN OF AN SRP REACTOR

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FINAL REPORT

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ANALYSIS OF POSTULATED CORE MELTDOWN OF AN SRP REACTOR

FINAL REPORT

by

William S. Durant
Robert J. Brown

October 1970

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INTRODUCTION

An analysis was made to determine the consequences of a postulated accident in which the core of a Savannah River Plant reactor melts down following the loss of coolant. The study was made to determine 1) the potential damage to the reactor building that could impair its integrity for confining activity and 2) the need for additional facilities to prevent the activity confinement system from being overheated by the decay heat in the debris.

A preliminary report on this analysis was issued previously.[1] The sequence of events during and following the loss of coolant has now been studied in more detail, and a computer program has been written and used to investigate transient heating effects. This is the final report of the analysis and presents the conclusions.

SUMMARY

Calculations show that in the event of the postulated complete meltdown of the core of an SRP reactor, the integrity of the confinement system would not be breached either by the penetration of a molten core through the floors and into the earth beneath or by rupture due to explosion (steam or hydrogen). The massive concrete structure housing the reactor facility affords an effective heat sink and is expected to be resistant to the effects of explosions of the magnitude probable for the accident. Heat would be removed by the ventilation system to the atmosphere.

The analysis also shows, however, that a heat removal system would be required to prevent the exhaust air from reaching temperatures in excess of 100°C. Above this temperature significant desorption of iodine from the carbon beds could occur. A system has been designed to cool the debris directly by means of a water spray external to the reactor vessel.

The longer-term effects calculated for a meltdown accident in which the debris is cooled by the proposed confinement heat removal system are as follows. The below-grade areas of C or K reactor building would fill with water in about 20 hours after meltdown; at this time the flow would be stopped. Heating from decay energy would increase the water temperature to about 95°C in about 30 days.

After 30 days, heat loss through the building structure to the earth would exceed the heat generation, and the pool would begin to cool. No addition of water to the pool would be required. Because of the larger below-grade volume in P reactor building, the maximum water temperature would be less than for C or K.

Forced air cooling of the carbon beds in the filtration-adsorption compartments would be required for about 24 hours after the meltdown, after which time natural draft would maintain the temperature of the beds below 100°C. After about one day, the remaining fission product inventory could be confined within the reactor building and the filtration-adsorption compartments for an extended period of time with no further equipment operation or maintenance.

DISCUSSION

Analyses were made of the consequences of two variations of a loss-of-coolant accident in SRP reactors (Figures 1-4):

1. failure to actuate the light water addition systems
2. addition of light water after core meltdown

Analyses of this type have many conceivable variations. The analyses in the present study are based on engineering judgment of conservative, but logical, sequences of conceivable events.

The first part of the discussion is concerned with heat transfer from the molten core and the sequence of events ending with debris at the -40 ft level if no light water were added. The second part of the discussion is concerned with the consequences of metal-water interaction that might result from the addition of water after the core melted, leading to either steam explosions or chemical reactions. The last part of the discussion deals with the events following a core meltdown in which cooling is provided by a proposed confinement heat removal system.[2]

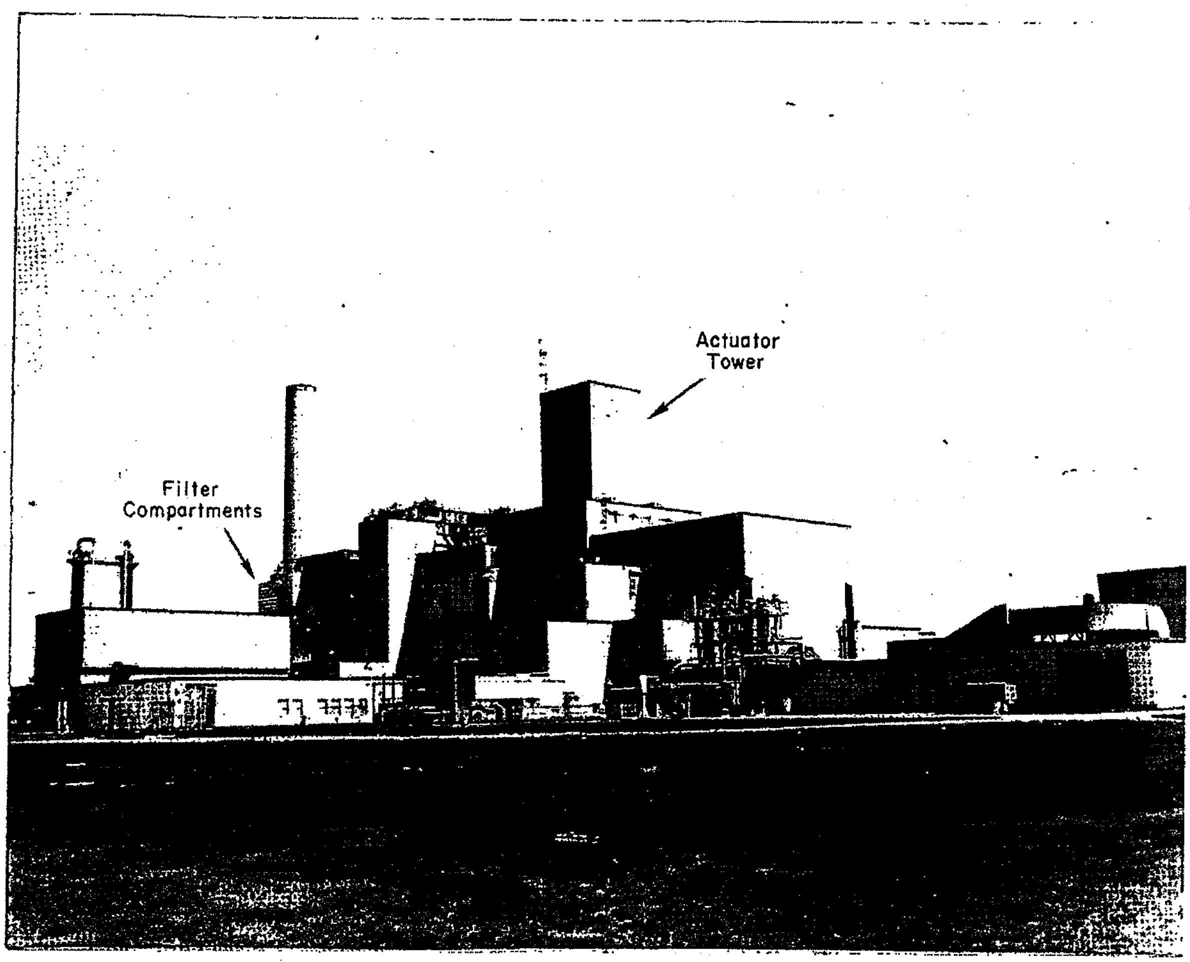

FIG. 1 SAVANNAH RIVER PLANT REACTOR BUILDING

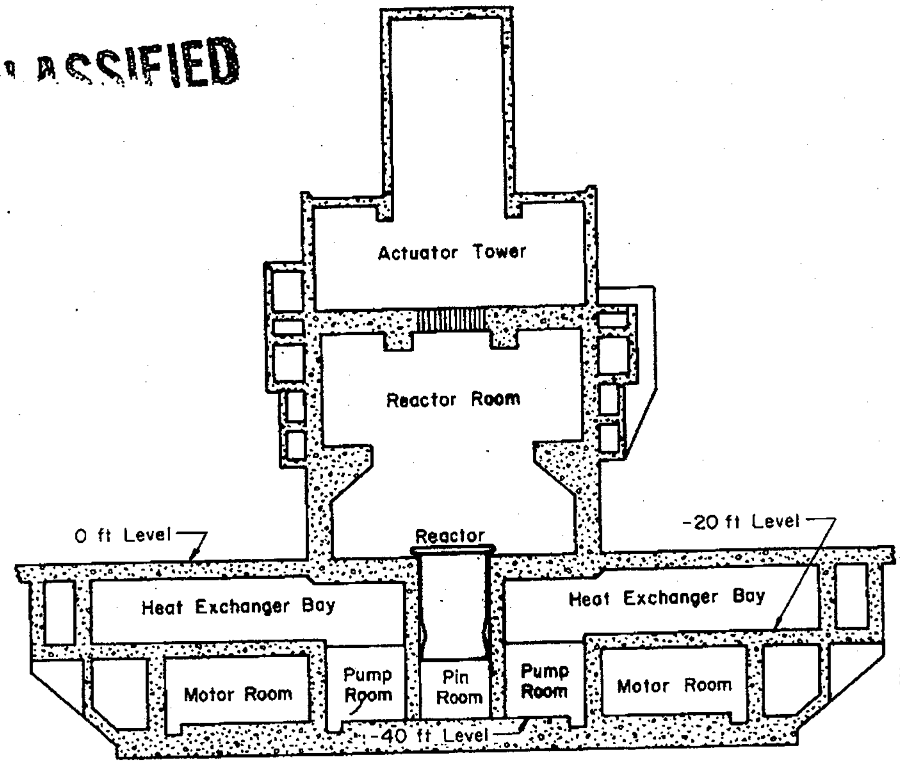

FIG. 2 SCHEMATIC CROSS SECTION OF REACTOR BUILDING

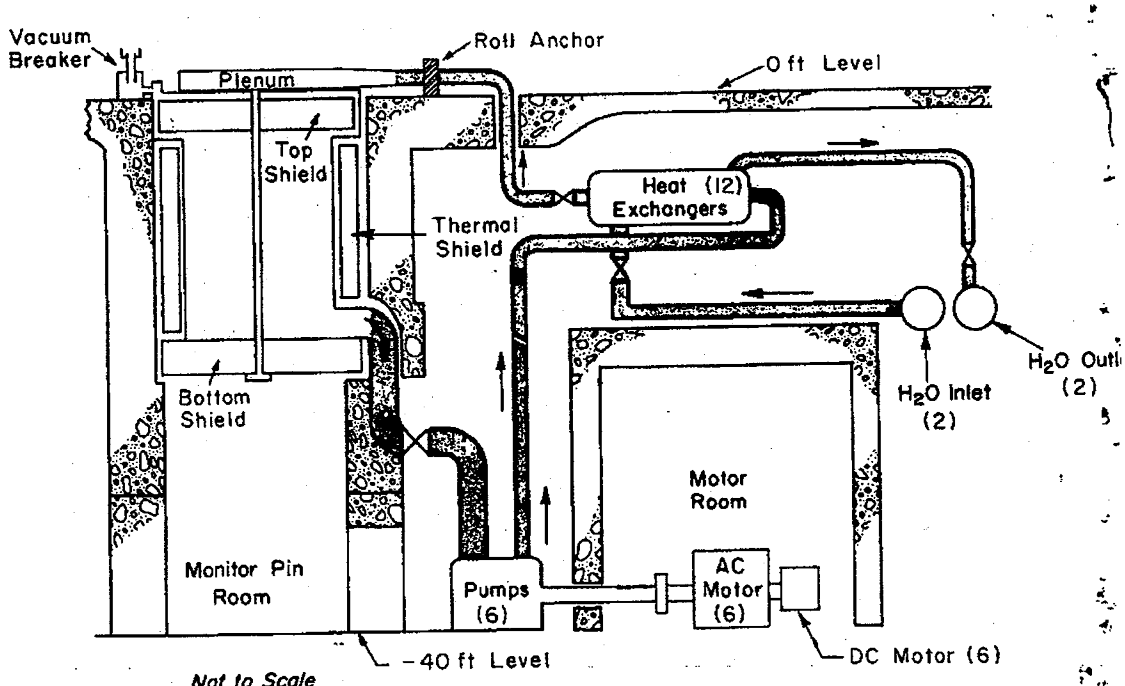

FIG. 3 SCHEMATIC OF REACTOR COOLING SYSTEM

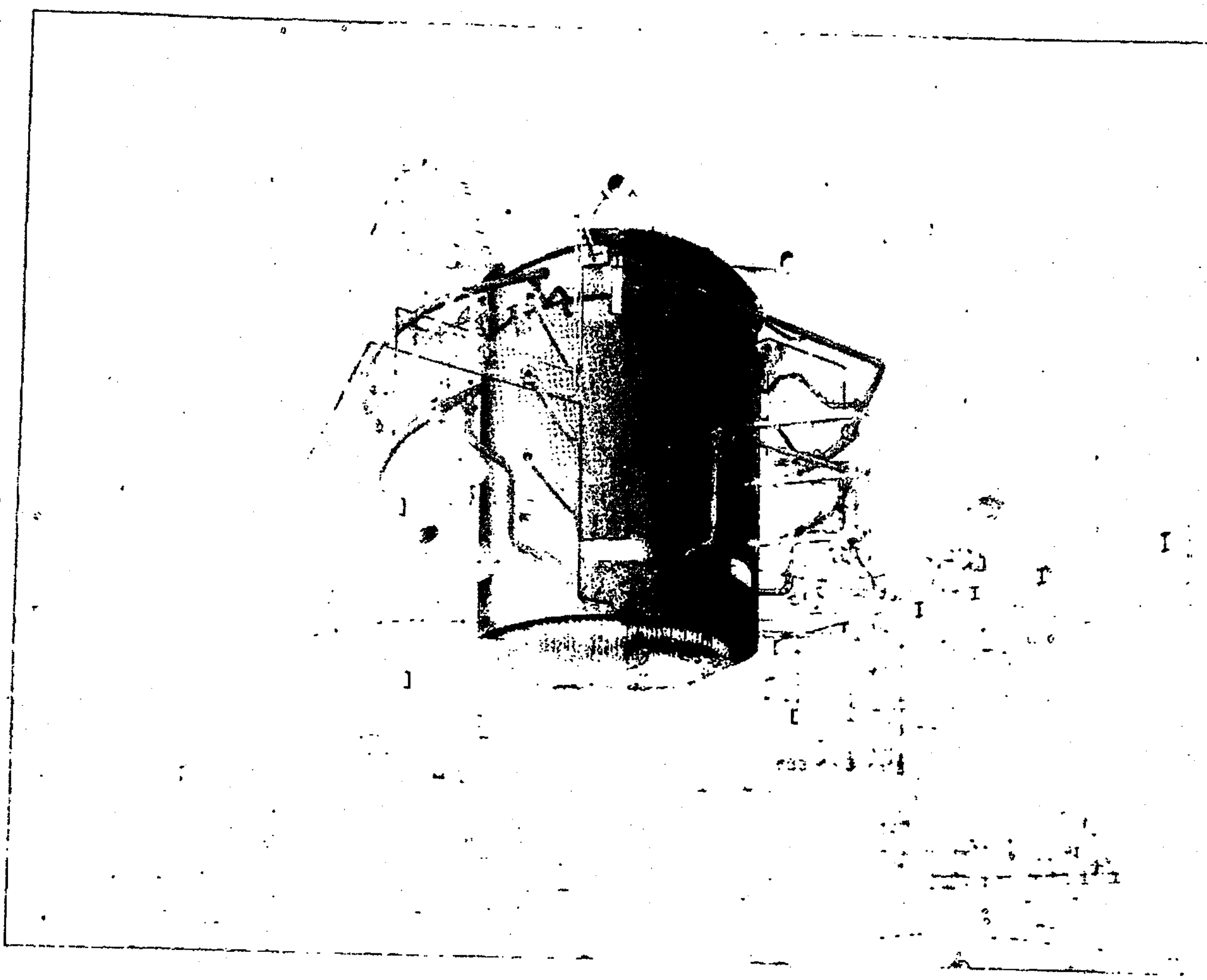

FIG. 4 SAVANNAH RIVER REACTOR

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FAILURE TO ACTUATE THE LIGHT-WATER ADDITION SYSTEMS

Description of the Accident

The following sequence of events was used as a basis for calculating heat transfer to the exhaust ventilation air following a loss-of-coolant accident (Figure 5) in which the light water addition systems were not actuated. The postulated accident, representing an extreme case, is a double-ended break in one process water plenum inlet line. Although no significant process line breaks have been experienced at SRP, if one were to occur, it would almost certainly be much smaller than a double-ended break and would cause much less severe consequences. The double-ended break is calculated to be the fastest means of removing D_2O from the reactor system; if a single pipe failed, $\sim$20 seconds would be required to drain all the D_2O from the reactor. The safety rod circuits would be actuated by any of several signals: low plenum pressure, low fuel flow, low blanket gas pressure, low septifoil header pressure, assembly high coolant temperatures, or low D_2O level. Even if the safety circuits failed, the reactor would be shut down by the loss of moderator. Actuation of the emergency core cooling system would then provide continued core cooling and limit core damage.

If the emergency core cooling system were not actuated, nearly adiabatic heating of the assemblies by decay heat would begin upon complete loss of coolant. Less than one minute is required for all of the fuel to melt in a high flux charge, about 2 minutes in a full reactor uranium-aluminum alloy charge, and about 9 minutes in a uranium metal charge. Significant melting in a uranium-aluminum alloy charge, however, could begin in about one minute. During melting, heat transfer to the atmosphere or the reactor structure would be negligible. Molten fuel and aluminum would slump to the bottom of the tank. Heat then would be transferred primarily to the bottom shield (Figures 6 and 7) by conduction and some fuel would solidify on the tank bottom. Radiant and convective heat transfer within the reactor tank is calculated to be negligible.

In K or P reactor, most of the molten mass would flow into the outlet nozzles and into the pumps at the -40 ft level. (Levels within the reactor building are referenced to the floor of the reactor room at the top of the reactor which is designated as the 0 ft level. The -40 ft level is the lowest process level within the building.) In C reactor, the outlet nozzles are 15 inches above the bottom of the reactor tank (Figure 7); thus the molten mass would be retained in the tank until the temperature became sufficiently high to melt through the tank bottom and bottom shield.

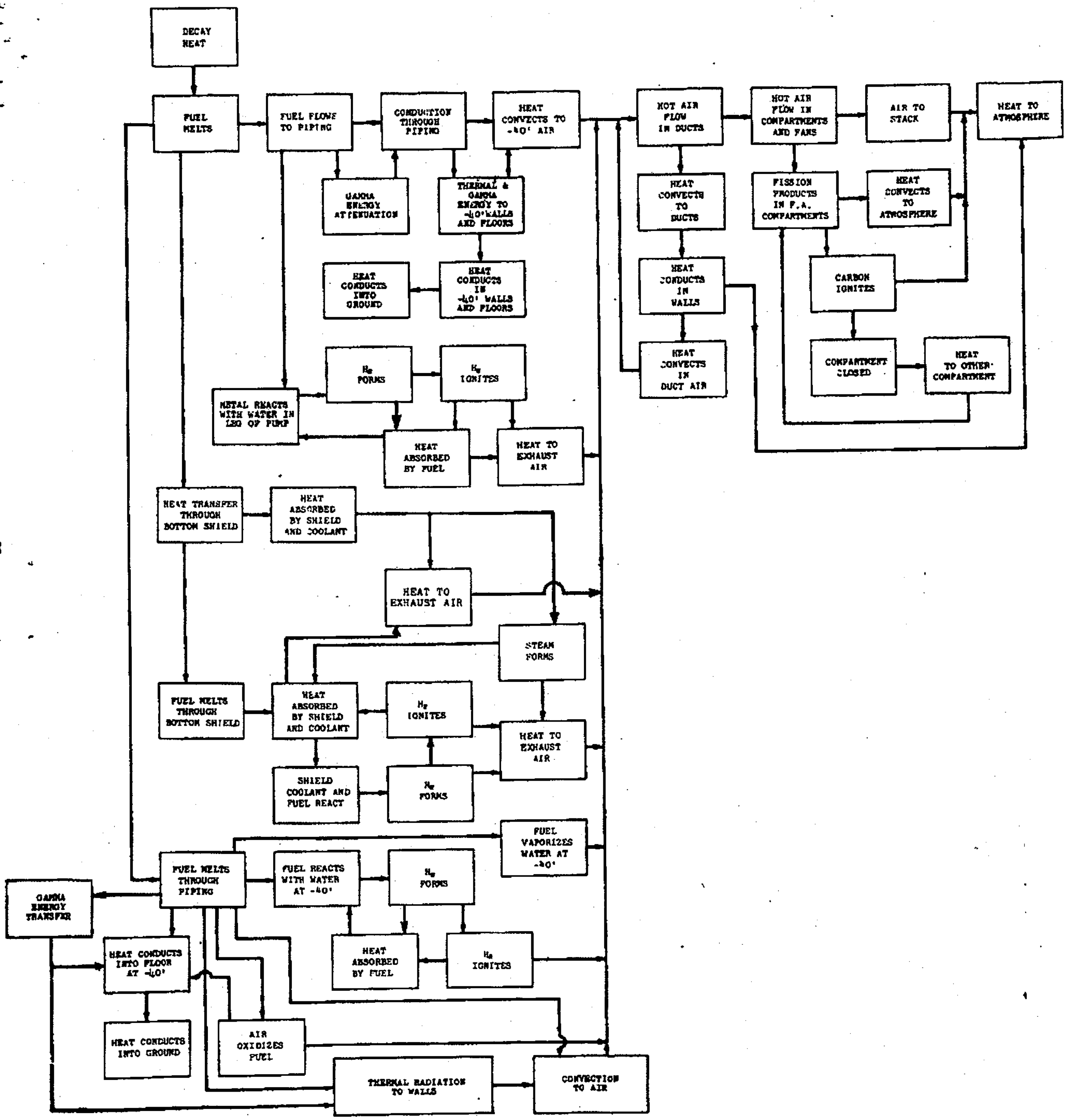

FIG. 5 ACCIDENT FLOWSHEET-NO ADDITION OF LIGHT WATER TO FUEL

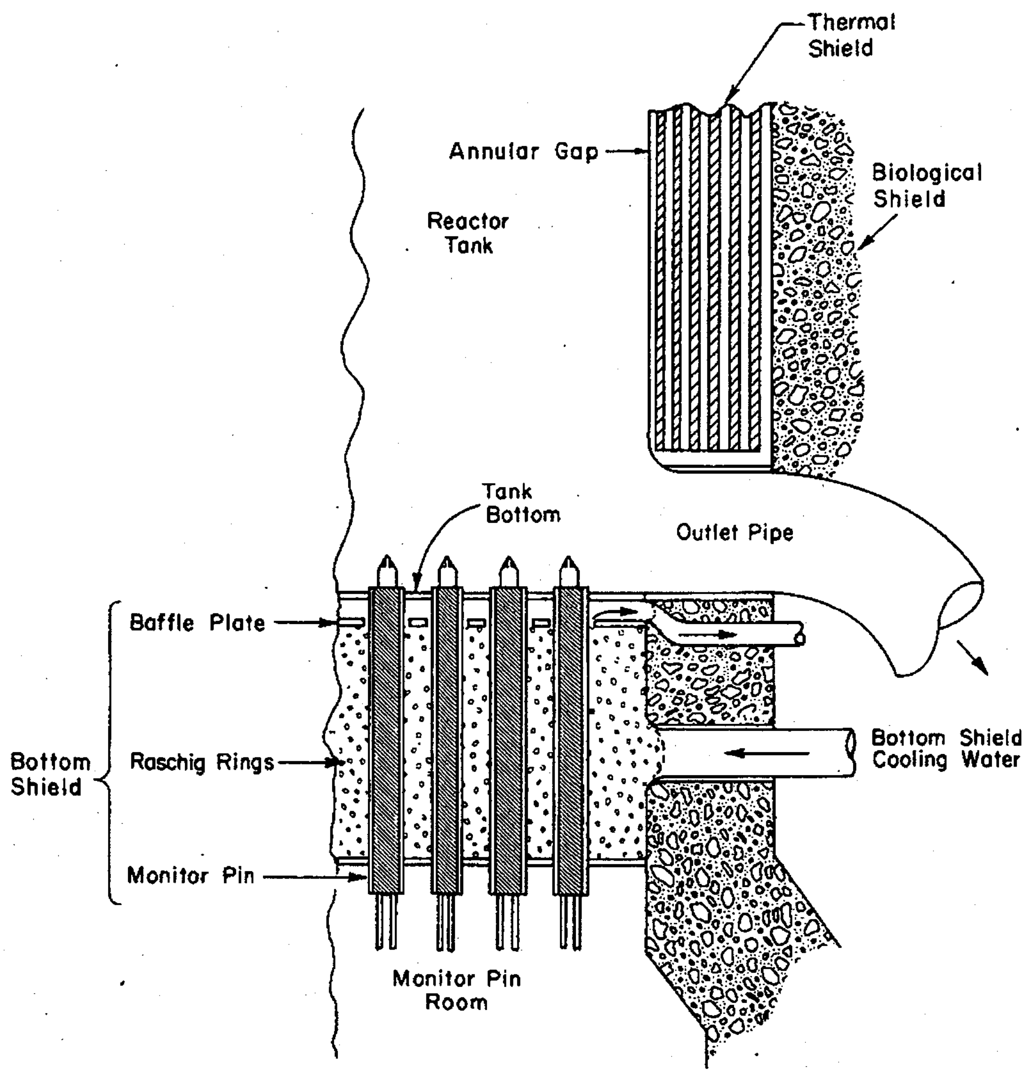

FIG. 6 SCHEMATIC OF BOTTOM SHIELD OF P AND K REACTORS

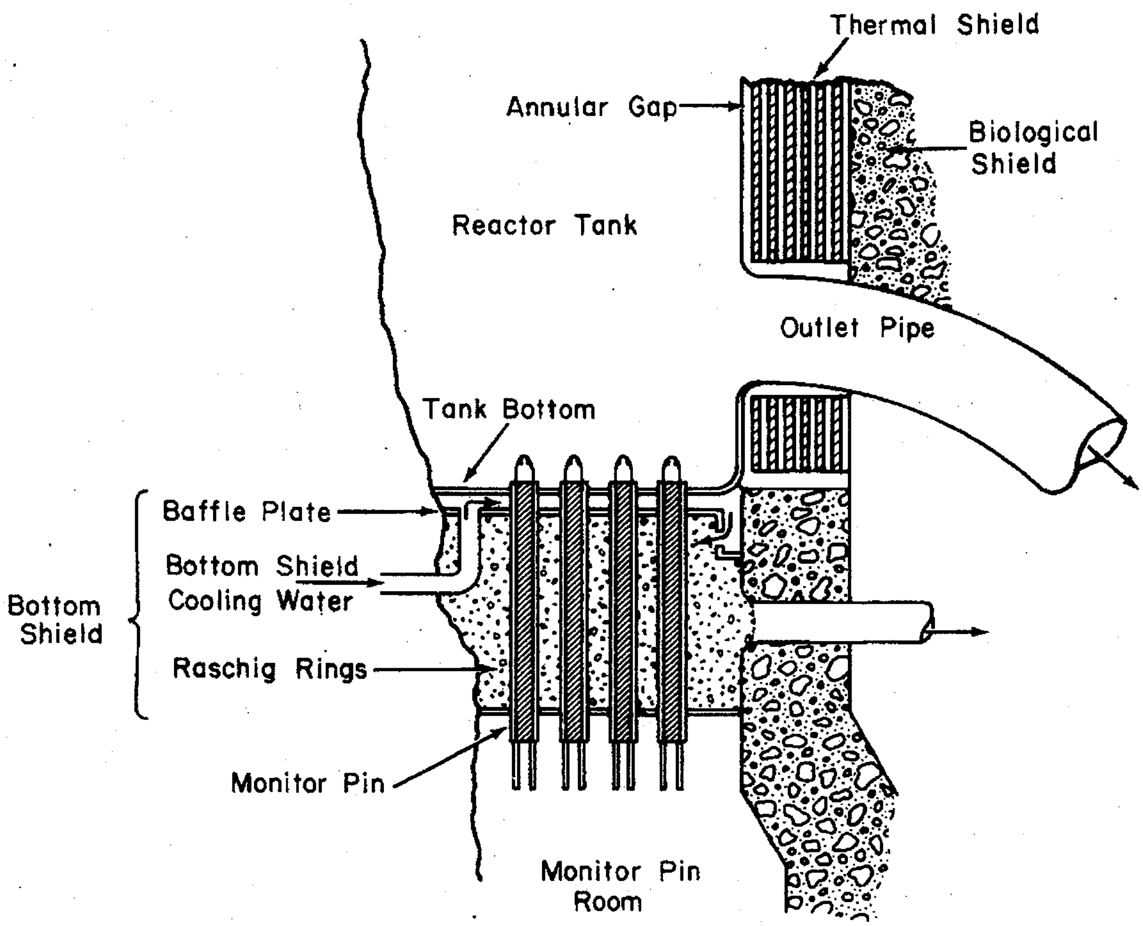

FIG. 7 SCHEMATIC OF BOTTOM SHIELD OF C REACTOR

Heat Transfer at the -40 ft Level

Mechanisms of heat transfer from the molten mass in the pumps and lower process piping for P and K reactors are shown schematically in Figure 8. As the molten fuel flows into the cold piping, sufficient heat would be conducted into the piping to resolidify the fuel in immediate contact with the stainless steel. Consequently, the fuel would not plunge en masse to the lower levels of the process piping system; dissolution of the stainless steel would be negligible as long as the fuel in contact with the piping surface remained solid. Heat conducted through the piping and pumps would be transmitted by thermal radiation to building surfaces and by convection to the air. Heat would also be transmitted from the building surfaces by conduction into the concrete and steel and by convection to the ventilation air. The energy transmission by gamma radiation is included in calculations of heat transfer.

Because heat would not be dissipated from the piping as rapidly as it would be generated, the temperature of the fuel would again increase. When the fuel in contact with the piping remelts, the uranium would dissolve the piping, and the molten "lava" would pour to the floor of the pump rooms. "Lava" refers to the combination of fuel, cladding, other core material, and stainless steel. Uranium forms a eutectic with stainless steel (at about 65 wt % uranium) which melts at about 725°C (Figure 9). The average melting temperature of the lava from a uranium metal charge would be about 1000°C because of the relative amounts of uranium and stainless steel. The lava would spread over the floor until the heat loss into the floor, walls, and air shown in Figure 10 is equal to the decay heat at the melting temperature ($\sim$1000°C). Significant oxidation of the uranium would not be expected until the debris is deposited on the floor at the -40 ft level because of the relatively small amount of oxygen in the reactor vessel. Oxidation at the -40 ft level is discussed later.

Significant heat transfer to the -40 ft areas would not occur in C reactor until the lava melted through the bottom shield. After this occurred, the mechanisms for heat transfer in all three reactors would be the same.

Surface disruption of the concrete floor and walls would be expected, but gross structural failure would not be likely because the lava would solidify rapidly, thus precluding further motion of the lava and concrete particles on the floor. Also, the massive walls and floor can dissipate a large amount of heat.

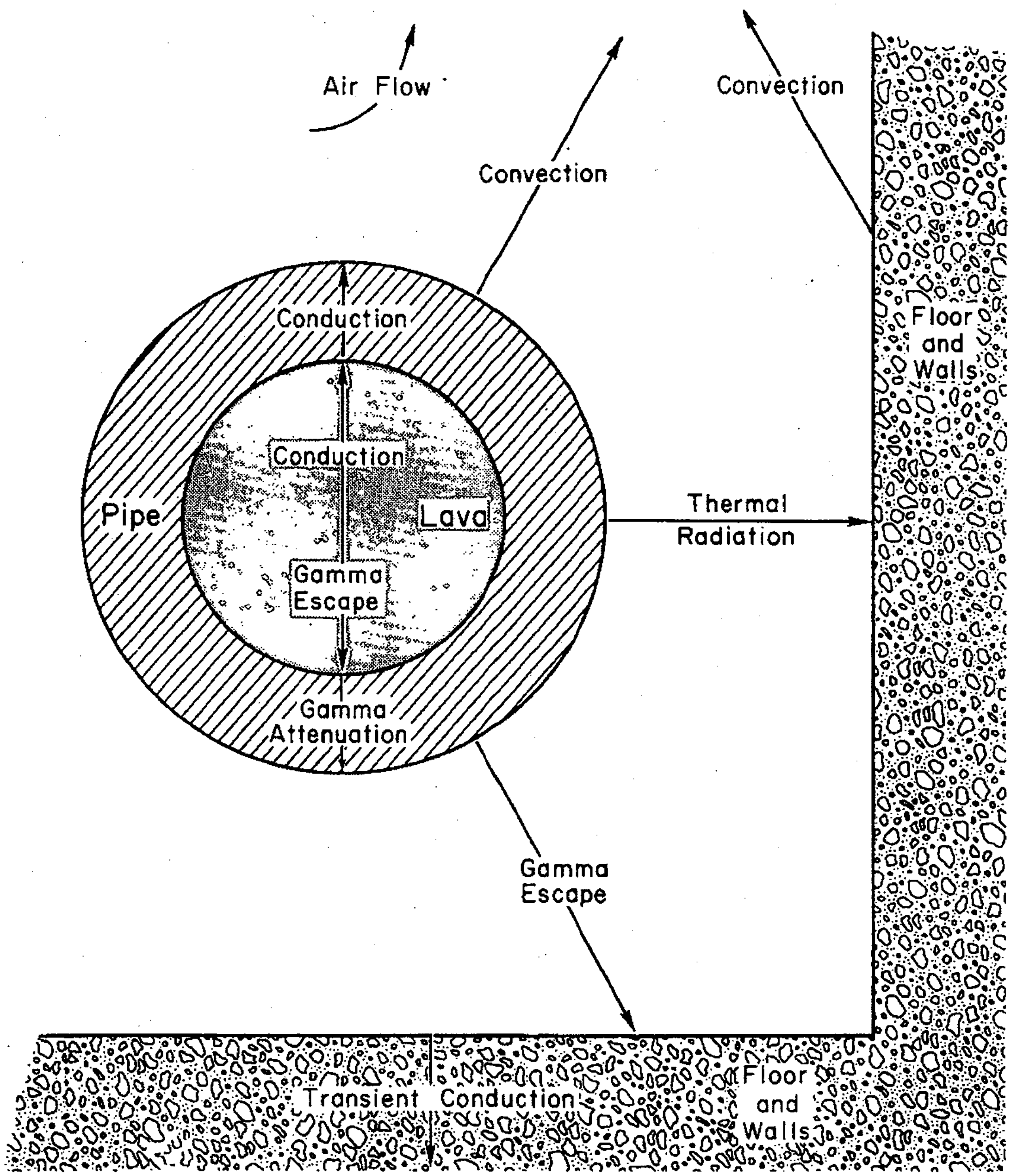

FIG. 8 LAVA CODE MODEL BEFORE PIPE COLLAPSE

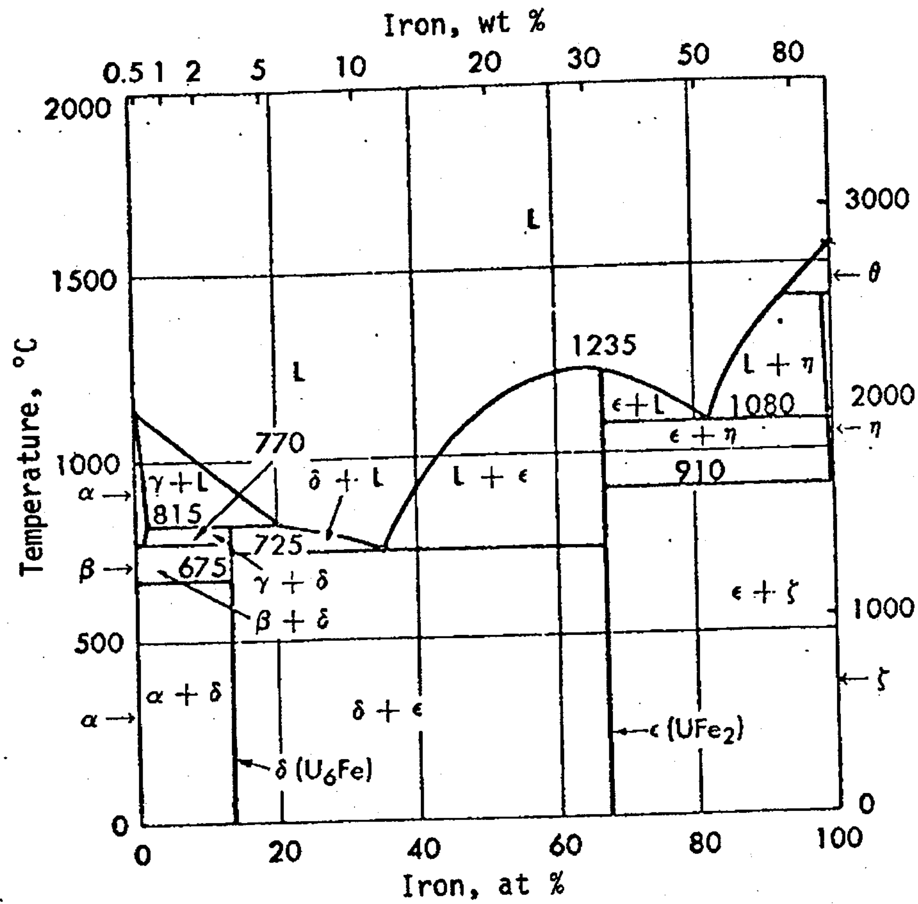

FIG. 9 PHASE DIAGRAM FOR URANIUM-IRON[36]

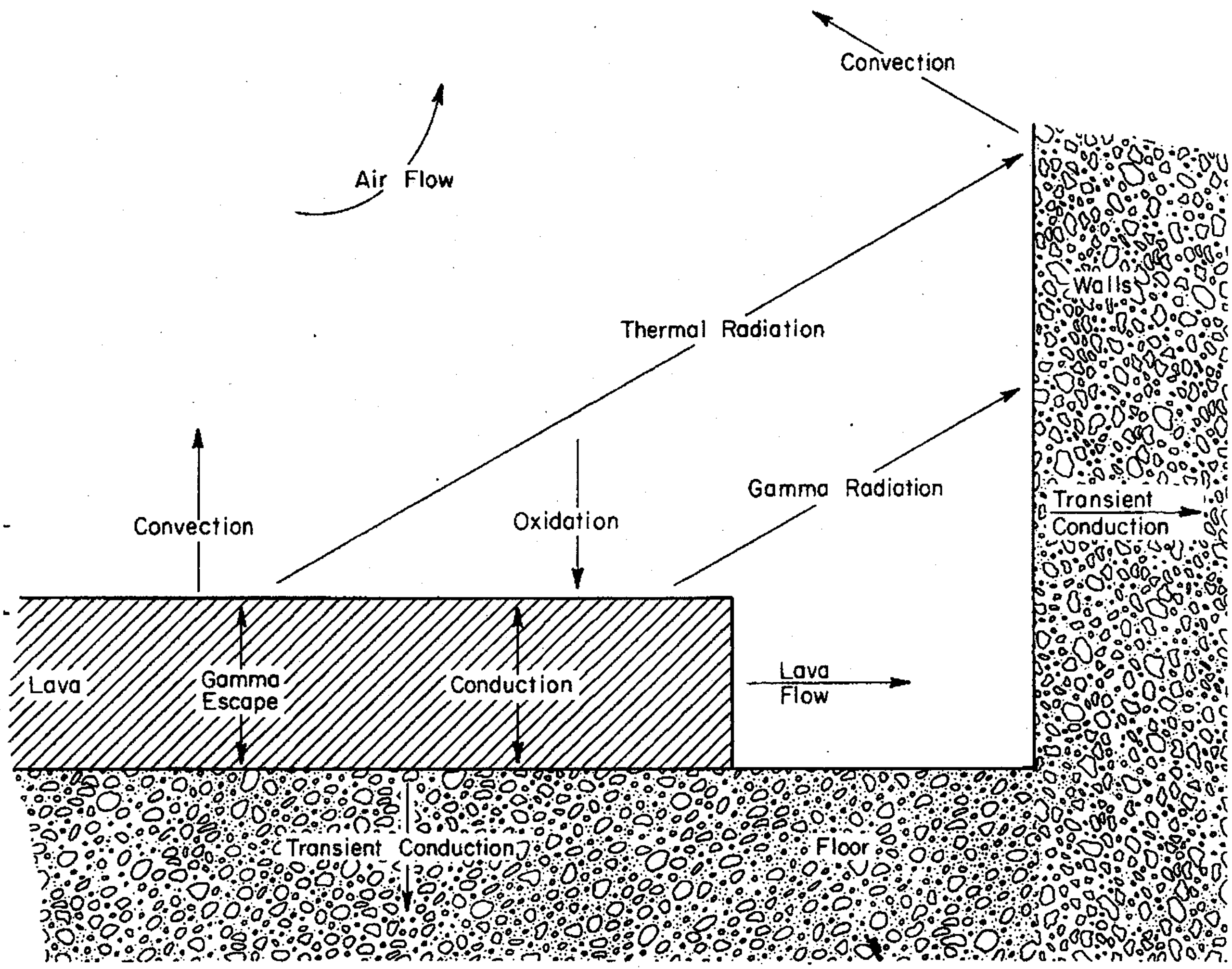

FIG. 10 LAVA CODE MODEL AFTER PIPE COLLAPSE

The possibility of an explosion between cold concrete and molten metal was considered. Although severe splattering of small amounts of hot metal on cold concrete can occur, experience indicates that large amounts of molten metal spilled on concrete will not result in an explosion. Three major steel producers[3] were questioned about spills of molten metal on concrete. Each had experienced incidents in which up to tens of tons of molten steel (nominal temperature: 1600°C) had been spilled onto concrete floors. On such occasions the concrete below the center of the spill was spalled and cracked to a depth of several inches. Missiles frequently resulted from spalling of the concrete adjacent to the periphery of the molten mass, but the objects did not have sufficient energy to cause damage to building or equipment strucures. Severe splattering of large amounts of metal or explosive gas expansion did not occur. The thermal conductivity of the concrete in both the monolithic and spalled condition is so low that only relatively small volumes of concrete are heated to temperatures at which release of the water of hydration is rapid.

Air oxidation of the lava would also heat the mass. The overall oxidation rate for uranium at 1200°C can be expressed as follows:[4]

$$\text{Oxidation Rate} = 30.0\, e^{(1.645\, A_o/W_o)}, \text{ expressed as mg/(cm}^2\text{)(min)}$$

where

A_o = original surface area, cm^2

W_o = original weight of uranium, mg

The effect of air oxidation on the thermal load to the exhaust air is included in the analysis. Unlike most power reactor containment systems, much of the heat from the mass would be removed by the continuous flow of ventilation air. This continuous heat removal reduces the destructive potential of the lava. During the time the ventilation air travels from the process areas to the stack, some heat is absorbed by the ducts or is conducted through the filter compartment walls to the outside atmosphere; however, most of the heat in the air is discharged from the stack. Less than 10% of the uranium in a uranium metal charge is calculated to be oxidized in the first 16 hours of the postulated meltdown.

Heat Transfer to the Bottom Reactor Shield

An analysis of the heat transfer to the bottom shield of the reactor was made to determine the amount of fuel debris that could remain in the tank following a core meltdown accident.

The bottom shields of the reactors (Figures 6 and 7) are 40 inches thick and are packed with stainless steel Raschig rings in the spaces between the monitor pin sleeves. Deionized light water in an independent cooling loop is provided to cool the shield. Decay heat from fuel remaining in the reactor tank cannot be transferred readily to the exhaust ventilation air; thus, the heat load to the confinement system would be inversely related to the quantity of fuel remaining in the tank.

In K or P reactor, about 93% of the debris from a natural uranium charge is calculated to flow into the reactor effluent piping within 5 minutes after a full core meltdown. The remainder of the fuel would be cooled by natural convection and radiation, and no gross melting of the tank bottom is calculated.

In C reactor, none of the molten fuel would flow into the reactor effluent piping because the nozzles are 15 inches above the tank bottom. The molten fuel would melt through the bottom shield in about 50 minutes if no metal-water reaction occurred. After the tank bottom melts, the shield would be vapor locked if a steam explosion does not occur.

Metal-water interaction in the bottom shield of K or P reactor is possible but not very probable. On the other hand, significant metal-water interaction could occur in the bottom shield of C reactor.

The air temperature inside the reactor vessel approaches the fuel temperature within 5 minutes after losing coolant flow because the annular air gap (Figures 6 and 7) adjacent to the reactor vessel wall provides a very high resistance to heat losses. For a uranium metal charge, the air temperatures inside K or P reactor would approach 1100°C; the air temperature inside C reactor would approach 1430°C before the bottom shield is completely penetrated. Lower temperatures would be expected for other types of charges because of less decay heat than associated with uranium metal charges.

Method of Calculation for Heat Removal from Molten Debris

A computer code, called LAVA, was written to compute the time-dependent heat transfer from the fuel debris after melting is completed and the debris has flowed into the effluent nozzles of the reactor tank. The code calculates:

- the location and physical state of the fuel,
- the temperature of the fuel,

- the temperature of the process equipment and nearby building structure, and

- the temperature of the exhaust ventilation air.

Preliminary calculations for a natural uranium charge were presented in Reference 1. Subsequently, refinements to the code have been made to account for thermal energy released from the debris as gamma radiation and for energy added to the debris from oxidation.

The LAVA computer code involves heat balance determinations between the energy released as decay heat and that dissipated from the debris. Two sets of calculations are made depending upon the location of the mass of debris. In the initial determination, molten core material at the melting temperature is assumed to flow into the effluent piping and pumps of the reactor. Because of the large, cold mass of the piping and pumps ($\sim$200,000 pounds), most of the debris will freeze. Interaction between the debris and the stainless steel piping is assumed to be negligible during the short time that the debris is solid, thus the piping remains intact. During this interval, the heat balance is made using the LAVA code model illustrated in Figure 8. Modes of energy transfer that are used in the calculations are listed in Table I. Although neglected in this analysis, a large quantity of energy can be removed by existing water sources at the -40 ft level as discused in a later section (Heat Removal from Potential Water Sources at the -40 ft Level).

After the debris remelts, the uranium will rapidly dissolve the piping, and the mass will fall to the floor. The heat transfer model for this condition is shown in Figure 10. Modes of energy transfer from the debris on the floor are listed in Table I.

Results of Calculations

The results of calculations for three types of reactor charges are presented in Figures 11 through 14.

TABLE I

Modes of Energy Transfer Used in LAVA Computer Program

Debris Contained in Piping (see Figure 8)

Conduction through a solid cylinder with internal heat generation (lava)

Gamma escape from a solid cylinder (lava)

Conduction through the wall of a hollow cylinder (S/S pipe)

Gamma attenuation by the wall of a hollow cylinder (S/S pipe)

Thermal radiation from a cylinder to completely enclosing surfaces (pipe to the building floor and walls)

Natural convection from vertical and horizontal surfaces (building floor and walls)

Gamma attenuation by a slab (debris on concrete walls and floor)

Transient conduction through a slab (debris on concrete walls and floor)

Absorption of heat of fusion (lava)

Debris on Floor of Pump Rooms (see Figure 10)

Conduction through a slab with internal heat generation (debris on floor)

Gamma escape from a slab (debris on floor)

Thermal radiation from a plane to partially enclosing surfaces (debris on floor to walls of building)

Natural convection from vertical and horizontal surfaces (building floor and walls)

Gamma attenuation by slabs (debris on floor)

Heat absorbed from oxidation reaction (debris on floor)

Transient conduction through slabs (concrete walls and floor)

Absorption of heat of fusion (debris on floor)

[redacted]

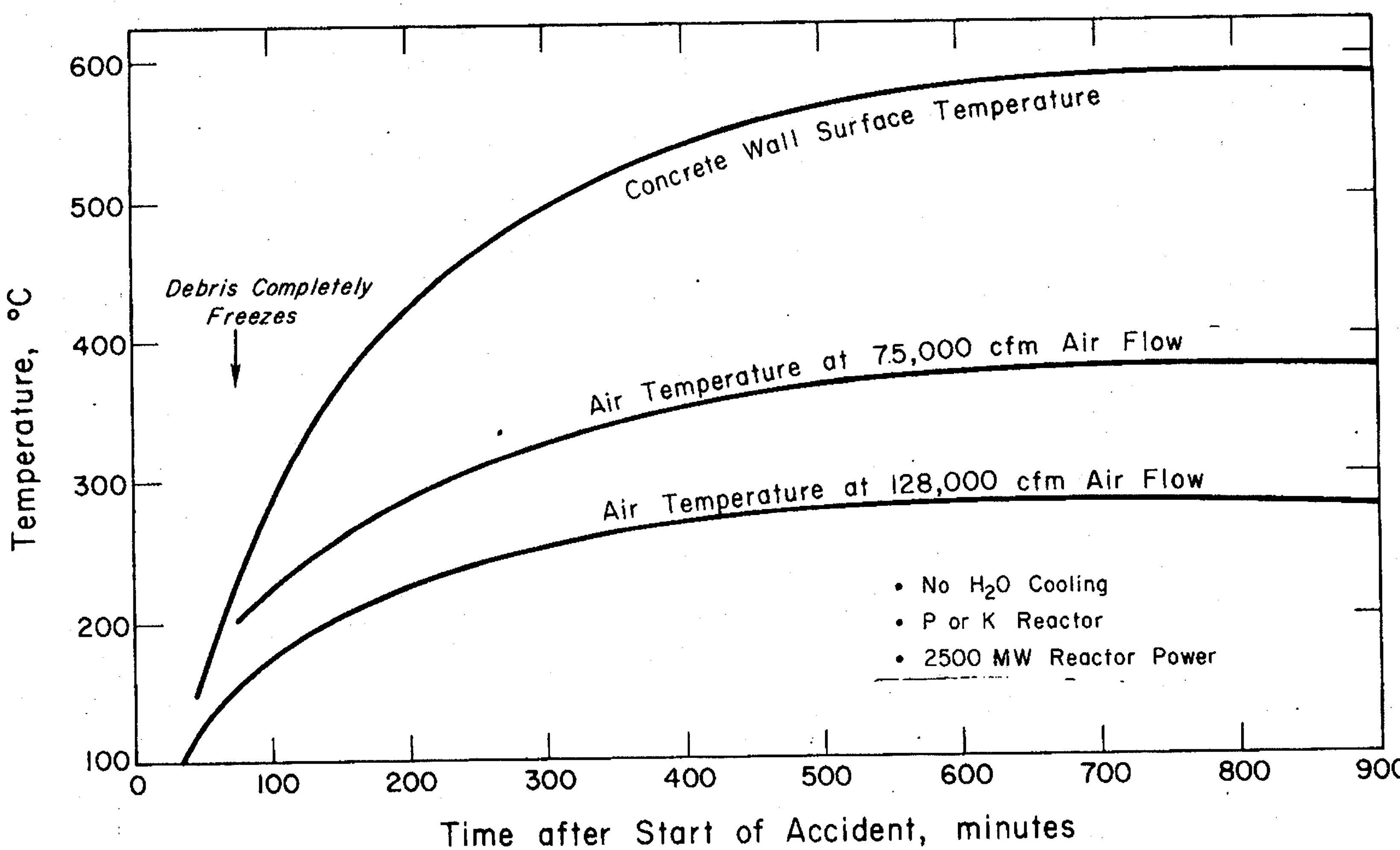

FIG. 11 CALCULATED TEMPERATURES FOLLOWING MELTDOWN OF FULL-REACTOR URANIUM METAL CHARGE

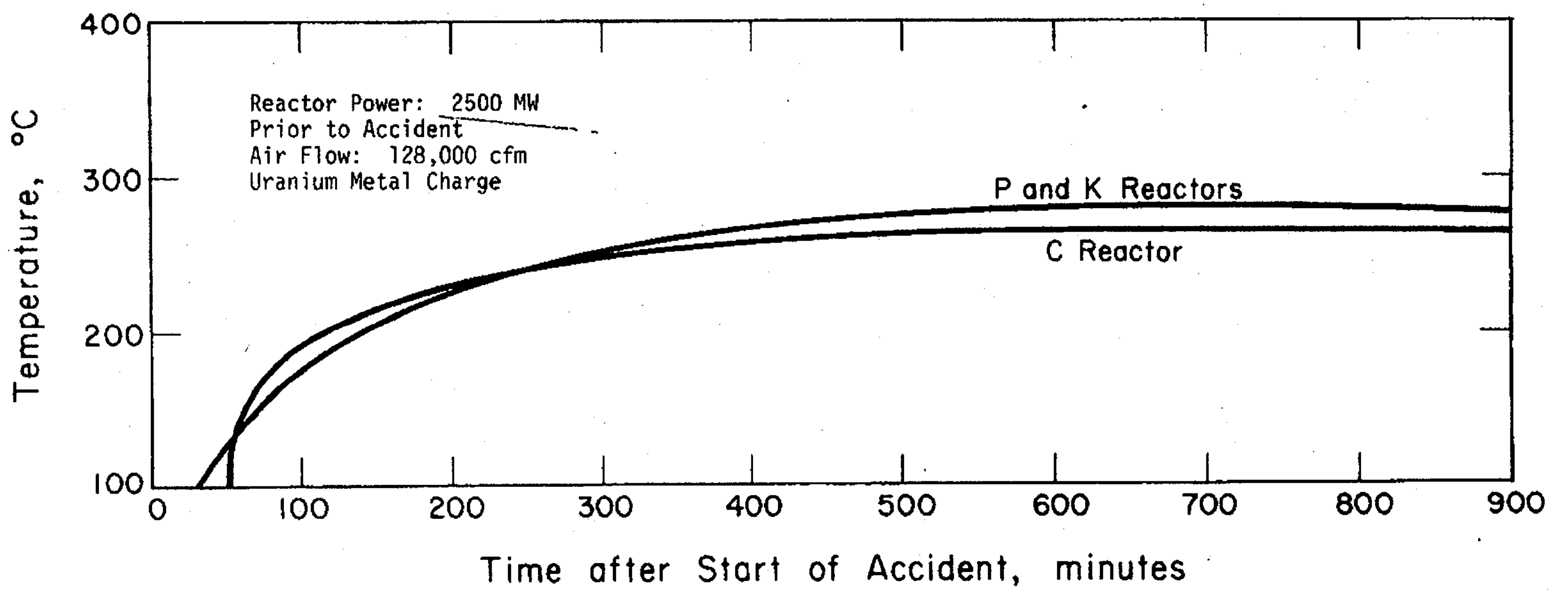

FIG. 12 COMPARISON OF CALCULATED EXHAUST AIR TEMPERATURES FOR C REACTOR AND P OR K REACTOR

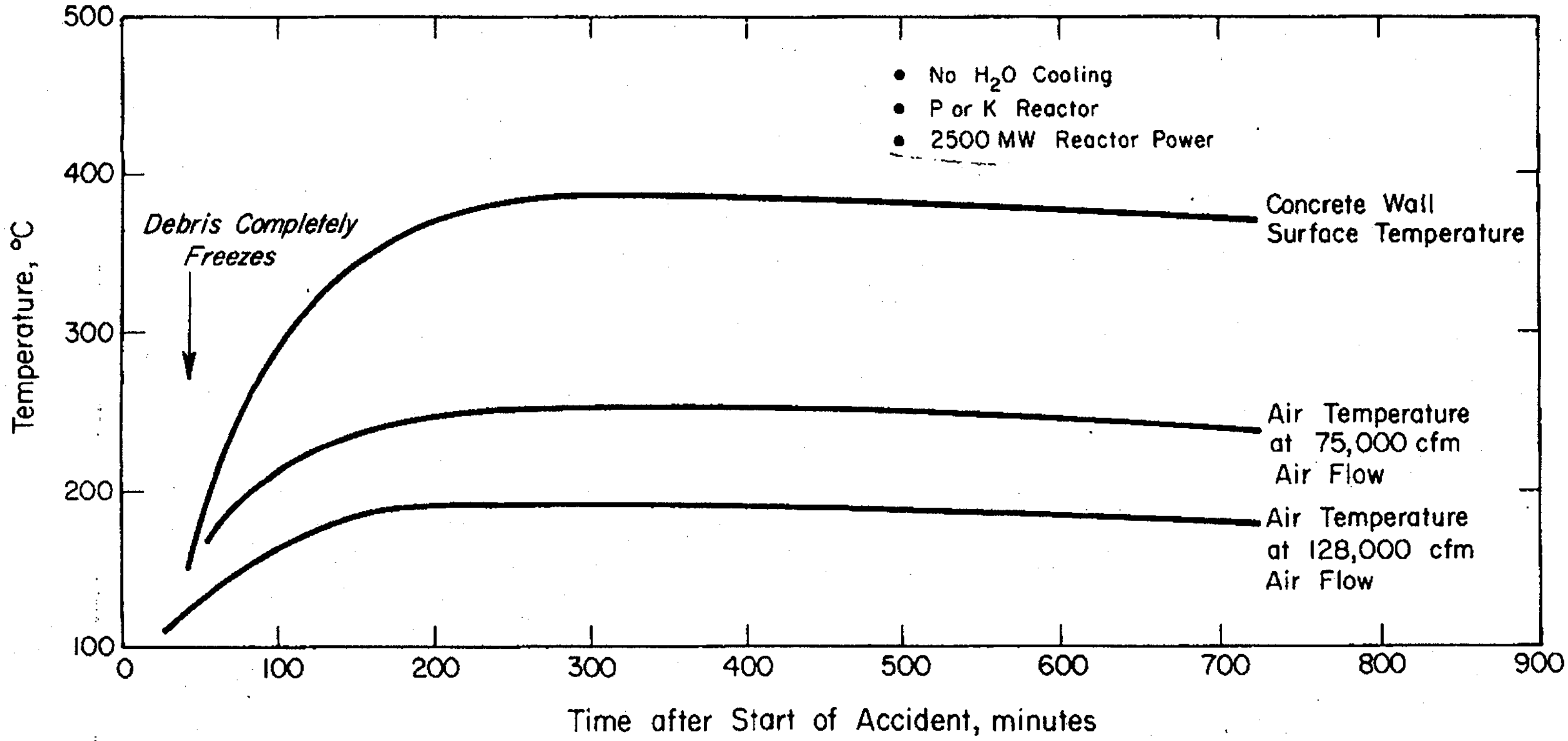

FIG. 13 CALCULATED TEMPERATURES FOLLOWING MELTDOWN OF FULL-REACTOR URANIUM-ALUMINUM ALLOY CHARGE

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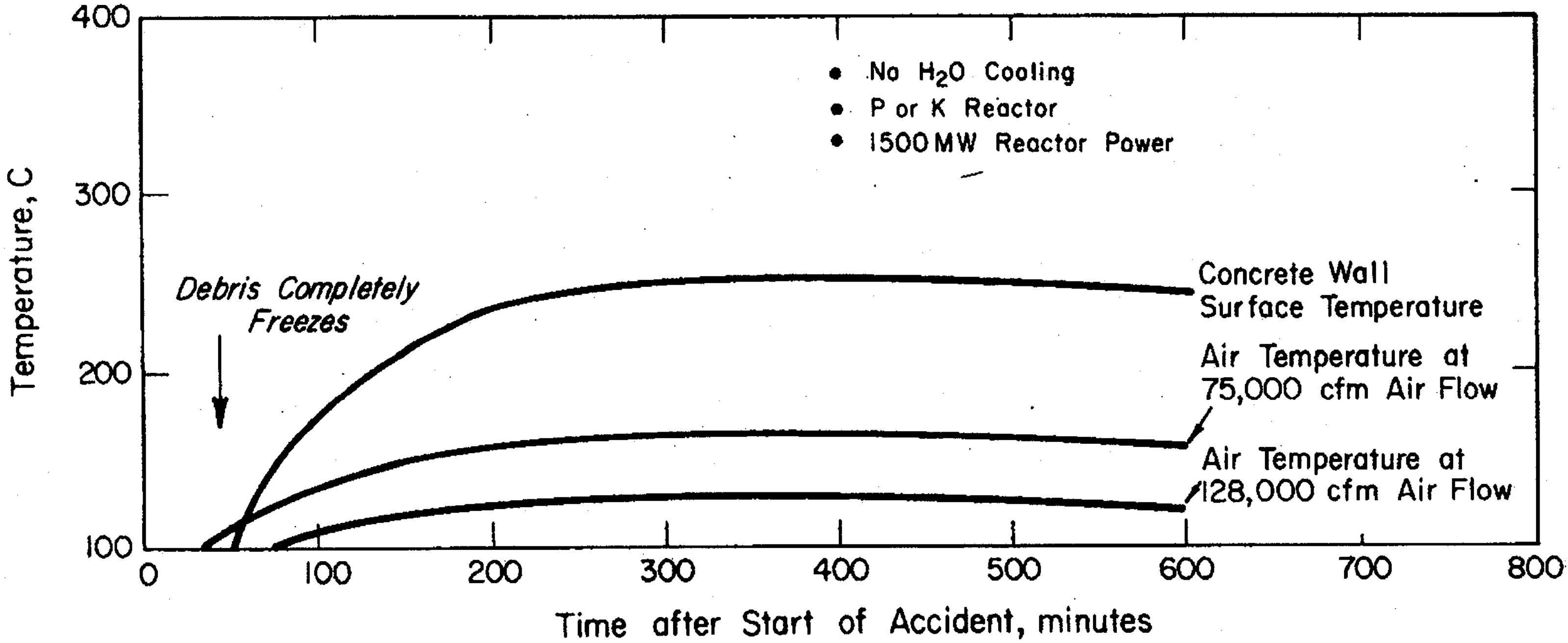

FIG. 14 CALCULATED TEMPERATURES FOLLOWING MELTDOWN OF HIGH FLUX CHARGE

URANIUM METAL CHARGE

For a full-reactor charge of uranium metal (Figure 11) in P or K reactor, no significant degradation of the confinement system would be expected for at least 30 minutes if an air flow of 128,000 cfm is maintained. At this time, the exhaust air temperature would reach a value at which desorption of iodine from carbon beds that have been in use for an extended time could become significant (about 100°C for 4-year-old carbon).[5] For newly installed carbon beds, no significant desorption would be expected for at least 3 hours when the air temperature would reach $\sim$220°C. Operation of the exhaust fans, however, is estimated to continue until the air temperature reaches $\sim$250°C, which would occur in about 5 hours at the design air flow of 128,000 cfm (or about 2-1/2 hours if the air flow is 75,000 cfm because of dusty filters or operation of only one fan).

The debris would spread over about 3500 ft^2 of floor area in about one hour and freeze. The average debris thickness would be about 2-1/4 inches. This area is about 2/3 of the pump room floor area (Figure 15); therefore, most of the debris would not flow into the sumps at the ends of the pump room where heat transfer to the exhaust air would be reduced. Wall surface average temperatures would exceed 500°C in about 5 hours at which time surface spalling would occur, but the walls would remain structurally intact because they are at least 5 feet thick. In C reactor, the air temperature would increase more rapidly than in P or K reactor because the debris would be hotter when melt-through occurred. A peak temperature of $\sim$270°C for the exhaust air was calculated as shown on Figure 12. The difference in peak exhaust air temperature between C reactor and P (or K) reactor is insignificant.

URANIUM-ALUMINUM ALLOY CHARGE

For a full-reactor charge of uranium-aluminum alloy (Figure 13) in P or K reactor, the exhaust air temperature also would reach 100°C in about 30 minutes. However, at a total process area air flow of 128,000 cfm a peak exhaust air temperature of only about 190°C would be reached, which is less than that estimated to cause fan failure or significant desorption from newly installed carbon beds. At 75,000 cfm, the peak temperature of 250°C is marginal for continuous operation of the fans. These maximum air temperatures would occur about 4 hours after the accident, then decrease at a rate of $\sim$2°C per hour for at least the next 12 hours, the extent to which the calculations were carried. Wall surface average temperatures would not exceed about 430°C even at reduced air flow; therefore, significant spalling of the concrete walls would not be expected. The calculated temperatures are lower for an alloy charge than for a uranium metal charge because of the higher decay heats caused by ^{239}Np in the latter. About half of the floor area of the pump rooms is calculated to be covered by the fuel debris with an average thickness of about 1 inch before the debris freezes.

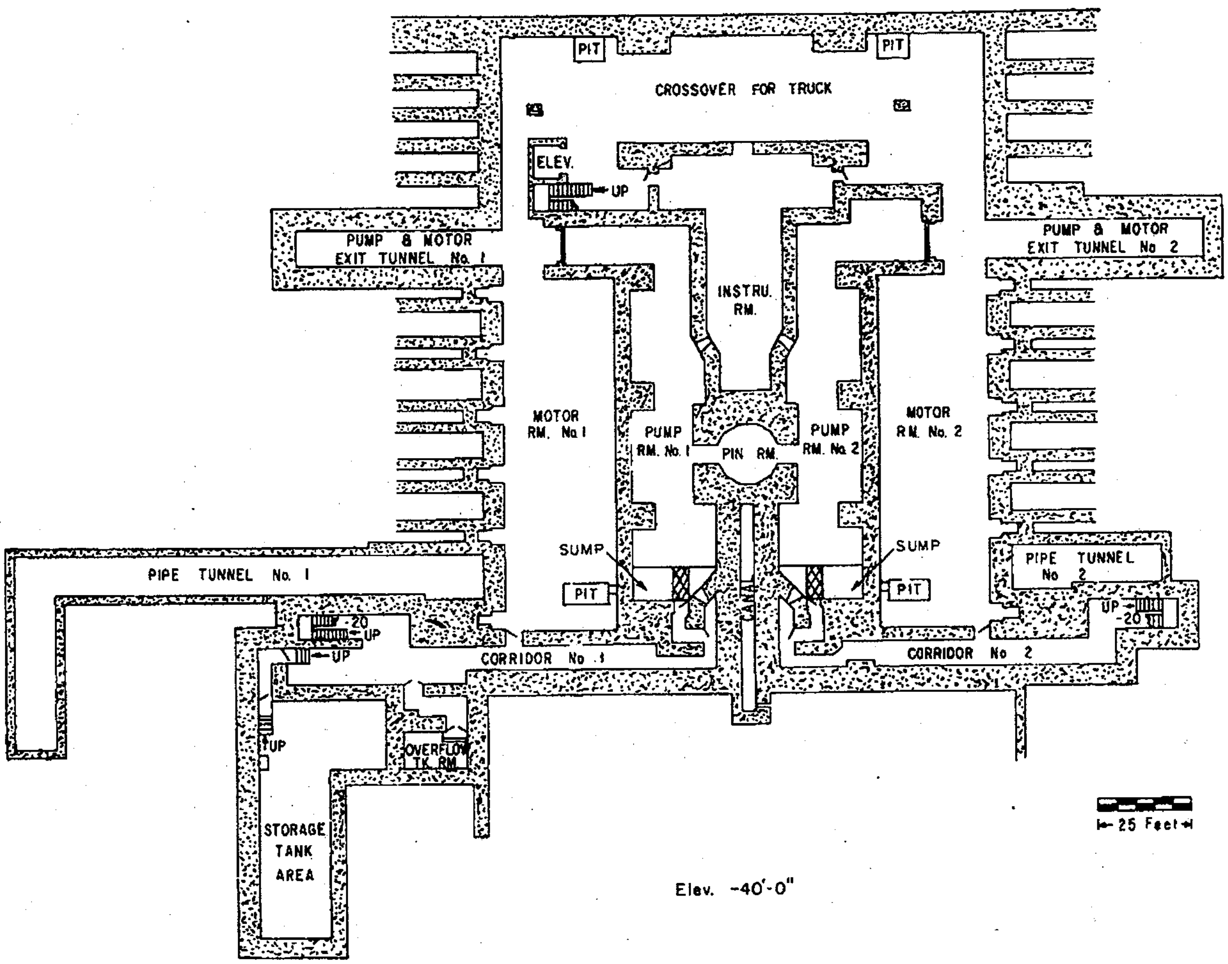

FIG. 15 C REACTOR BUILDING FLOOR PLAN AT -40 FT LEVEL

HIGH FLUX CHARGE

Design performance of the activity confinement system probably would be maintained following a meltdown of a high flux charge because of the lower reactor power (less decay heat), lower inventory of fission products, and smaller mass of materials involved without modifications. At a design ventilation flow of 128,000 cfm, a peak air temperature of about 130°C would be reached in about 4 hours, then decrease at an initial rate of about 2°C per hour (Figure 14). At a reduced flow of 75,000 cfm, the peak temperature would be 165°C. Continuous operation of the exhaust fans would be expected at this temperature. For new activated carbon, desorption of iodine would be insignificant at 165°C. The debris would cover only about 1500 ft^2, or less than 1/3 of the floor area, with an average thickness of less than 1/2 inch.

The refined analysis of effects with no light water addition confirms the earlier conclusion[1] that some additional means to cool the debris would be necessary to preserve the activity confinement system.

Effect of Parametric Variations

The following parameters in the calculation (LAVA) were varied to determine which were significant:

- mass flow of air through below-grade process areas (10,000 to 90,000 cfm)
- inlet air temperature (10 to 40°C)
- effective area of process piping for heat transfer (1000 to 1560 ft^2)
- effective area of concrete walls for heat transfer (20,000 to 40,000 ft^2)
- thermal conductivity of concrete (0.5 to 2.0 pcu/(hr)(ft)(°C))
- natural convection heat transfer coefficient (0.122 $\Delta t^{0.333}$ to 0.366 $\Delta t^{0.333}$ pcu/(hr)(ft^2)(°C))
- emissivity of both concrete and process piping (0.5 to 0.8),
- reactor power at time of postulated accident (2000 to 3000 MW).

C.G. 15.5

The reactor power was the only variable to affect significantly the time at which the lava would flow to the floor; at 3000 MW, the time was 15 minutes, and at 2000 MW, the time was 30 minutes. Other parameters affected the time only about one minute.

Maximum exhaust air temperatures were affected noticeably by variations in several of the parameters. One of the most significant effects was the mass flow of air. Temperatures of 530 and 180°C were calculated for mass flows of 10,000 and 90,000 cfm from the below-grade areas, respectively.

A decrease in the inlet air temperature from 40 to 10°C resulted in a decrease in exhaust air temperature from 260 to 235°C. The calculated air temperature increased from about 230 to 290°C when the effective building surface area was increased from 20,000 to 40,000 ft^2. The thermal conductivity of the concrete walls is also significant. Variations of from 0.5 to 2.0 pcu/(hr)(ft)(°C) decreased the air temperature from 260°C down to 175°C.

The type of fuel in the reactor also affects the exhaust air temperature. More decay heat is generated for a uranium metal charge than for a uranium-aluminum alloy charge because of the decay of ^{239}Np. The higher melting temperature of uranium metal and the larger fraction of uranium in the molten metal would also cause an earlier penetration of the process piping.

Parameters that had little effect on the air temperature included the emissivity of both the stainless steel piping and the concrete walls, and the effective surface area of the stainless steel piping for heat transfer. The insensitivity of the calculated air temperature as a function of pipe area indicates that the distribution of the fuel among the six outlet systems is relatively unimportant.

Although the convective heat transfer coefficient is significant in determining the air temperature for large surfaces, this coefficient is basically affected only by the air velocity and the temperature difference between the surface and the air. Calculations indicate that the mean velocity of the air in the process areas has an insignificant effect on the coefficient. Therefore, the coefficient would be controlled by the temperature difference between the surface involved and the bulk exhaust air.

Effect of Air Temperature on Methyl Iodide Formation

At high temperatures, the conversion of elemental iodine to methyl iodide increases rapidly. Based on the method in References

6 and 7, the amount of methyl iodide that could be formed during the postulated accident, with no light water addition, is less than 0.1% of the reactor inventory of iodine. This is a factor of 10^3 to 10^4 greater than would be expected if light water were added immediately after the fuel melted. Most of the iodine would be released early in the course of the core meltdown. Therefore, methyl iodide formation would be greatly suppressed because exhaust air temperature would be low.

Effect of Air Temperature on the Activity Confinement System

No deterioration of any component in the activity confinement system would be expected below 150°C.[6,7] At about 150°C, deterioration of gaskets and adhesive on the moisture separators, particulate filters, and activated carbon beds could begin. Also the binder in the particulate filters could begin to deteriorate. However, the efficiency of the filters would not be affected significantly because:

- Significant deterioration would not occur until after most of the fission products were confined.
- All components of the filters are self-extinguishing.
- The strength of the filter is adequate to resist damage under these accident conditions without binder.
- Underwriters Laboratory tests showed that damage to the filters is negligible below 370°C.

In ignition tests of a full-size prototype of an SRP activated carbon bed, the neoprene gaskets were not significantly damaged by exposure to an air flow of 70 ft/min at 280°C for 35 minutes.

The rate of iodine desorption from activated carbon is a function of temperature, air velocity, iodine loading, bed depth, moisture content, and service history. Experiments at SRL showed that more iodine was desorbed from used carbon than from new carbon. The used carbon had been in service for 45 months and had been regenerated by passing air through the bed at 60°C for 48 hours. To test desorption, air at 140°C was passed through the bed for $\sim$5 hours. The iodine desorbed was $\sim$0.2% for the used carbon but less than 0.001% for the new carbon. The carbon in both experiments was Type 416, a product of Barnebey-Cheney Company. Tests at SRL to measure desorption over a wide range of temperature, iodine loading, service, moisture content, and air flow are part of a continuing program related to capability of the SRP confinement system.

If the exhaust air were ~250°C for a sustained period of time, the exhaust fan motors would approach their rated limit of 90°C. Heat is transferred to the motors primarily by convection from the fan casing; however, compensating cool air is drawn into the rooms through "blast gates" immediately upstream of the fans. Belts used to drive the fans would deteriorate after several hours under these adverse conditions.

At about 340°C, ignition of the activated carbon beds would be expected. This is the measured ignition temperature of new Type 416 carbon in stainless steel at 65 ft/min face velocity. The ignition temperature of Type 416 carbon increases with service.[8]

Heat Removal from Potential Water Sources at the -40 ft Level

Although neglected in the analyses previously described, water would most likely be present at the -40 ft level, which would provide partial cooling of the molten fuel mass. The potential sources include:

shield cooling water systems,

D_2O process pump cooling water systems,

D_2O or H_2O cooling water systems,

miscellaneous H_2O piping, and

the proposed confinement heat removal system.[2]

If a reactor lost its primary coolant and the emergency cooling system failed, the core would melt down. The proposed confinement heat removal system is designed to cool the molten debris that would result from this postulated accident. The design includes two below-grade spray systems, each with a cooling water header with nozzles for spraying water on three reactor effluent lines and on the floor of the monitor-pin room beneath the reactor (see Figure 16). The sprays will be automatically actuated by heat-sensing devices requiring no external power.

The water from the sprays will collect on the floor of the below-grade areas, where it will provide cooling for any fuel that melts through the reactor structure or effluent piping. Thus, the system should perform its function regardless of the accuracy of the analysis of molten fuel behavior and the extent of damage within the reactor building. The system requires no external power or human intervention and, except for the fusible heat sensors, can be easily tested. Its presence will create no significant new reactor hazards.

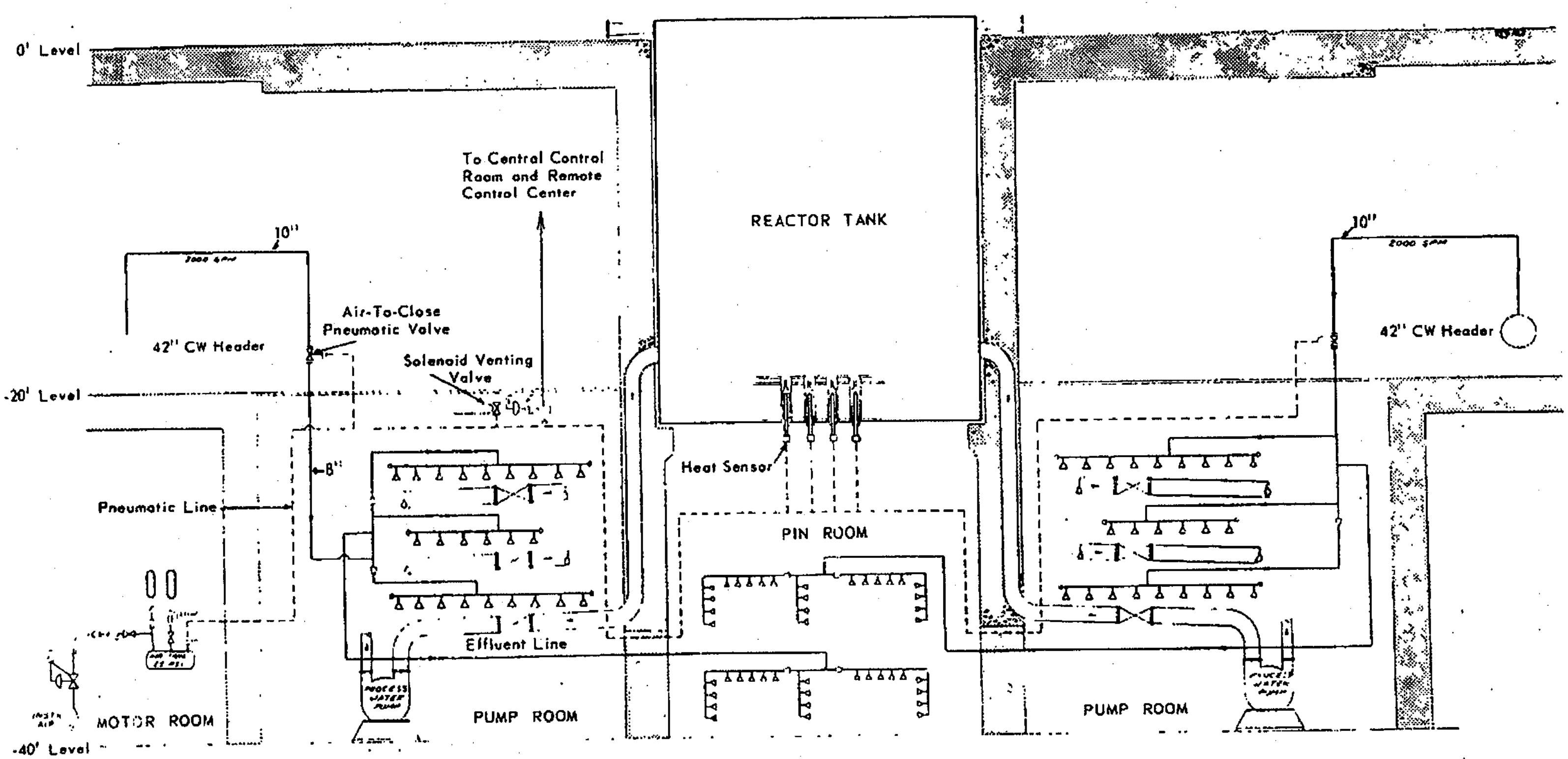

FIG. 16 CONFINEMENT HEAT REMOVAL SYSTEM

In addition, the existing reactor-room spray system is available as a supplement to the proposed below-grade spray systems. The reactor-room spray is manually actuated, and this spray water (2000 gpm) will also flow to (and be contained in) the below-grade area of the reactor building.

The occurrence of a major meltdown accident without rupture of several light-water lines is highly improbable because of the proximity of the molten debris to these lines. Failure of either the top or bottom shield would contribute the inventory of the shield cooling system plus a continuous flow of about 30 gpm from the automatic make-up. Failure of the thermal shield coolant system also would be likely at the -40 ft level, because the pumps for this system are located adjacent to the D_2O process pumps. Rupture of the thermal shield would provide the inventory in this system plus 30 gpm make-up flow.

Each of the six D_2O process pumps is equipped with a light-water cooler for the seals with a normal flow of 5 gpm each. The pump rooms are also equipped with water supplies for general purposes such as cleaning. In addition to the water from the ruptured light-water lines, D_2O from the reactor process system would be present in the vicinity of the molten fuel.

Hence, a minimum of about 100 gpm light-water flow plus the inventory would be expected from destruction of the systems listed above. Vaporization of this water could remove about half of the decay heat being generated by the time the debris reaches the floor of the pump rooms and all of the decay heat within 10 hours after the accident. Vaporization of 100 gpm of water would not adversely affect the activity confinement system as indicated by tests under far more stringent conditions.[6]

ADDITION OF LIGHT WATER AFTER CORE MELTDOWN

Because of the potential for steam explosions and chemical reactions if water is added to molten metal, a detailed review and analysis of the consequences of molten metal and water interactions was made. A steam explosion results from the extremely rapid generation of steam associated with rapid contact of molten metal and water. A metal-water reaction is a chemical reaction caused by the interaction of hot metal (usually molten) and steam. This chemical reaction produces heat and hydrogen. The hydrogen could explode or burn, and the heat of reaction would add to the total energy of the system. The sequences of potential metal-water interactions are shown schematically in Figure 17. The consequences of steam explosions and metal-water reactions in the reactor building following a loss-of-coolant accident have been studied and are discussed in detail in the following sections.

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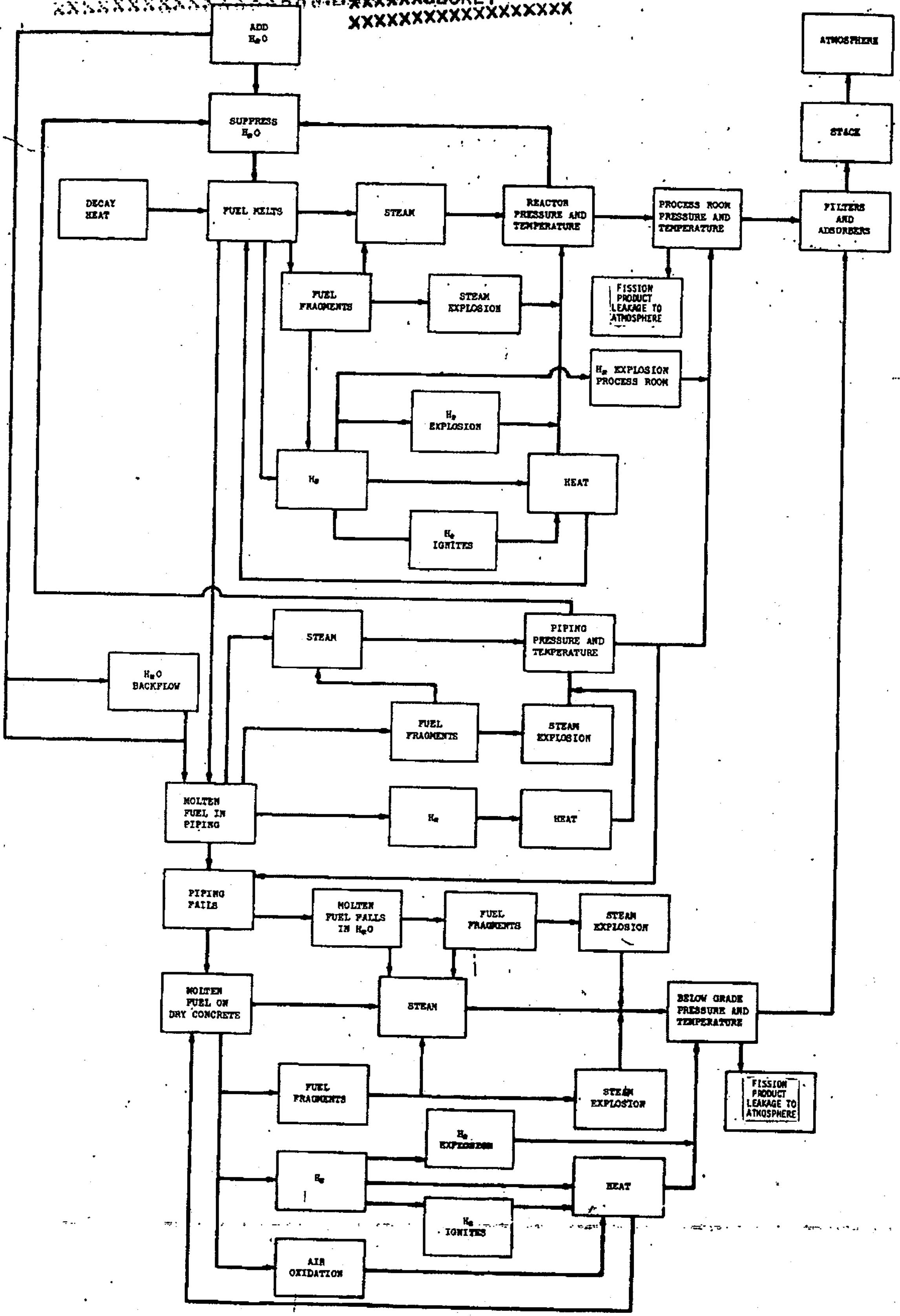

FIG. 17 ACCIDENT FLOWSHEET-LIGHT WATER ADDED AFTER MELTING

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Mechanisms of Interaction

STEAM EXPLOSIONS

Damaging steam explosions caused by molten metal in contact with water have occurred in several industries.[9,10] The mechanisms and conditions necessary for steam explosions by molten metal falling into a pool of water have been studied by Long,[11] Hess and Brondyke,[12] and Brauer.[13]

Long concluded from some 880 experiments that explosive forces are created by the entrapment of a very thin layer of water at the bottom surface of the water pool. An initial explosion disperses the small molten particles throughout the water so that rapid vaporization of the water is possible because of the increased metal surface area.

Long observed that:

- Explosions occurred when the depth of the water was between 3 inches and 10 inches. No explosions occurred with depths between 20 inches and 30 inches probably because the metal was cooled before it reached the bottom of the container. With depths less than 2 inches, the metal was simply blown out of the container. The depth of water necessary to cause an explosion depended slightly on the metal temperature. Lateral dimensions were found to influence the magnitude of the explosion. Explosions in containers 48 inches square were milder than in containers 24 inches square.
- Explosions occurred only when the metal was initially molten.
- Explosions occurred when the metal fell 1-1/2 to 4 ft into water. No explosions occurred when the metal drop was 10 ft, probably because the metal was cooled during the vertical drop.
- Coatings of paint or grease on the container bottom prevented explosions.
- Soluble oils and wetting agents prevented explosions, but sodium chloride seemed to enhance explosions.

Brauer[13] proposed a mechanism similar to that used by Long except for the method of trapping water. Brauer suggests that water is trapped in the molten metal as it falls through the water, not necessarily trapped under the metal as it strikes the floor.

Steam explosions caused by high pressure water striking a pool of molten metal have been studied at TRW Systems.[14] Tests were run in which a column of water, supported by a thin diaphragm, was placed above a pool of molten aluminum in a vacuum. The diaphragm was punctured, and the water impacted the molten metal, with the water surface parallel to the metal surface. This caused the initial fragmentation of the metal. Peak pressures of ~3000 psi were measured. In the SRP system, water would fall by gravity from the plenum to the molten fuel in the reactor tank. All the water and fuel would not impact simultaneously with parallel surfaces. Without simultaneous impaction, the water that reaches the molten metal initially would cushion the main impact and produce a much lower disruptive force. Hence, the peak pressure associated with the metal-water interaction would be much less severe than in the TRW controlled tests.

Based on the results of investigations reviewed, some arguments can be made pro and con concerning the probability of steam explosions. In the analyses presented in the following sections, it is assumed that any metal-water interaction with a probability of causing an explosion will cause an explosion. Calculations are made to determine the amount of energy in the molten fuel that could be converted to shock energy. The calculations are based on analyses of the SPERT-ID destructive tests and laboratory shock tube tests.[14,15] Pressure pulses during the SPERT tests indicate that ~10% of the total thermal energy (sensible heat plus heat of fusion) of the molten metal (above 20°C) could be converted into high pressure steam to produce a shock wave.[14] The SPERT-ID tests involved rapid and extensive in-core melting and vaporization of large amounts of fuel initiated by a prompt increase in power. These conditions would cause higher destructive pressures than in a slower loss-of-coolant accident where molten metal might fall into an open pool of water.[14] Therefore, the 10% energy conversion factor used in the following analysis is considered to be conservative. The energy available for shock wave generation is related to that produced by an equivalent mass of TNT.[35] Overpressures are then calculated from the TNT values and compared with estimated shock pressures required for gross failure of various critical areas of the confinement system.

METAL-WATER REACTIONS

Metal-water reactions must be considered from two standpoints: 1) the total energy released to the system, and 2) the possibility and consequence of hydrogen explosions. The explosions may create large shock pressures similar to those caused by steam explosions. The rate and extent of these reactions are as follows with respect to loss of coolant accidents in SRP reactors:

Reaction Rate Equations. The reactions of specific concern at SRP are those of molten fuel and cladding with light water.

• Aluminum-Water. At high temperatures aluminum reacts with water according to the following equation:

$$2Al + 3H_2O \rightarrow Al_2O_3 + 3H_2$$

The heat of reaction is 256 kcal/gram-mole of aluminum (above ∿1100°C).

The reaction rate of aluminum and H_2O (steam) can be described by a cubic rate law between 800 and 1300°C and at one atmosphere pressure.

$$W^3 = 4.0 \times 10^9 t\ [\exp(-73{,}500/RT)]$$

where

W = metal reacted, mg/cm^2 of surface area (1 mg Al reacted forms 1.246 ml (STP) H_2)

t = time, min

R = gas constant, 1.987 cal/(mole)(°K)

T = temperature, °K

Within this temperature range, the reaction results in the formation of a protective oxide film. Between 1400 and 1600°C, the oxide film apparently disintegrates[16] and the reaction follows a linear rate law:

$$W = 2.5 \times 10^6 t\ [\exp(-73{,}500/RT)]$$

Higgins and Schultz[17] determined experimentally that below about 1170°C metal-water reactions are insignificant.

Above 1750°C, ignition of the aluminum in contact with steam or air can be expected.[18] The aluminum is apparently vaporized and then reacts. The reaction product is a very fine aluminum oxide powder.

• Uranium-Water. At high temperatures, uranium reacts with steam according to the following equation:

$$U + 2H_2O \rightarrow UO_2 + 2H_2$$

The heat of reaction is 142 kcal/gram-mole of uranium. Scott[19] showed experimentally that at about 1200°C, almost all of the solid reaction product would be in the dioxide form. U_3O_8 would be expected only below 300°C.[20]

Below 400°C, the reaction follows a linear rate, probably because the oxide formed on the surface is nearly colloidal. From 500 to at least 1600°C, the reaction follows a parabolic rate law.

The following equation[21] was determined for 600 to 1200°C

$$V^2 = 1.95 \times 10^5 t \, [\exp(-18{,}600/RT)]$$

where

V = volume of hydrogen generated from the uranium-water reaction, ml(STP)/cm^2 of reaction area.

Between 1200 and 1600°C the activation energy is 25 to 30 kcal/mole and the reaction follows the equation:

$$V^2 = 8.69 \times 10^6 t \, [\exp(-30{,}000/RT)]$$

Rate Limiting Mechanisms. Metal-water reaction rates are controlled both by reaction rate laws discussed in the preceding paragraphs and by rate-limiting mechanisms. During a reactor accident the physical and chemical reactions are complex because of the interactions involved. The total time required for a given amount of the overall process to occur is the sum of the times required for each of the individual processes. The maximum rate that can be approached by the overall process is the rate of the slowest step.[22]

Rate limiting mechanisms that affect metal-water reactions are (1) availability of steam, (2) thickness of oxide (reaction product) layer, and (3) the metal surface temperature. The previously discussed reaction rate equations are obtained from tests in which sufficient steam is present near the reacting surface. In the following analysis it is assumed that unlimited steam is available for the reaction. If only limited steam is available, the diffusion of steam to the surface could limit the reaction rate.

The reaction rates, as discussed previously, depend on the temperature at the metal-metal oxide interface and implicitly on the transport rate of the reactants to the reaction site. After a very short reaction time (on the order of milliseconds[23]) a metal oxide layer is formed on the surface. After formation of the oxide layer, the reaction proceeds by diffusion of either the oxide ions or metal ions through this layer. The reaction rate depends on whether the oxide film adheres to the surface or whether it cracks and spalls off. As shown in Figure 18, after the initial reaction, the reaction rate is probably severely limited by this oxide layer.

Metal-water reactions are significant only at high temperature, generally at temperatures well above the melting point. To sustain a reaction, a high temperature environment must exist over a period of time. Heat generated in the metal by decay heating and by reaction must be greater than the heat losses to maintain the elevated temperature. Excessive water in the system could cool the metal and terminate the reaction.

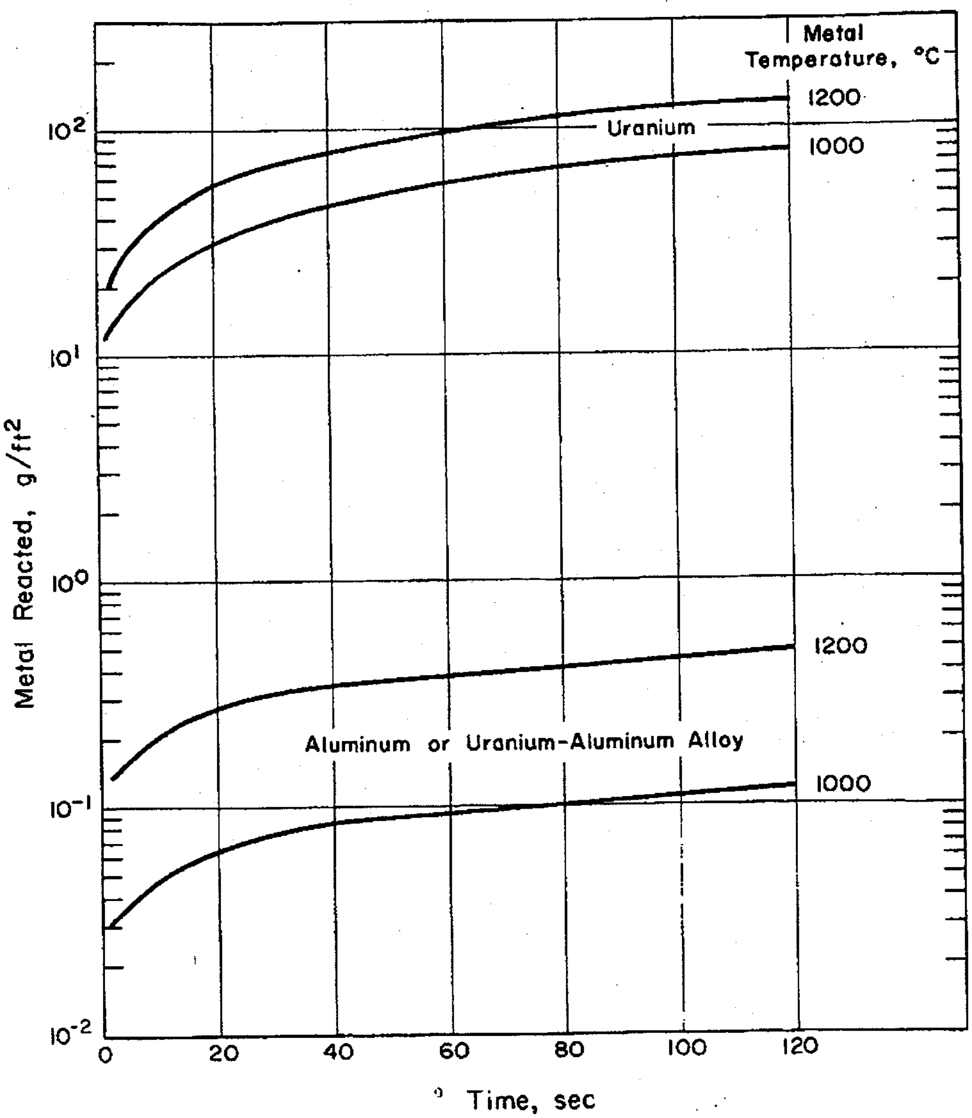

FIG. 18 EFFECT OF METAL TEMPERATURE AND COMPOSITION ON METAL-WATER REACTION

[redacted]

Extent of Metal-Water Reaction. The extent of the reaction (i.e., amount of energy released and hydrogen formed) is dependent upon the metal surface area in addition to the reaction rate as discussed in the preceding section. For significant reaction, the metal must be dispersed such that surface area is large. The breaking up of the molten metal into small fragments could occur following a steam explosion.

Hydrogen Combustion. Combustion of hydrogen generated by chemical reaction between water and molten metal is important for two reasons: burning could increase the temperature of the exhaust ventilation air, and damaging explosions could result. The exothermic heat of reaction is 5.6×10^4 pcu/lb-mole of hydrogen reacted. The molar quantity of hydrogen produced is equal to the steam consumed in the metal-water reaction.

A hydrogen mixture is flammable if the composition is within the "flammability limits". A combustible gas mixture has two limits of flammability, an upper limit and a lower one. Only when the composition is between these limits will the gaseous mixture burn or explode. For hydrogen-air mixtures the flammability range generally is quoted as between 4.1% and 74%, although some variation exists as a function of temperature, pressure, and dilution by a third component, such as steam. The flammability limits of hydrogen-air-steam mixtures are shown in Figure 19.

Exceedingly small energy inputs cause hydrogen ignition, particularly when the mixture is well within the flammability limits. If hydrogen is not ignited from an external source, hydrogen would be ignited when the temperature exceeded the spontaneous ignition temperature. This temperature depends on the concentration of hydrogen and steam.[24]

In the following sections, hydrogen is assumed to exist in flammable mixtures if there is a possibility of a metal-water reaction. It is also assumed that a source of ignition is present. Because the minimum spontaneous ignition temperature is ∿520°C,[25] it is very unlikely that spontaneous ignition would occur.

Hydrogen Explosion Analysis. Based on the instantaneous combustion model of Moore,[26,27] a maximum pressure of ∿80 psig could be developed at the explosion site. An analysis based on the work of Brode[28] gave results which agreed well with those of Moore.

The maximum pressure is independent of the volume of the combustible mixture. Even though a large volume of hydrogen would have the same maximum pressure as a smaller volume, the larger volume mixture would cause a larger shock pressure at the wall because the attenuation distance would be less.

The instantaneous combustion model was used to calculate hydrogen explosion pressures for ambient pressures near atmospheric for various hydrogen concentrations.[27] The calculations greatly overpredict shock pressures, taken from limited available data. All the calculations are based on zero steam content. Experiments and analysis indicate that the presence of steam in the reactor room would reduce the maximum shock pressures.[27] Pressures at various distances from the explosion site are predicted by geometric scaling laws.[29]

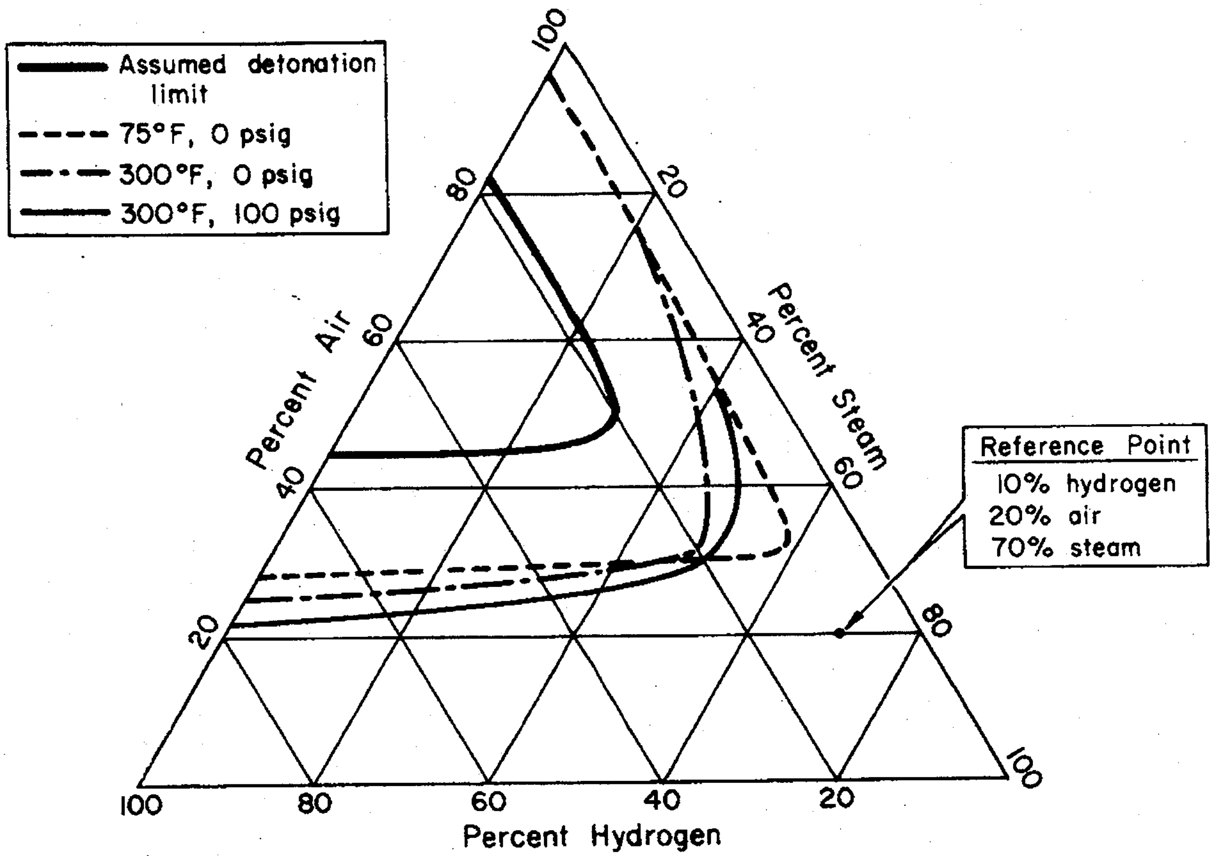

FIG. 19 FLAMMABILITY LIMITS OF HYDROGEN-AIR-STEAM MIXTURES

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Location and Degree of Interaction

BELOW GRADE

Possible steam explosions from molten metal falling into water below the reactor are calculated not to significantly damage the building structure or confinement system.[2] Explosions could occur when molten metal melts through the effluent piping and/or the bottom shield, and falls into water on the floor. Potential water sources are discussed in the section Heat Removal from Potential Water Sources at the -40 ft Level above.

The shock pressures in the below-grade areas are calculated assuming that:

- One reactor effluent pipe full of molten metal drops into water.
- The shock reflection from walls and floors is compensated by shock absorption in building equipment such as pipes and pumps.

The calculated energy in the shock wave is 2.5×10^5 pcu. The overpressures associated with this wave are $\sim$140 psig at 15 ft, the approximate distance of the nearest concrete wall (Figure 15), 28 psig at 35 ft, and 14 psig at 50 ft.

Steam explosions could scatter a portion of the molten fuel over the walls and ceiling where no water is available for cooling. Only a slight increase in exhaust air temperature is expected because of the small amount of fuel involved, heat transfer into the concrete, and cooling by the water vapor in the air. The spray mist from the proposed confinement heat removal system will also offer cooling of the scattered fuel.[2]

The effects of metal-water reaction in the below-grade areas will be less severe than the steam explosions. The molten fuel either will be quenched as it falls in the large quantity of water on the floor or will cause a steam explosion. The molten fuel would be scattered over the walls and ceiling. In the former case, the metal is cooled and the reaction is quickly terminated. Some metal-water reaction is expected following a steam explosion while the metal is still hot, vapor is present in sufficient quantities, and the metal is greatly fragmented. The hydrogen produced in the metal-water reaction could cause an explosion. The explosive energy is calculated to be $\sim$10% of the energy of a below grade steam explosion and the damage would be much less. Baker[32] showed that hydrogen cannot be produced at rates sufficient [redacted]effects of a steam explosion.

Expected shock pressures and the estimated shock pressure to cause gross failure for several critical areas of the activity confinement system that may be damaged by these pressures are summarized in Table II and discussed in the following paragraphs.

TABLE II

Structural Limitations

	Max Expected Shock Pressure, psig[a]	Estimated Shock Pressure for Gross Failure, psig	Is Failure Acceptable?[b]
Discharge canal	$\sim$140	$\sim$400-900	No
Concrete walls and ceiling in below-grade areas	$\sim$140	$\sim$400-900	Yes
Concrete slabs above main heat exchangers	$\sim$12	$\sim$20-100	Marginal
Sheet metal exhaust ducts	$\sim$140	$\sim$1-2	Yes
Exhaust fan casings	<0.1	$\sim$1-2	No
Filter compartments	$\sim$0.2	$\sim$3	No

a. These values are the result of improved calculations from DPST-68-546, Reference 2.

b. If failure occurs, will the activity confinement system function as designed?

Discharge Canal. The wall that separates the discharge canal from the pump rooms would not be breached by a steam explosion. Gross failure or large cracks in the wall could cause major leakage of water from the disassembly basin and subsequent decreased cooling of spent fuel in the basin. Failure of the wall cannot be accepted. However, the maximum estimated shock pressure is $\sim$140 psig at this wall that is reinforced concrete 7-ft thick. Pressures between 400 and 900 psig are estimated to be necessary to produce gross failure of walls that are reinforced concrete 5- to 7-ft-thick (based on data for 10-inch-thick walls[30]).

In the event of a steam explosion, equipment such as pipes and valves can become missiles. Calculations of penetration distances in reinforced concrete (Reference 31) show that the missiles will not penetrate the disassembly basin wall. For calculational purposes, the missiles are assumed to be long heavy cylinders with ogive-shaped noses and are assumed to strike the wall with normal incidence for maximum penetration. Actual missiles would probably be rough fragments that would strike the wall obliquely.

Concrete Walls and Ceiling in Below-Grade Areas. The shock pressure at the ceiling of the below-grade areas, approximately 35 ft from the source of the postulated explosion, is 28 psig. The maximum shock pressure at the 5-ft-thick walls of the pump rooms is ∿140 psig. The ceiling of the below grade areas is the floor slab of the reactor room. Both shock pressures are less than the 400-900 psig required to cause gross damage. The external building structure will not be damaged because the major structural supports for the building are shielded from the postulated explosion site by the pump-room walls and ceiling.

Concrete Slabs Above Main Heat Exchangers. The concrete slabs that form the roof of the heat exchanger bays would not be displaced by a steam explosion. An estimated shock pressure of ∿12 psig would be expected at this location. A pressure of ∿20 to 100 psig is required to lift the 4 ft thick slabs which open directly to the atmosphere at ground level. Subatmospheric operation of the confinement system will not be impaired by a shock wave from a steam explosion because the slabs cannot be lifted a sufficient distance from their normal positions to cause improper reseating.

Exhaust Ducts. Some of the sheet metal ducts of the exhaust system in the below grade areas would collapse as a result of a steam explosion. Air ventilation from these areas would be reduced. However, the activity confinement system would not overheat because of cooling obtained by air from above grade process areas. External ducts, which are reinforced concrete, would not be affected by a steam explosion because of their higher strength and distance from the source of the explosion.

Exhaust Fan Casings. Circulation of building air to the air-cleaning facilities and stack will not be impaired. A calculated pressure of <0.1 psig would be present in the exhaust fan casings because of the tortuous path the shock wave must travel. A shock pressure of ∿1 to 2 psig would be required to damage the fan casings.

Filter Compartments. Estimated pressures of ∿0.2 psig are expected in the filter compartment equipment. These pressures are well below the ∿3 psig required to cause failure of the moisture separators and particulate filters.

REACTOR TANK

Steam explosions could be caused by water contacting molten fuel in the reactor tank if the emergency cooling water system was delayed until after fuel meltdown, but while the molten fuel was still in the reactor tank. Calculations show that possible steam explosions in the reactor tank will not significantly damage the building structure or confinement system. A steam explosion may cause failure of the bottom shield supports and displacement of the bottom shield downward into the pin room (Figure 3). The top shield, plenum, and forest would be moved upward, breaking the plenum skirt and roll anchors (Figures 20, 21, and 22). The plenum inlet lines would restrict the amount of motion. Except for negligible energy released to the reactor room through breaks in the plenum skirt, all the shock energy and the slower pressure buildup would be vented to the below-grade areas because the bottom shield is more likely to be damaged than the plenum. This energy will cause less extensive damage to the below-grade areas than that from a steam explosion occurring below-grade because the energy is directed downward into the pin room and is dissipated in the massive concrete walls and floor.

Instead of calculating shock pressures as was done for steam explosions in the below-grade areas, the energy to deform various reactor components is calculated, and this energy is compared to the total available energy. The assumptions made in this analysis are that:

- The total fuel inventory interacts with water.
- Half of shock energy is available to move and deform the bottom shield.
- Half of the shock energy is available to move and deform the top shield, plenum, and forest.
- 1% of the upward shock energy is transformed into kinetic energy.

The total shock energy, based on the assumptions, is calculated to be 2×10^9 ft-lbs force.

The concrete supporting structure underneath the edge of the bottom shield is estimated to fail at a static internal pressure of $\sim$75 psig.[33] Because the explosion occurs in contact with the tank bottom, shock pressures above 1000 psig are expected in the bottom shield.[14] Even though the structure could withstand dynamic stresses several times the static ultimate stress, this shock pressure would be sufficient to cause failure of the supporting structure. Hence, the bottom shield will come to rest in the pin room and the excess tank pressure will be vented below grade.

About 5×10^6 ft-lbs force of shock energy is transmitted radially to deform the walls of the reactor tank and side thermal shields. Radial deformation would collapse the annular gas spaces. Because of the short time of the pressure pulse, very little water would be vented during the explosion.

The remainder of the shock energy (50% of total shock energy) is transmitted upward through air in the reactor vessel. Conservative calculations based on momentum-energy considerations and analysis by other investigators.[34,35] show that $\sim 1\%$ of the energy in the upward traveling shock wave is transmitted from the air to the top shield, plenum, and forest in the form of kinetic energy. The other 99% of this shock energy is dissipated by distortion of the reactor upper structure and by reflection of the shock wave downward. Thus, the top components are given a kinetic energy of 10^7 ft-lbs force.

The top shield is given a sudden upward velocity. As soon as upward movement begins, the seal at the edge of the plenum (plenum skirt) is broken (Figure 20) and any pressure behind the shock front is released to the reactor room, if it has not already been released by failure of the bottom shield. Further upward movement breaks the roll anchors (Figure 21). The shield would not jam within the upper section of the tank even though all the roll anchors probably do not fail simultaneously. An analysis showed that, even though jamming is possible from a geometrical standpoint, the necessary unbalance of forces on the top shield for jamming cannot exist.[33] As the plenum-shield-forest system moves upward, it is restrained by the plenum inlet piping. Calculations, conservatively based on having only 3 inlet pipes intact, show that these pipes can absorb up to 2×10^7 ft-lbs force of energy by plastic deformation. The calculations are based on a 7% maximum elongation of the stainless-steel plenum pipes and an average dynamic stress of 50,000 psi.[34,35] The total energy in the top components (10^7 ft-lbs force) is less than that required to make the inlet piping fail. Therefore, the upper portion of the reactor would remain intact and no damage would occur above grade. Even if all the shock energy were projected upward (no energy released to the bottom shield), the upper portion of the reactor probably would remain intact and the results would not be changed significantly.

Metal-water reactions in the reactor tank would cause negligible hydrogen burning or explosions. Prior to melting of the cladding, the reaction rate would be very small because of the relatively low temperature as discussed previously. The sequence of events after melting is different for uranium-alloy charges and uranium metal charges as discussed in the following sections.

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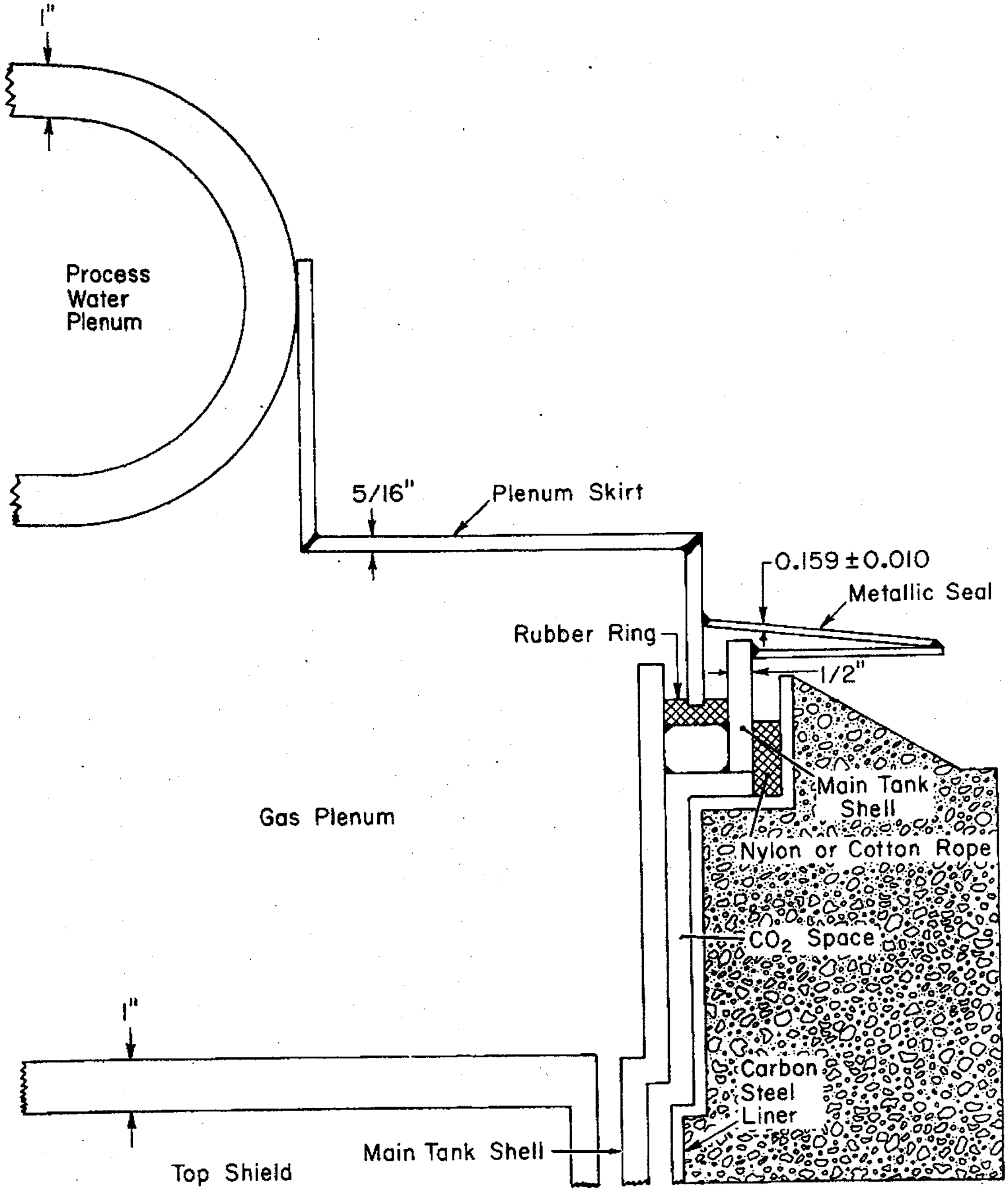

FIG. 20 REACTOR PLENUM SKIRT

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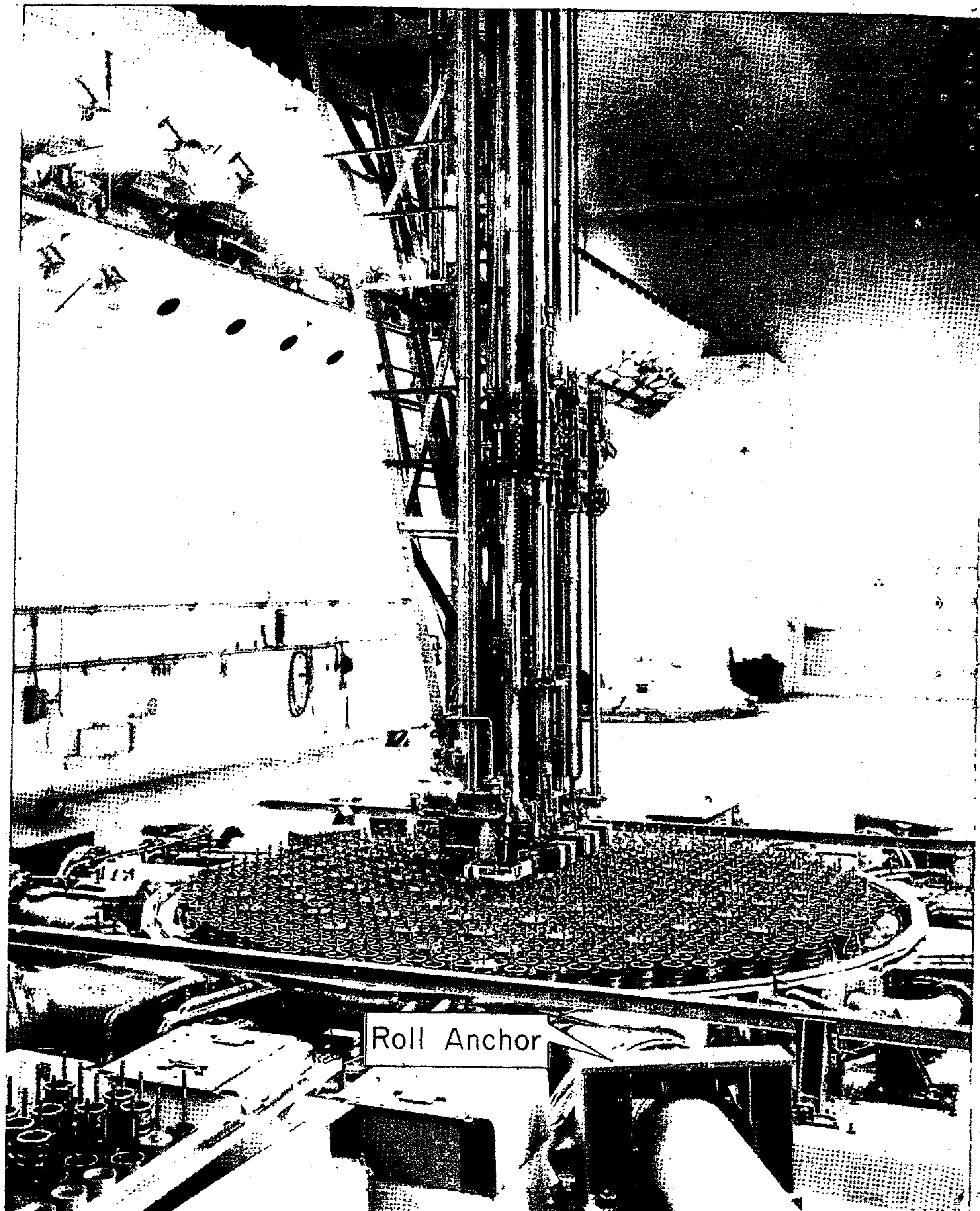

FIG. 21 PLENUM INLET ROLL ANCHOR

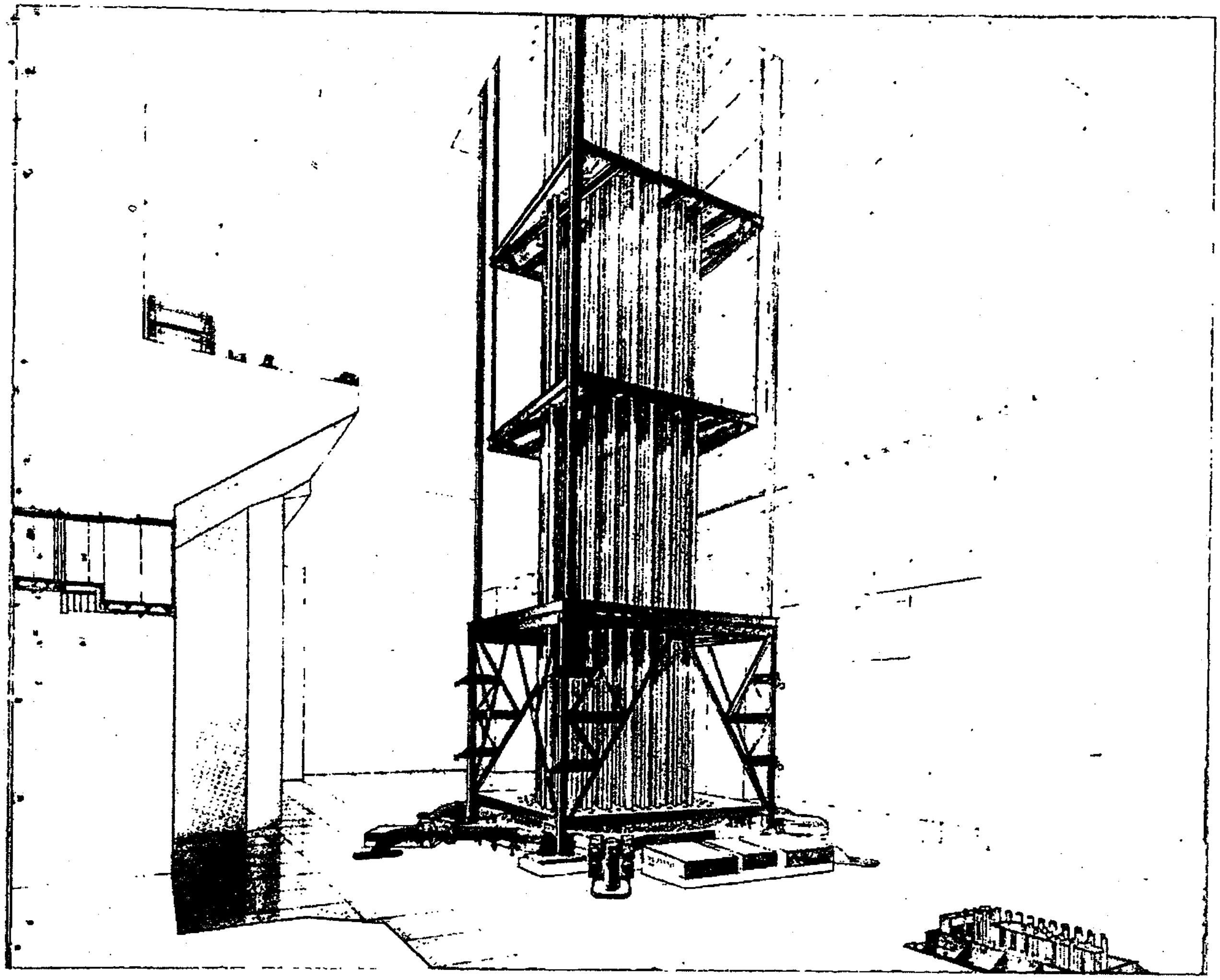

FIG. 22 ACTUATOR GUIDE TUBE FOREST

Uranium Alloy Charges. Above ~660°C the cladding and fuel would melt and fall to the bottom of the tank. In P or K reactor it is assumed that no water would be available for cooling, and the molten metal would form a large pool at the bottom. Figure 6 shows the bottom of the tank. The surface area of the molten pool would be insufficient to cause extensive metal-water reaction. In C reactor, a 15-inch depth of water would remain after a loss-of-coolant accident (see Figure 7). The aluminum cladding and fuel would melt at ~660°C and fall into the water, forming steam. No significant steam explosions would be expected because of the small amount of cladding involved and because the cladding would fall into the water over an extended period of time. Any explosions would be expected to scatter the molten fuel over the walls of the tank. Even though the surface area could be greatly enhanced, the extent of the reaction would be small because the molten particles would quickly fall back into the water and be quenched.

Uranium Metal Charges. The fuel slugs (~8 inches long) in uranium metal charges would be supported by the inner and outer housing tubes after the cladding melted. The housings would be heated primarily by the air in the tank and by conduction from the slugs. The slugs would fall to the bottom of the tank either after melting or after the housings melted. The uranium would not necessarily fall into water, even in C reactor. The uranium above water would be exposed to steam and a metal-water reaction could take place. Hydrogen from the metal-water reaction probably would not burn or explode inside the tank because the low oxygen content and high steam content in the tank would cause the mixture to be outside the limits of flammability (see Figure 19). However, if the hydrogen does explode or burn the resulting shock damage would be insignificant compared to the steam explosions.

ABOVE GRADE

The possibility exists for hydrogen from metal-water reactions in the reactor tank to escape to the reactor room (Figure 23) through breaks in the inlet piping and plenum skirt, and through the vacuum breakers and forest guide tubes. Explosive concentrations probably would exist in a very small volume because of rapid dilution by the ventilation system. The hydrogen explosion analysis is based on the explosion of a combustible mixture of hydrogen 20 ft in diameter on the reactor top. Calculations show that a maximum pressure of ~7 psig would be transmitted to the nearest concrete wall. The corresponding pressure in the filter compartments would be <0.1 psig. These pressures are much less than the pressures necessary to cause damage as shown in Table II.

Hydrogen explosion or steam surging in the reactor room could cause a momentary pressure surge that could result in some leakage of fission products from the building. Leakage would substantially increase at positive pressures of 12 inches of H_2O or more within the reactor room because of venting to the disassembly area. Actual release of fission products would be low because the pressure surge, although probably in excess of 12 inches of H_2O, would last only a few seconds, and because most of the fission products would have been released and drawn into the activity confinement system earlier.

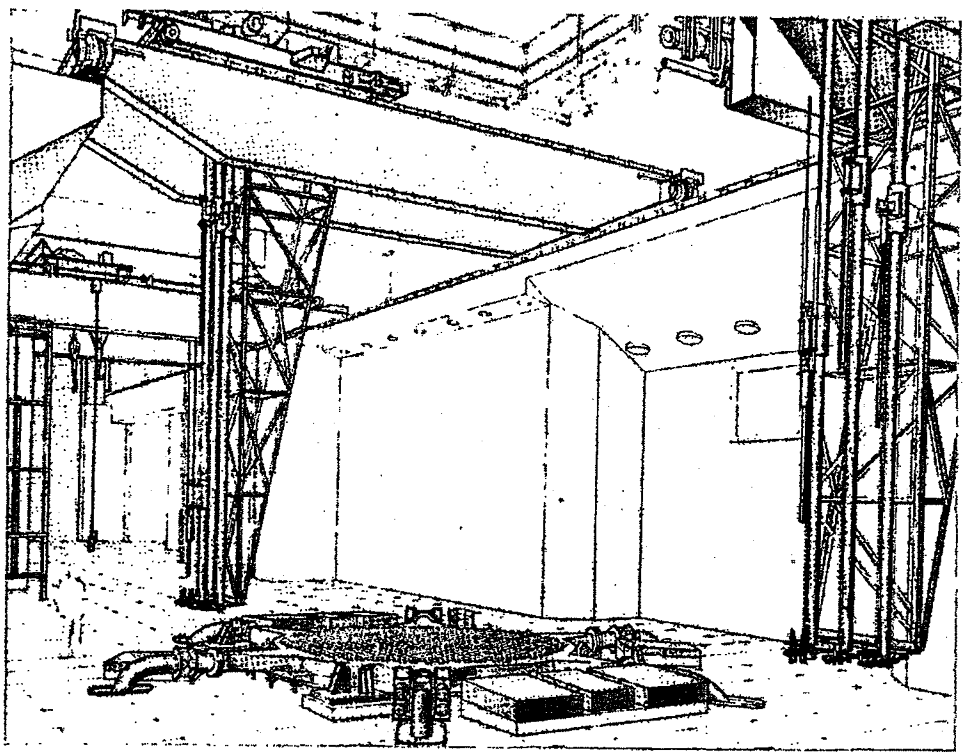

FIG. 23 SAVANNAH RIVER REACTOR ROOM

Results of Calculations

Steam explosions occurring in the below-grade areas and the reactor tank were calculated not to damage seriously the activity confinement system. The maximum expected shock pressures caused by a steam explosion in the below-grade areas and the estimated shock pressures required to cause gross failure of various structural components are shown in Table II. The energy from a steam explosion in the reactor tank would be dissipated in the tank and in the below-grade areas. Negligible energy would be released to the reactor room because the reactor upper structure is restrained by the plenum inlet piping. Steam explosions could not occur in the reactor room because of the absence of fuel.

Significant metal-water reactions could occur in the reactor tank. Hydrogen from this reaction could exist in explosive concentrations in the reactor room but the shock pressures from such explosions are not sufficient to cause damage to the activity confinement system. Although metal-water reactions could occur at the -40 ft level, they would be less significant than steam explosions.

SUBSEQUENT EFFECTS FROM CORE MELTDOWN

The preceding sections were concerned primarily with the short-term consequences of a core meltdown. Also vital to an analysis of this type is consideration of the capability to cope with problems which could arise during the ensuing days or weeks after the accident. Potential problems were identified and the subsequent effects were calculated. These problems would be largely associated with cooling water requirements, exhaust ventilation requirements, and water and debris disposition. Several miscellaneous items are also discussed in this section. The following discussion assumes that debris cooling is obtained primarily from the proposed confinement heat removal system.

Cooling Requirements for Debris

As discussed in Reference 2, a controlled volume of water from the proposed confinement heat removal system would be sprayed primarily on the effluent piping from the reactor vessel. Because of the limited surface area of the piping and the inaccessibility of some of the piping within the biological shield surrounding the vessel, total containment of the debris within the piping would be unlikely. However, several advantages are gained by this approach. First, local rather than gross melting of the piping would be more likely which would reduce the volume of lava. This would control the thermal energy available for conversion into shock pressures during possible steam explosions. Second, the heat absorbed by the spray would increase the temperature of the water which is reported to reduce the probability of a steam explosion.[11] Third, the time at which penetration of the effluent piping occurred would be delayed until a significant depth of water would be on the floor of the -40 ft level, which also could reduce the probability of a steam explosion.[11] Rapid flooding could achieve about the same depth at the time of debris penetration, but the smaller flows of the proposed system could be more easily controlled if the system were accidently actuated. Although calculations show that the SRP confinement system could survive the effects of a steam explosion, reasonable steps to preclude or to reduce the consequences of such an explosion are desirable.

About 20 hours would be required to flood the below-grade areas of the C or K reactor building with the proposed confinement heat removal system. This is based on a flow from the above- and below-grade systems of 6000 gpm and a below-grade building volume of seven million gallons. In P reactor building, the below grade volume is about 13 million gallons; the flooding time for P reactor would be about twice that of the other buildings. After the building fills with water, flow would be stopped.[2] Even with gravity flow of 800 gpm to the confinement heat removal system, adequate debris cooling would be obtained and complete flooding of the C or K reactor building would be obtained in about one week.

Based on ground water seepage into the building, outleakage would be expected to be small; thus, only a very small fraction of the fission product inventory would penetrate to the surrounding earth. The concrete disassembly basins, which are adjacent to the -40 ft areas and contain over 3 million gallons of water, much of which is 30 ft in depth, experience negligible outleakage.

The average temperature of the water in C or K reactor building at 20 hours after meltdown would be about 40°C for peak summer conditions and about 25°C for winter. Decay heat generation is calculated to exceed heat removal into the ground for about 30 days after the accident. The excess heat would raise the water temperature to about 95°C for C or K reactor in about 30 days; cooling would then begin. In P reactor building, the maximum temperature is calculated to approach about 75°C. The heat loss calculations were based on an overall heat transfer coefficient of 1.0 pcu/(hr)(ft^2)(°C) and an effective surface area of 40,000 ft^2 (approximate nominal area of floor and outer walls of below grade areas). If only half of the estimated heat transfer to the surroundings were obtained, boil-off of about 7 gpm is calculated to occur, decreasing to zero boiling in about one week.

[redacted]

Cooling Requirements for Carbon Beds

Operation of at least one of the three exhaust ventilation fans would be required for about 24 hours immediately following the accident to maintain the temperature of the carbon beds below 100°C. After that, sufficient cooling would be obtained by natural draft. Operation of at least one fan for much longer than the minimum requirement would be expected because of redundancies in the system.

The exhaust fans are centrifugal fans that are belt-driven by electric motors. These fans remove the air from the reactor room, purification area, and -20 ft and -40 ft levels, (Figures 24 and 25) through the filtration-adsorption compartments and discharge it into the exhaust stack. During normal operation, two exhaust fans are on-line (in parallel) and a third fan remains on standby. The two operating fans can move the rated flow of 128,000 cfm with a differential pressure of 5.9 inches H_2O. One fan can move 75,000 cfm. The fan suction pressure is normally -4 to -5 inches H_2O. The fan housings have a positive pressure rating of 10 inches H_2O (about 0.4 psig), but gross failure is unlikely at pressures below 1-2 psig.

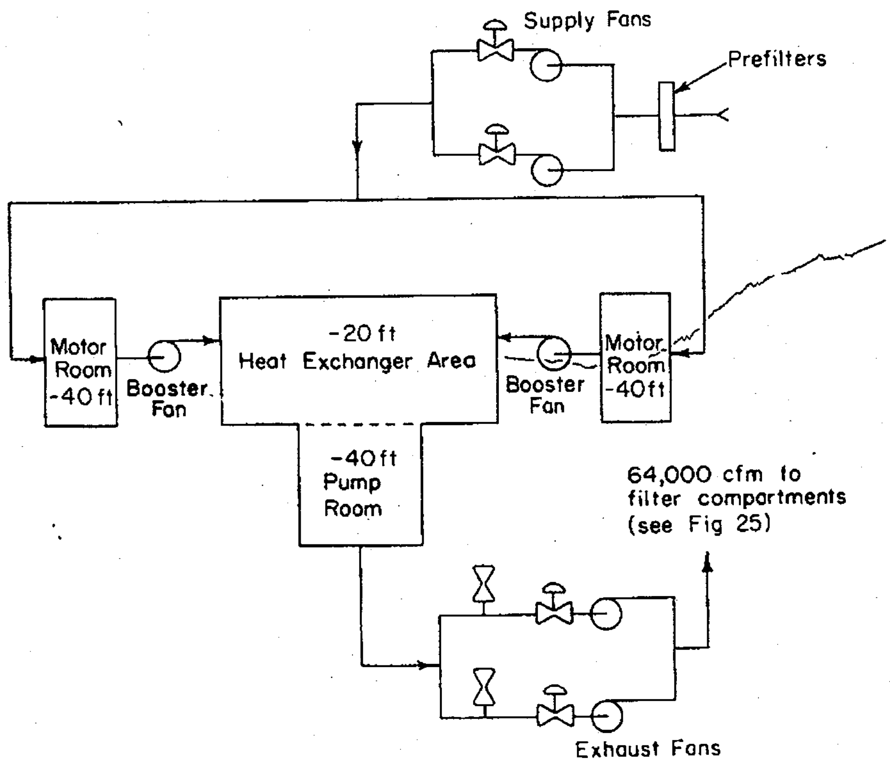

FIG. 24 VENTILATION SYSTEM FOR -20 FT AND -40 FT LEVELS

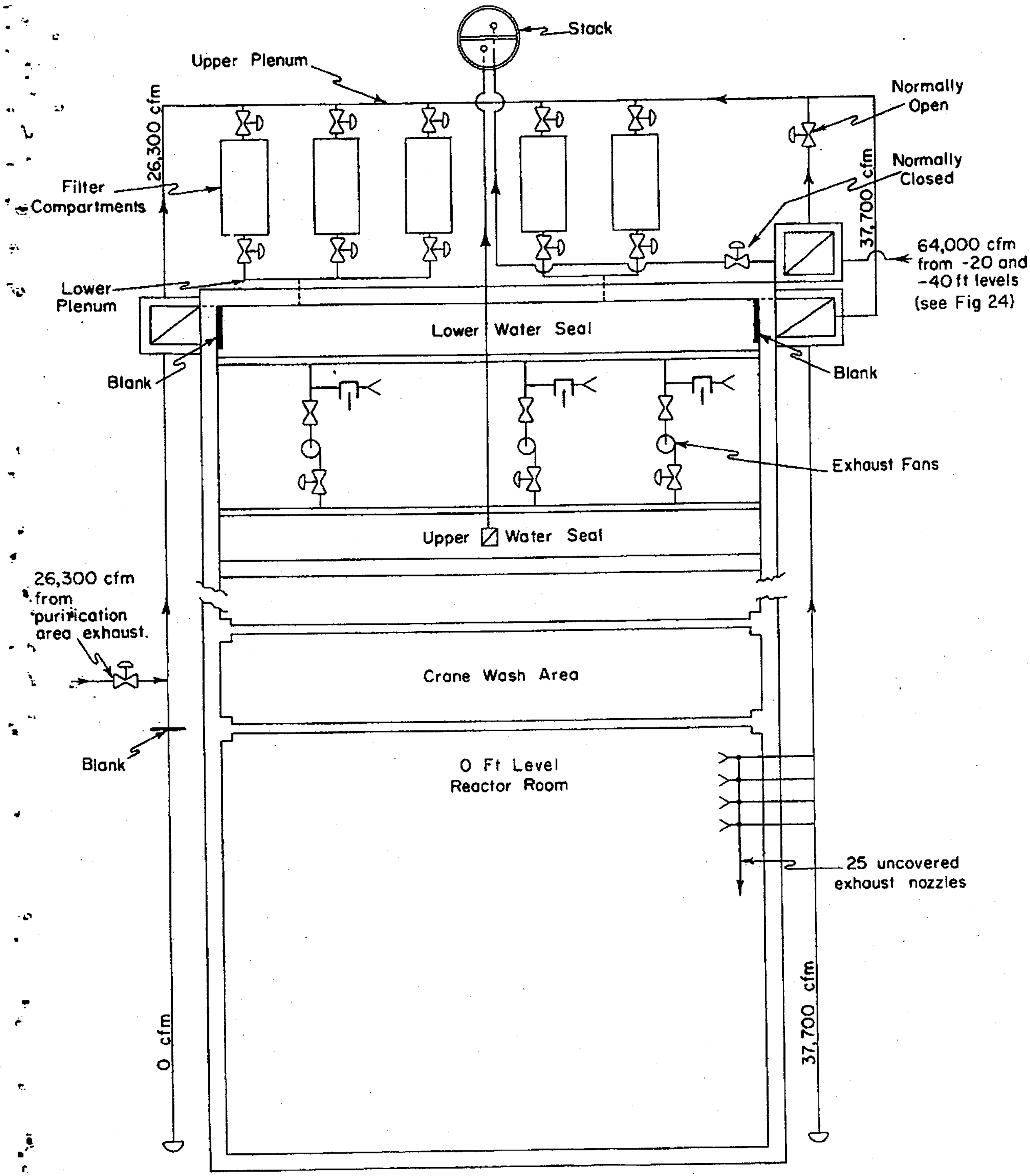

FIG. 25 PROCESS AREA EXHAUST AIR SYSTEM

Each primary exhaust fan motor has two separate sources of AC power: the electrical distribution systems for the reactor building and the confinement substation. Either power source can be selected as the primary source. The systems are supplied by the area powerhouse (except for C reactor) and outside area powerhouses (D_2O-Production Area and South Carolina Electric and Gas Company). Emergency AC power is also provided to the reactor building distribution system by diesels that start automatically if normal power fails. Automatic switching between normal and emergency power supplies provides reliable power service to the fans.

In addition to the primary motors, each fan is equipped with a backup motor. Two diesel driven generators supply power directly and independently to any two of the backup motors. Power from these sources is used solely for the exhaust fan drive motors, and a 30-day fuel supply is available for the diesels.

Debris Configuration

Although considerable dispersion of the debris would be expected, the consequence of an accumulation of debris in an adverse configuration was examined. If a mound of material formed such that an outer shell of solid debris protected an inner core of liquid, interaction between the molten debris and concrete floor would be possible. Heat removal from a mound, however, would govern its size and shape. For a configuration with a relatively low surface-to-volume ratio, such as a hemisphere, and with a high volumetric heat generation rate, the surface heat flux could be in excess of the capacity to remove the heat by natural convection boiling. As a consequence, the material would tend to slump into a slab geometry with a larger surface-to-volume ratio and a lesser thickness. For C reactor, in which vessel melt-through was calculated to occur in one hour, a maximum slab thickness of about one foot with a surface area of about 250 ft^2 could be sustained with boiling from the top surface. After about 15 hours, the volumetric heat generation would have decreased enough that the debris would freeze through its entire thickness. In P or K reactor, melt-through would be more rapid than for C; therefore, the maximum thickness that could be sustained would be less. Complete freezing is calculated to occur in about 6 hours. Penetration of a significant depth of concrete floor, which contains granite aggregate, would not be expected over these relatively short time intervals.

In an unmoderated geometry, criticality within the debris regardless of location would not occur. Even a homogenous sphere of core debris under H_2O would not be critical; however, a subcritical state cannot be guaranteed under all possible combinations of geometry, concentration, and moderation that could be postulated. Even though criticality is considered highly remote, such an occurrence would be unlikely to contribute significantly to either the energy of the system or the fission product inventory, but would serve to further disperse the debris within the confines of the below grade areas.

It is concluded that after about one day the remaining fission product inventory could be essentially confined within the reactor building and the filtration-adsorption compartments for an extended period of time with no further equipment operation or maintenance. Hence, the addition of further cooling would not be required and building overflow to the 50 million gallon retention basin would be unnecessary.

~~CONFIDENTIAL~~ ~~SECRET~~

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REFERENCES

1. W. S. Durant and R. J. Brown. *Analysis of Postulated Core Meltdown of an SRP Reactor.* SRL Report DPST-67-639 (1968) (Secret).

2. R. J. Brown. *Confinement Heat Removal System for the Savannah River Reactors.* SRL Report DPST-68-546 (1968) (Secret).

3. United States Steel Corp., Armco Steel Corp., and Lukens Steel Corp., personal communication (November 13, 1967).

4. P. K. Hilliard. *Oxidation of Uranium in Air at High Temperatures.* USAEC Report HW-58022, General Electric Co., Richland, Wash. (1958).

5. R. C. Milham and L. R. Jones. *Iodine and Noble Gas Retention Studies.* USAEC Report DP-1209, Savannah River Laboratory, E. I. du Pont de Nemours & Co., Aiken, S. C. (1969).

6. W. S. Durant, et al. *Activity Confinement System of the Savannah River Plant Reactors.* SRL Report DPST-66-300 (1966) (Secret).

7. W. S. Durant, et al. *Activity Confinement System of the Savannah River Plant Reactors.* USAEC Report DP-1071, Savannah River Laboratory, E. I. du Pont de Nemours & Co., Aiken, S. C. (1966).

8. R. C. Milham. *High Temperature Adsorbents for Iodine - Progress Report: January 1965-September 1966.* USAEC Report DP-1075, Savannah River Laboratory, E. I. du Pont de Nemours & Co., Aiken, S. C. (1966).

9. *Emergency Core Cooling.* Report of Advisory Task Force on Power Reactor Emergency Cooling. USAEC Division of Technical Information Extension, Oak Ridge, Tenn. (1966).

10. S. G. Lipsett. "Explosions from Molten Materials and Water." *Fire Technology 2(2)*, 118 (1966).

11. G. Long. "Explosions of Molten Aluminum in Water - Cause and Prevention." *Metal Progress 71(5)*, 107 (1957).

12. P. D. Hess and K. J. Brondyke. "Cause of Molten Aluminum-Water Explosions and Their Prevention." *Metal Progress 83(4)*, 93 (1969).

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13. F. E. Brauer, et al. "Metal/Water Explosions." *Nuclear Science and Engineering 31(3)*, 551 (1968).

14. *Kinetic Studies of Heterogeneous Water Reactors - 1964. Annual Summary Report - 1964.* USAEC Report STL-372-50. TRW Space Technology Laboratories, Inc., Redondo Beach, Calif. (1964).

15. *Chemical Engineering Division Semiannual Report.* USAEC Report ANL-6900, pp 270-280, Argonne National Laboratory, Argonne, Ill. (1964).

16. W. F. Zelezny. *Metal-Water Reactions: Rates of Reaction of Aluminum and Aluminum-Uranium Alloys with Water Vapor at Elevated Temperatures.* USAEC Report IDO-16629, Phillips Petroleum Company, Idaho Falls, Idaho (1960).

17. H. M. Higgins and R. D. Schultz. *The Reaction of Metals with Water and Oxidizing Gases at High Temperatures.* USAEC Report IDO-28000, Aerojet General Corp., Azusa, Calif. (1957).

18. R. O. Ivins and R. Koonz. "Study of the Combustion of Aluminum in Water." *Chemical Engineering Division Semiannual Report, July-December 1963.* USAEC Report ANL-6800, pp 338-341, Argonne National Laboratory, Argonne, Ill. (1964).

19. A. J. Scott. *Fission Product Release by the High Temperature Uranium-Steam Reaction.* USAEC Report HW-62603, General Electric Co., Richland, Wash. (1959).

20. R. K. Hilliard. *Oxidation of Uranium in Air at High Temperatures.* USAEC Report HW-58022, General Electric Co., Richland Wash. (1958).

21. R. E. Wilson and C. Barnes. "Isothermal Studies of the Uranium-Steam Reaction by the Volumetric Method." *Chemical Engineering Division Semiannual Report, July-December 1963.* USAEC Report ANL-6800, pp 330-335, Argonne National Laboratory, Argonne, Ill. (1964).

22. S. E. Jensen. *Effects of Metal-Water Reactions on the Loss-of-Coolant Accident.* USAEC Report IDO-17168, Phillips Petroleum Co., Idaho Falls, Idaho (1966).

23. L. F. Epstein. *Metal-Water Reactions: VII. Reactor Safety Aspects of Metal Water Reactions.* GEAP-3335, General Electric Company, Pleasanton, Calif. (1960).

24. J. G. Moore and E. V. Gilby. *A review of the Problems Arriving from the Combustion of Hydrogen in a Water Reactor Containment.* AHSB(S)R101, United Kingdom Atomic Energy Authority, London, England (1966).

25. Z. M. Shapiro and T. R. Moffette. *Hydrogen Flammability Data and Application to PWR Loss-of-Coolant Accident.* USAEC Report WAPD-SC-545, Westinghouse Electric Corp., Pittsburg, Pa. (1957).

26. J. G. Moore. "The Formation of Shock Waves from the Combustion of Gaseous Mixtures with Special Consideration of the Sodium-Air Reaction." *Nuclear Engineering and Design 3*, 320 (1966).

27. J. G. Moore and E. V. Gilby. *A Review of the Problems Arising from the Combustion of Hydrogen in a Water Reactor Containment.* AHSB(S)R101, United Kingdom Atomic Energy Authority, London, England (1966).

28. H. L. Brode. "The Numerical Solution of Spherical Blast Waves." *J. Applied Physics 26(6)*, 766 (1955).

29. S. Glasstone (ed). *The Effects of Nuclear Weapons.* U. S. Atomic Energy Commission (1962).

30. *Studies in Atomic Defense Engineering.* NAVDOCKS P-290 Department of the Navy, Bureau of Yards and Docks, Washington 25, D. C. (1962).

31. R. C. Gwaltney. *Missile Generation and Protection in Light-Water-Cooled Power Reactor Plants.* USAEC Report ORNL-NSIC-22, Oak Ridge National Laboratory, Oak Ridge, Tenn. (1968).

32. L. Baker. "Metal-Water Reactions." *Nuclear Safety 7(1)*, 75 (1965).

33. R. H. Cornwell to J. B. Tinker, J. A. List. *Containment Incident Analysis: Investigation of Main Tank Stresses.* Du Pont Engineering Department Report DPE-2162 (1962)(Secret).

34. J. F. Proctor. "Explosion Containment for Nuclear Reactor Vessels." *Nuclear Safety 7(4)*, 459 (1966).

35. R. W. Wise and J. F. Proctor. *Explosion Containment Laws for Nuclear Reactor Vessels.* NOLTR 63-140, United States Naval Ordinance Laboratory, White Oak, Maryland (1965).

36. A. N. Holden. *Physical Metallurgy of Uranium.* Addison-Wesley Publishing Co., Reading, Mass. (1958).

APPENDIX A

LAVA COMPUTER CODE

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* T I D Y *
     LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA

C     LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA
C
C       AN ANALYSIS OF TRANSIENT HEAT TRANSFER IN THE REACTOR BUILDING
C                     FOLLOWING A DESIGN BASIS ACCIDENT
C
C     *****      *****      *****      *****      *****      *****      *****
C
      DIMENSION QFU001(999),QRAD(999),QCONV2(999),AFLO(999),QFLOR1(999),
     1QESCAP(999)
      READ (5,330) TFUEL1,WFUEL,CFUEL,TSYS,AKMET,AMET,WMET,TMET1,XMET,TA
     1IR1,TWALL1,EPS1,EPS2,WWALL,CWALL,XWALL,AWALL,AKWALL,CAIR,WAIR,CMET
     2,QCONV1,TSYS1,TSYS2,TMET4,TWALL0,TWALL3,COF,PO,AKFUEL,TAIR2,TAIR4,
     3FAT,AFLOOR,EPS3,ZAP,ABSFU,ABSMET,DENFU,DENMET,HF,HFMET,TFUELM
C
      READ (5,340) NQFU
      READ (5,350) (QFU001(I),I=1,NQFU)
      READ (5,360) QRAD(1)
      READ (5,370) QCONV2(1)
      WRITE (6,380)
      WRITE (6,390)
      WRITE (6,400)
      WRITE (6,410) TFUELM
      WRITE (6,420) WAIR
      WRITE (6,430) TSYS
      WRITE (6,440) WFUEL
      WRITE (6,450) WMET
      WRITE (6,460) AMET
      WRITE (6,470) EPS1
      WRITE (6,480) EPS2
      WRITE (6,490) AWALL
      WRITE (6,500) AKWALL
      WRITE (6,510) COF
      WRITE (6,520) PO
      WRITE (6,530)
      WRITE (6,540)
C
C         CALCULATE ENERGY STORED IN FUEL AS FUSION HEAT
      HFUS=HF*WFUEL
C
C         CALCULATE HEAT IN FUEL AT TIME FUEL MELTING BEGINS
      QFUEL=WFUEL*CFUEL*(TFUEL1-TSYS)
C
C         ENTER MASTER DO LOOP FOR FUEL CONTAINED IN PIPING. UNLESS
C         OTHERWISE NOTED CALCULATIONS ARE FOR ONE MINUTE INCREMENTS.
C         TIME 0 INDICATES START OF INCREMENT, 1/2 INDICATES MIDDLE
C         OF INCREMENT, AND 1 INDICATES END OF INCREMENT
      DO 160  I=1,NQFU
      NTIME=I
C
C         INITIALIZE SUBROUTINES FOR TRANSIENT CONDUCTION IN CONCRETE
```

```
*TIDY*
      LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA
      CALL COND(QWALL1,TWALL1,NTIME,0)
      CALL COND1(QFLORA,TLAVA5,NTIME,0)
      CALL BOND(QWALL1,TWALL1,NTIME,0)
C
C        CALCULATE HEAT IN FUEL AT TIME 1 WITH NO HEAT LOSS, EQUATE
C        HEAT AT TIME 0 TO TIME 1
      QFUEL2=QFUEL+(QFU001(I)*PO)
      QFUEL=QFUEL2
C
C        CALCULATE GAMMA ESCAPE FROM FUEL CONTAINED IN EFFLUENT PIPING
      AMU=20.3*ABSFU
      CALL GETOUT(AMU,PCC)
      TOPIPE=QFU001(I)*PO*(1.0-PCC)*0.5
C
C        CALCULATE GAMMA ATTENUATION BY EFFLUENT PIPING
      GONE=TOPIPE/EXP(0.57*XMET*61.0)
      SOAKED=TOPIPE-GONE
C
C                  LOOP A
C
C        CALCULATE AVERAGE FUEL TEMPERATURE AT TIME 1 WITH NO HEAT LOSS
100   TFUEL2=(QFUEL/(WFUEL*CFUEL))+TSYS
      IF (TFUEL2.GE.TFUELM) GO TO 110
C
C        CALCULATE ENERGY STORED AS HEAT OF FUSION, THEN RECALCULATE
C        FUEL TEMPERATURE AT TIME 1
      QAVAIL=WFUEL*CFUEL*(TFUEL2-TSYS)+HFUS
      TFUEL2=(QAVAIL/(WFUEL*CFUEL))+TSYS
      IF (TFUEL2.LT.TFUELM) GO TO 110
      QSENS=WFUEL*CFUEL*(TFUELM-TSYS)
      QFUSE=QAVAIL-QSENS
      IF (QFUSE.LE.HFUS) TFUEL2=TFUELM
      IF (QFUSE.LE.HFUS) GO TO 110
      HLEFT=QFUSE-HFUS
      TFUEL2=(HLEFT/(WFUEL*CFUEL))+TFUEL2
C
C        CALCULATE FUEL-PIPE INTERFACE TEMPERATURE
110   DELT=QFUEL*7.5E-6
      TFUEL4=TFUEL2-(DELT/2.0)
C
C        CALCULATE HEAT STORED IN PIPE WITH HEAT LOSS DURING 1 TIME
C        INCREMENT, AVERAGE PIPE TEMPERATURES AT TIMES 1 AND 0
      QMET1=(((AKMET*(((TFUEL1+TFUEL4)/2.0)-TMET1))*AMET/XMET)/60.0)-QCO
     1NV1-QRAD(I)+SOAKED
      TMET2=(QMET1/(WMET*CMET))+TSYS1
      TMET3=(TMET2+TSYS1)/2.0
C
C        CALCULATE HEAT IN FUEL AND FUEL TEMPERATURE AT TIME 1 WITH
C        HEAT LOSS
      QFUEL3=QFUEL2-QMET1-QCONV1-QRAD(I)-GONE
```

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* T I D Y *
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      TFUEL3=(QFUEL3/(WFUEL*CFUEL))+TSYS
      IF (TFUEL3.GE.TFUELM) GO TO 120
C
C         CALCULATE ENERGY STORED AS HEAT OF FUSION, THEN RECALCULATE
C         FUEL TEMPERATURE AT TIME 1
      QAVAIL=WFUEL*CFUEL*(TFUEL3-TSYS)+HFUS
      TFUEL3=(QAVAIL/(WFUEL*CFUEL))+TSYS
      IF (TFUEL3.LT.TFUELM) GO TO 120
      QSENS=WFUEL*CFUEL*(TFUELM-TSYS)
      QFUSE=QAVAIL-QSENS
      IF (QFUSE.LE.HFUS) TFUEL3=TFUELM
      IF (TFUEL4.GT.2020.0) GO TO 170
      IF (QFUSE.LE.HFUS) GO TO 120
      HLEFT=QFUSE-HFUS
      TFUEL3=(HLEFT/(WFUEL*CFUEL))+TFUEL3
120   TMET1=TMET3
      QFUEL=(QFUEL3+QFUEL)/2.0
C         TEST FOR GOODNESS OF FUEL TEMPERATURE AT TIME 1. ITERATE TO
C         LOOP A BASED ON RECALCULATED HEAT IN FUEL IF NECESSARY
      IF (ABS(TFUEL3-TFUEL2).GT.0.5) GO TO 130
C
C         CALCULATE SURFACE TEMPERATURE OF PIPE AND AVERAGE TEMPERATURE
C         OF AIR AT TIME 1/2
      TSURF=TMET3-(((TFUEL1+TFUEL4)/2.0)-TMET3)
      IF (TSURF.LT.TSYS) TSURF=TSYS
      TAIR2=(TAIR2+TAIR4)/2.0
C
C         CALCULATE HEAT LOST FROM PIPE BY NATURAL CONVECTION
      XX=TSURF-TAIR2
      QCONV1=(COF*(ABS(XX))**0.333*AMET*(XX))/60.0
C
C         CALCULATE HEAT LOST FROM PIPE BY THERMAL RADIATION TO WALLS
C         AND FLOOR
      RR=(TSURF+460.0)**4.0
      RS=(TWALL1+460.0)**4.0
      QRAD(I)=1.73E-9*EPS1*AMET*(RR-RS)/60.0
C
C         CALCULATE HEAT CONTENT AND AVERAGE TEMPERATURE OF PIPE
C         CONSIDERING CONDUCTION FROM FUEL, GAMMA ATENUATION, CONVECTION
C         TO AIR AND THERMAL RADIATION TO WALLS AND FLOOR AT TIME 1
      QMET2=((AKMET*AMET*(((TFUEL1+TFUEL4)/2.0)-TMET1)/XMET)/60.)-QCONV1
     1-QRAD(I)+SOAKED
      QMET2=(QMET2+QMET1)/2.0
      TMET5=(QMET2/(WMET*CMET))+TSYS1
      TMET1=(TMET5+TSYS1)/2.0
C
C         CALCULATE HEAT CONTENT OF FUEL CONSIDERING HEAT LOST TO PIPE
C         AND ENVIRONMENT AT TIME 1
      QFUEL4=QFUEL2-QMET2-QCONV1-QRAD(I)-GONE
      QFUEL=(QFUEL4+QFUEL)/2
```

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C
C       TEST FOR GOODNESS OF PIPE TEMPERATURE AT TIME 1. ITERATE TO
C       LOOP A BASED ON RECALCULATED HEAT IN FUEL IF NECESSARY
      IF (ABS(TMET2-TMET5).GT.2.0) GO TO 100
C
C       SUM HEAT TRANSFERRED TO WALLS AND FLOOR BY THERMAL RADIATION
      RSUM=0.0
      DO 130 J=1,NTIME
      RSUM=RSUM+QRAD(J)
130   CONTINUE
C
C       SUM HEAT TRANSFERRED FROM WALLS AND FLOOR BY NATURAL
C       CONVECTION
      CSUM=0.0
      DO 140 J=1,NTIME
      CSUM=CSUM+QCONV2(J)
140   CONTINUE
C
C       CALCULATE HEAT CONTENT OF WALLS AND FLOOR BASED ON THERMAL
C       RADIATION TO WALLS AND FLOOR, NATURAL CONVECTION FROM WALLS
C       AND FLOOR, AND GAMMA RADIATION TO WALLS AND FLOOR AT TIME 1
      QWALL1=RSUM-CSUM+GONE
C
C       CALCULATE SURFACE TEMPERATURE OF WALLS AND FLOOR FROM
C       TRANSIENT CONDUCTION SUBROUTINE AT TIME 1
      L=1
      CALL COND(QWALL1,TWALLA,NTIME,L)
      TWALL2=TWALLA
C
C       CALCULATE HEAT TRANSFERRED FROM WALLS AND FLOOR BY NATURAL
C       CONVECTION
      QCONV2(I)=(COF*(ABS(((((TWALL2+TWALL3)/2)+TWALL0)/2)-TAIR2))**0.33
     13*AWALL*(((((TWALL2+TWALL3)/2)+TWALL0)/2)-TAIR2))/60.0
C
C       SUM HEAT TRANSFERRED FROM WALLS AND FLOOR BY NATURAL
C       CONVECTION
      CSUM=0.0
      DO 150 J=1,NTIME
      CSUM=CSUM+QCONV2(J)
150   CONTINUE
C
C       CALCULATE HEAT CONTENT OF WALLS AND FLOOR BASED ON THERMAL
C       RADIATION TO WALLS AND FLOOR, NATURAL CONVECTION FROM WALLS
C       AND FLOOR, AND GAMMA RADIATION TO WALLS AND FLOOR AT TIME 1
      QWALL2=RSUM-CSUM+GONE
C
C       CALCULATE SURFACE TEMPERATURE OF WALLS AND FLOOR FROM
C       TRANSIENT CONDUCTION SUBROUTINE AT TIME 1
      L=1
      CALL COND(QWALL2,TWALLC,NTIME,L)
```

[redacted]

```
* T I D Y *
      LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA

      TWALL3=TWALLC
      TWALL1=(((TWALL2+TWALL3)/2)+TWALL1)/2
C
C         TEST FOR GOODNESS OF WALL AND FLOOR TEMPERATURE AT TIME 1.
C         ITERATE TO LOOP A IF NECESSARY
      IF (TWALL3-TWALL2.GT.2.0.OR.TWALL2-TWALL3.GT.2.0) GO TO 100
C
C         CALCULATE HEAT CONTENT AND TEMPERATURE OF AIR AT TIME 1
      QAIR2=QCONV1+QCONV2(I)
C
      TAIR3=(QAIR2/(WAIR*CAIR))+TSYS
      IF (TAIR3.LT.100.0) TAIR3=100.0
      QAIR2=WAIR*CAIR*(TAIR3-TSYS)
C
      TAIR4=(TAIR3+TAIR1)/2.0
C
C         TEST FOR GOODNESS OF AIR TEMPERATURE AT TIME 1/2. ITERATE
C         TO LOOP A IF NECESSARY
      IF (TAIR4-TAIR2.GT.2.0.OR.TAIR2-TAIR4.GT.2.0) GO TO 100
      WRITE (6,550) NTIME,TFUEL3,TMET3,TWALL3,TAIR3,TSURF,TFUEL4,QFU001(
     1I)
C
C         RESTATE VARIABLES FOR SECOND TIME INCREMENT
      TFUEL1=TFUEL4
      TAIR1=TAIR3
      TSYS1=TMET5
      TMET4=TMET3
      TWALL0=TWALL3
      TWALL1=TWALL3
      TSYS2=TWALL3
      TMET1=TMET3
C
C         IF THE TEMPERATURE OF THE FUEL AT THE FUEL-PIPE INTERFACE
C         EXCEEDS THE MELTING POINT OF THE FUEL, THE PIPE IS ASSUMED TO
C         COLLAPSE BECAUSE OF INTERACTION OF URANIUM AND STAINLESS
C         STEEL
      IF (TFUEL4.GT.TFUELM) GO TO 170
      QRAD(I+1)=QRAD(I)
      QCONV2(I+1)=QCONV2(I)
160   CONTINUE
C
C     AT THIS CONDITION THE PIPES COLLAPSE
C
170   WRITE (6,560)
      WRITE (6,570)
      WRITE (6,580)
C
C         CALCULATE HEAT IN LAVA AT TIME OF PIPE COLLAPSE
      QLAVA=WFUEL*CFUEL*(TFUEL3-TSYS)+WMET*CMET*(TMET3-TSYS)
C
```

```
* T I D Y *
      LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA
C        CALCULATE MASS OF STEEL IN LAVA
      WMAT=((QLAVA/(ZAP-TSYS))-WFUEL*CFUEL)/CMET
      IF (WMAT.GE.WMET) GO TO 180
      WMET=WMAT
C
180   NTIME=NTIME+1
      KTIME=NTIME
      WRITE (6,590) AFLOOR
      WRITE (6,600) ZAP
      WRITE (6,610) ABSFU
      WRITE (6,620) ABSMET
      WRITE (6,630) DENFU
      WRITE (6,640) DENMET
      WRITE (6,650) WMET
      WRITE (6,660) EPS3
C
C        WALLS AND FLOOR MUST NOW BE TREATED SEPARATELY. ADJUST FOR
C        AREA DIFFERENCE
      RSUMPB=RSUM*((AWALL-AFLOOR)/AWALL)
      CSUMPB=CSUM*((AWALL-AFLOOR)/AWALL)
C
      WRITE (6,670)
      WRITE (6,680)
C
      K=NTIME
      NNN=1
      AFLO(K-1)=0.0
      QFLOR1(K-1)=0.0
C
C        COMBINE THERMAL RADIATION CONSTANTS
      BR=1.0/((1.0/EPS3)+(1.0/EPS2)-1.0)
C
C        CALCULATE VOLUME OF LAVA
      VOL=(WFUEL/DENFU)+(WMET/DENMET)
C
C        CALCULATE GAMMA ABSORPTION COEFFICIENT OF LAVA
      ABSMIX=(((WFUEL/DENFU)/VOL)*ABSFU)+(((WMET/DENMET)/VOL)*ABSMET)
C
C        CALCULATE HEAT OF FUSION OF LAVA
      HFMIX=((WFUEL/DENFU)/VOL)*HF+(((WMET/DENMET)/VOL)*HFMET)
      HFUSL=HFMIX*(WFUEL+WMET)
C
C        ENTER MASTER DO LOOP FOR LAVA ON FLOOR OF PUMP ROOMS
      DO 320 K=NTIME,NQFU
      LTIME=K
C
C        INITIALIZE SUBROUTINES FOR TRANSIENT CONDUCTION INTO CONCRETE
C        WALLS AND FLOOR TO TIME OF PIPE COLLAPSE
      CALL COND1(QFLORA,TLAVA5,LTIME,0)
      CALL BOND(QWALL1,TWALL1,LTIME,0)
```

```
* T I O Y *
     LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA

      KKK=NNN
      IF (KKK.GT.1) AFLO(K)=AFLO(K-1)
      IF (KKK.GT.1) GO TO 190
      AFLO(K)=AFLOOR
C
C         CALCULATE TEMPERATURE OF LAVA AT TIME 0
190   TLAVA=(QLAVA/((WFUEL*CFUEL)+(WMET*CMET)))+TSYS
      IF (TLAVA.GE.ZAP) GO TO 200
C
C         CALCULATE ENERGY STORED AS HEAT OF FUSION, THEN RECALCULATE
C         LAVA TEMPERATURE AT TIME 0
      QAVAIL=(WFUEL*CFUEL+WMET*CMET)*(TLAVA-TSYS)+HFUSL
      TLAVA=(QAVAIL/(WFUEL*CFUEL+WMET*CMET))+TSYS
      IF (TLAVA.LT.ZAP) GO TO 200
      QSENS=(WFUEL*CFUEL+WMET*CMET)*(TLAVA-TSYS)
      QFUSE=QAVAIL-QSENS
      IF (QFUSE.LE.HFUSL) TLAVA=ZAP
      IF (QFUSE.LE.HFUSL) GO TO 200
      HLEFT=QFUSE-HFUSL
      TLAVA=(HLEFT/(WFUEL*CFUEL+WMET*CMET))+TLAVA
200   QRAD(K)=0.0
      QFLOR2=0.0
      QOXIDE=0.0
      QCONV1=0.0
      TLAVA4=0.0
      KKKK=0
C
C         CALCULATE THICKNESS OF LAVA ON FLOOR ASSUMING FLOOR COMPLETELY
C         COVERED
      THICK=((WFUEL/DENFU)+(WMET/DENMET))/AFLO(K)
C
C         CALCULATE GAMMA ESCAPE FROM LAVA
      BMU=(THICK*30.48/2.0)*ABSMIX
      CALL ESCAPE(BMU,PC)
      RETAIN=1.0-((1.0-PC)/2.0)
C
C         CALCULATE HEAT IN FUEL WITH CONSIDERATION TO HEAT LOSSES AT
C         TIME 1
      QLAVA2=QLAVA+QFU001(K)*PO*RETAIN-QRAD(K)-QCONV1-QFLOR2+QOXIDE
      QLAVA=QLAVA2
C                  LOOP B
210   NNN=KKK+1
C
C         CALCULATE TEMPERATURE OF LAVA AT TIME 1
      TLAVA2=(QLAVA/((WFUEL*CFUEL)+(WMET*CMET)))+TSYS
      IF (TLAVA2.GE.ZAP) GO TO 220
C
C         CALCULATE ENERGY STORED AS HEAT OF FUSION, THEN RECALCULATE
C         LAVA TEMPERATURE AT TIME 1
      QAVAIL=(WFUEL*CFUEL+WMET*CMET)*(TLAVA2-TSYS)+HFUSL
```

[redacted]

```
*T I D Y *
  LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA

      TLAVA2=(QAVAIL/(WFUEL*CFUEL+WMET*CMET))+TSYS
      IF (TLAVA2.LT.ZAP) GO TO 220
      QSENS=(WFUEL*CFUEL+WMET*CMET)*(TLAVA2-TSYS)
      QFUSE=QAVAIL-QSENS
      IF (QFUSE.LE.HFUSL) TLAVA2=ZAP
      IF (QFUSE.LE.HFUSL) GO TO 220
      HLEFT=QFUSE-HFUSL
      TLAVA2=(HLEFT/(WFUEL*CFUEL+WMET*CMET))+TLAVA2
C
C          ESTIMATE FLOOR AREA COVERED BY LAVA
220   AFLO(K)=AFLOOR*(1.0-0.002*KKKK)
      IF (AFLO(K).LT.AFLO(K-1)) AFLO(K)=AFLO(K-1)
      IF (KKKK.EQ.0) GO TO 230
      IF (TLAVA4.GT.ZAP) GO TO 240
C
C          CALCULATE SURFACE TEMPERATURE OF LAVA FROM AVERAGE TEMPERATURE
C          AND GRADIENT THROUGH THICKNESS AT TIME 1
230   DELT1=QFU001(K)*PO*60.0*RETAIN*((WFUEL/DENFU)+(WMET/DENMET))/(2.0*
     1AFLO(K))**2.0/(AKFUEL+AKMET)
240   TLAVA4=(TLAVA2+TLAVA)/2.0
      TLAVA5=TLAVA4-(DELT1/2.0)
      TAIR2=(TAIR2+TAIR4)/2.0
      YY=TLAVA5-TAIR2
C
C          CALCULATE HEAT TRANSFERRED FROM LAVA BY NATURAL CONVECTION
      QCONV1=(COF*(ABS(YY))**0.333*AFLO(K)*YY)/60.0
      TLAVA5=(TLAVA5-32.0)/1.8
C
C          CALCULATE HEAT GENERATION IN LAVA FROM SURFACE OXIDATION
      TOX1=TLAVA5
      TOX2=TLAVA5
      IF (TOX1.LT.1000.0) TOX1=1000.0
      IF (TOX2.GT.1200.0) TOX2=1200.0
      C1=EXP(0.00346*TOX1-0.79)
      C2=(0.00218*TOX2-1.0)*(AFLO(K)/WFUEL)*2.04
      QOXIDE=(AFLO(K)/2.0)*C1*(2.71**C2)*1.8
      TLAVA5=TLAVA5*1.8+32.0
C
C          CALCULATE HEAT TRANSFERRED FROM LAVA BY THERMAL RADIATION
      TR=(TLAVA5+460.0)**4.0
      TS=(TWALL1+460.0)**4.0
      QRAD(K)=1.73E-9*BR*AFLO(K)*(TR-TS)/60.0
C
C          CALCULATE ENERGY TRANSFERRED FROM LAVA TO FLOOR BY GAMMA
C          ESCAPE
      QESCAP(K)=(QFU001(K)*PO*(1.0-RETAIN))/2.0
C
C          CALCULATE HEAT TRANSFERRED INTO FLOOR FROM TRANSIENT
C          CONDUCTION SUBROUTINE
      L=2
```

[redacted]

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*TIDY*
      LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA

      CALL COND1(QFLORA,TLAVA5,LTIME,L)
      QFLOR1(K)=QFLORA
      FAT=AFLO(K)/AFLOOR
      QFLOR2=(QFLOR1(K)-QFLOR1(K-1))*FAT+QESCAP(K)
      IF (KKK.EQ.1) GO TO 260
C
C         CALCULATE HEAT TRANSFERRED INTO FLOOR DURING ONE TIME
C         INCREMENT
      N=K-KTIME+1
      QFLOR2=0.0
      DO 250 M=1,N
      L8=K-M
      L7=K-N+M
      QFLOR2=QFLOR2+(QFLOR1(L8+1)-QFLOR1(L8))*((AFLO(L7)-AFLO(L7-1))/AFL
     1OOR)
  250 CONTINUE
C
C         CALCULATE HEAT CONTENT AND TEMPERATURE AT TIME 1 OF LAVA
C         CONSIDERING HEAT LOSSES AND GAINS
  260 QLAVA3=QLAVA2-QCONV1-QRAD(K)-QFLOR2+QOXIDE+QESCAP(K)
      TLAVA3=(QLAVA3/((WFUEL*CFUEL)+(WMET*CMET)))+TSYS
      IF (TLAVA3.GE.ZAP) GO TO 270
C
C         CALCULATE ENERGY STORED AS HEAT OF FUSION, THEN RECALCULATE
C         LAVA TEMPERATURE
      QAVAIL=(WFUEL*CFUEL+WMET*CMET)*(TLAVA3-TSYS)+HFUSL
      TLAVA3=(QAVAIL/(WFUEL*CFUEL+WMET*CMET))+TSYS
      IF (TLAVA3.LT.ZAP) GO TO 270
      QSENS=(WFUEL*CFUEL+WMET*CMET)*(TLAVA3-TSYS)
      QFUSE=QAVAIL-QSENS
      IF (QFUSE.LE.HFUSL) TLAVA3=ZAP
      IF (QFUSE.LE.HFUSL) GO TO 270
      HLEFT=QFUSE-HFUSL
      TLAVA3=(HLEFT/(WFUEL*CFUEL+WMET*CMET))+TLAVA3
  270 TLAVA6=TLAVA3-(DELT1/2.0)
      IF (AFLO(K).GT.AFLOOR) AFLO(K)=AFLOOR
      QLAVA=QLAVA3
C
C         TEST FOR GOODNESS OF LAVA TEMPERATURE AT TIME 1. ITERATE TO
C         LOOP 8 IF NECESSARY
      IF (ABS(TLAVA3-TLAVA2).GT.2.0) GO TO 210
      IF (TLAVA6.LT.ZAP) KKKK=KKKK+1
      IF (AFLO(K).EQ.AFLO(K-1)) GO TO 280
      IF (TLAVA5.LT.ZAP) GO TO 210
C
C         SUM HEAT TRANSFERRED TO WALLS BY THERMAL RADIATION AND GAMMA
C         ESCAPE
  280 RSUM=RSUMP8
      DO 290 J=KTIME,K
      RSUM=RSUM+QRAD(J)+QESCAP(J)
```

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*T I D Y *
 .LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA
90    CONTINUE

          SUM HEAT TRANSFERRED FROM WALLS BY NATURAL CONVECTION
      CSUM=CSUMPB
      DO 300 J=KTIME,K
      CSUM=CSUM+QCONV2(J)
00    CONTINUE

          CALCULATE HEAT CONTENT AND SURFACE TEMPERATURE OF WALLS AT
          TIME 1
      QWALL1=RSUM-CSUM
      L=1
      CALL BOND(QWALL1,TWALLA,LTIME,L)
      TWALL2=TWALLA

          CALCULATE HEAT TRANSFERRED FROM WALLS BY NATURAL CONVECTION
      QCONV2(K)=(COF*(ABS(((((TWALL2+TWALL3)/2.0)+TWALLO)/2.0)-TAIR2))**
     10.333*(AWALL-AFLO(K))*(((((TWALL2+TWALL3)/2.0)+TWALLO)/2.0)-TAIR2)
     2)/60.0

          SUM HEAT TRANSFERRED FROM WALLS BY NATURAL CONVECTION
      CSUM=CSUMPB
      DO 310 J=KTIME,K
      CSUM=CSUM+QCONV2(J)
10    CONTINUE

          CALCULATE HEAT CONTENT AND SURFACE TEMPERATURE OF WALLS AT
          TIME 1/2
      QWALL2=RSUM-CSUM
      L=1
      CALL BOND(QWALL2,TWALLC,LTIME,L)
      TWALL3=TWALLC
      TWALL1=(((TWALL2+TWALL3)/2)+TWALL1)/2.0

          TEST FOR GOODNESS OF WALL SURFACE TEMPERATURE AT TIME 1.
          ITERATE TO LOOP B IF NECESSARY
      IF (ABS(TWALL3-TWALL2).GT.2.0) GO TO 210

          CALCULATE HEAT CONTENT AND TEMPERATURE OF AIR AT TIME 1
      QAIR2=QCONV1+QCONV2(K)
      TAIR3=(QAIR2/(WAIR*CAIR))+TSYS
      IF (TAIR3.LT.100.0) TAIR3=100.0
      QAIR2=WAIR*CAIR*(TAIR3-TSYS)
      TAIR4=(TAIR3+TAIR1)/2.0

          TEST FOR GOODNESS OF AIR TEMPERATURE AT TIME 1/2. ITERATE TO
          LOOP B IF NECESSARY
      IF (ABS(TAIR4-TAIR2).GT.2.0) GO TO 210

      WRITE (6,690) LTIME,TLAVA3,TLAVA6,AFLO(K),TWALL3,TAIR3,QFU001(K),Q
```

[redacted]

```
     1OXIDE,QCONV1,QFLOR2,QRAD(K)
C
C          RESTATE VARIABLES FOR SECOND TIME INCREMENT
      TWALL0=TWALL3
      TWALL1=TWALL3
      QRAD(K+1)=QRAD(K)
      QCONV2(K+1)=QCONV2(K)
      TAIR1=TAIR3
320   CONTINUE
      STOP
C
330   FORMAT (7F10.5)
340   FORMAT (I3)
350   FORMAT (7F10.5)
360   FORMAT (F10.5)
370   FORMAT (F10.5)
380   FORMAT (1H1)
390   FORMAT (45X,'LAVA - OR THE CREEPING HEAP'///)
400   FORMAT (25X,'INPUT INFORMATION'//)
410   FORMAT (5X,'MELTING TEMPERATURE OF FUEL',3X,F10.1/)
420   FORMAT (5X,'MASS OF AIR FROM -40',24X,F10.1/)
430   FORMAT (5X,'TEMPERATURE OF AMBIENT SYSTEM',14X,F10.1/)
440   FORMAT (5X,'EFFECTIVE MASS OF CORE',22X,F10.1/)
450   FORMAT (5X,'EFFECTIVE MASS OF EFFLUENT PIPING AND PUMPS',1X,F10.1/
     1)
460   FORMAT (5X,'EFFECTIVE AREA OF EFFLUENT PIPING AND PUMPS',1X,F10.1/
     1)
470   FORMAT (5X,'EMISSIVITY OF EFFLUENT PIPING AND PUMPS',5X,F10.1/)
480   FORMAT (5X,'EMISSIVITY OF CONCRETE WALLS AT -40',9X,F10.1/)
490   FORMAT (5X,'EFFECTIVE SURFACE AREA OF CONCRETE WALLS',4X,F10.1/)
500   FORMAT (5X,'THERMAL CONDUCTIVITY OF CONCRETE',12X,F10.1/)
510   FORMAT (5X,'COEFFICIENT FOR CONVECTION EQUATION',9X,F10.1/)
520   FORMAT (5X,'RELATIVE HEAT GENERATION',20X,F10.1///)
530   FORMAT (25X,'OUTPUT INFORMATION'//)
540   FORMAT (5X,'TIME      FUEL TEMP      AVG PIPE TEMP      CONCRETE WALL
     1 TEMP      AIR TEMP      PIPE SURFACE TEMP      INTERFACE TEMP      HE
     2AT IN'//)
550   FORMAT (3X,I3,5X,F10.1,5X,F10.1,11X,F10.1,8X,F10.1,7X,F10.1,12X,F1
     10.1,4X,F10.1/)
560   FORMAT (1H-)
570   FORMAT (2X,'AT THIS CONDITION THE PIPE COLLAPSES'///)
580   FORMAT (25X,'INPUT INFORMATION'//)
590   FORMAT (5X,'FLOOR AREA AT -40',27X,F10.1/)
600   FORMAT (5X,'MELTING TEMPERATURE OF LAVA',17X,F10.1/)
610   FORMAT (5X,'LINEAR ABSORPTION COEF OF FUEL',14X,F10.5/)
620   FORMAT (5X,'LINEAR ABSORPTION COEF OF STAINLESS STEEL',3X,F10.5/)
630   FORMAT (5X,'DENSITY OF FUEL',29X,F10.5/)
640   FORMAT (5X,'DENSITY OF STAINLESS STEEL',18X,F10.5/)
650   FORMAT (5X,'MASS OF STEEL IN LAVA',23X,F10.1/)
660   FORMAT (5X,'EMISSIVITY OF DEBRIS',24X,F10.1///)
```

```
T I D Y *
   LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA LAVA

70    FORMAT (25X,'OUTPUT INFORMATION'//)
80    FORMAT (3X,'TIME  AVG LAVA TEMP  LAVA SURF TEMP  LAVA AREA  WALL T
     1EMP  AIR TEMP     HEAT IN    HEAT OXID   HEAT CONV   HEAT COND   HE
     2AT RAD'//)
90    FORMAT (3X,I4,5X,F10.1,6X,F10.1,F11.1,F11.1,F10.1,2X,F10.1,2X,F10.
     11,2X,F10.1,2X,F10.1,F11.1/)
      END
```

* T I D Y *

UNSTEADY STATE HEAT TRANSFER INTO WALLS AND FLOOR

```
C        UNSTEADY STATE HEAT TRANSFER INTO WALLS AND FLOOR
C
      SUBROUTINE COND(QWALL1,TWALL1,NTIME,L)
C
      DIMENSION T(2,100,10),Q(10)
C
C
      NTEMP=10
      NLAY=100
      WWALL=110000.0
      CWALL=0.25
      AM=2.0
      IF (L.GT.0) GO TO 220
      IF (NTIME.GT.1) GO TO 130
      DO 110 K=1,NTEMP
      DO 100 J=1,NLAY
100   T(1,J,K)=100.0
110   CONTINUE
C
      DO 120 K=1,NTEMP
      T(2,100,K)=100.0
120   T(2,1,K)=100.0+100.0*(K-1)
      GO TO 160
130   DO 140 K=1,NTEMP
      DO 140 J=1,NLAY
      T(1,J,K)=T(2,J,K)
140   T(2,J,K)=0.0
C
      DO 150 K=1,NTEMP
      T(2,1,K)=T(1,1,K)
150   T(2,100,K)=100.0
160   NLAY2=NLAY-1
      DO 180 K=1,NTEMP
      DO 170 J=2,NLAY2
170   T(2,J,K)=(T(1,J-1,K)+T(1,J+1,K)+(T(1,J,K)*(AM-2.0)))/AM
180   CONTINUE
C
      DO 190 K=1,NTEMP
      Q(K)=0.0
190   CONTINUE
C
      DO 210 K=1,NTEMP
      DO 200 J=1,NLAY2
200   Q(K)=Q(K)+WWALL*CWALL*(T(2,J,K)-100.0)
210   CONTINUE
      RETURN
C
220   GO TO (230,270),L
230   IF (QWALL1.LE.Q(1)) GO TO 260
      DO 240 K=2,10
```

[redacted]

TIDY

UNSTEADY STATE HEAT TRANSFER INTO WALLS AND FLOOR

```
      IF (QWALL1.LE.Q(K)) GO TO 250
 40   CONTINUE
      TWALL1=T(2,1,10)
      RETURN
 50   TWALL1=(T(2,1,K)-T(2,1,K-1))*(QWALL1-Q(K-1))/(Q(K)-Q(K-1))+T(2,1,K
     1-1)
      RETURN
 60   TWALL1=T(2,1,1)
      RETURN

 70   IF (TWALL1.LE.T(2,1,1)) GO TO 300
      DO 280 K=2,10
      IF (TWALL1.LE.T(2,1,K)) GO TO 290
 80   CONTINUE
      QWALL1=Q(10)
      RETURN
 90   QWALL1=(Q(K)-Q(K-1))*(TWALL1-T(2,1,K-1))/(T(2,1,K)-T(2,1,K-1))+Q(K
     1-1)
      RETURN
 00   QWALL1=Q(1)
      RETURN
      END
```

* T I D Y *

UNSTEADY STATE HEAT TRANSFER INTO WALLS

```
C        UNSTEADY STATE HEAT TRANSFER INTO WALLS
C
      SUBROUTINE COND1(QWALL1,TWALL1,NTIME,L)
C
      DIMENSION T(2,100,30),Q(30)
C
C
      NTEMP=30
      NLAY=100
      WWALL=18200.0
      CWALL=0.25
      AM=2.0
      IF (L.GT.0) GO TO 220
      IF (NTIME.GT.1) GO TO 130
      DO 110 K=1,NTEMP
      DO 100 J=1,NLAY
100   T(1,J,K)=100.0
110   CONTINUE
C
      DO 120 K=1,NTEMP
      T(2,100,K)=100.0
120   T(2,1,K)=100.0+100.0*(K-1)
      GO TO 160
130   DO 140 K=1,NTEMP
      DO 140 J=1,NLAY
      T(1,J,K)=T(2,J,K)
140   T(2,J,K)=0.0
C
      DO 150 K=1,NTEMP
      T(2,1,K)=T(1,1,K)
150   T(2,100,K)=100.0
160   NLAY2=NLAY-1
      DO 180 K=1,NTEMP
      DO 170 J=2,NLAY2
170   T(2,J,K)=(T(1,J-1,K)+T(1,J+1,K)+(T(1,J,K)*(AM-2.0)))/AM
180   CONTINUE
C
      DO 190 K=1,NTEMP
      Q(K)=0.0
190   CONTINUE
C
      DO 210 K=1,NTEMP
      DO 200 J=1,NLAY2
200   Q(K)=Q(K)+WWALL*CWALL*(T(2,J,K)-100.0)
210   CONTINUE
      RETURN
C
220   GO TO (230,270),L
230   IF (QWALL1.LE.Q(1)) GO TO 260
      DO 240 K=2,30
```

* T I D Y *

UNSTEADY STATE HEAT TRANSFER INTO WALLS

```
      IF (QWALL1.LE.Q(K)) GO TO 250
240   CONTINUE
      TWALL1=T(2,1,30)
      RETURN
250   TWALL1=(T(2,1,K)-T(2,1,K-1))*(QWALL1-Q(K-1))/(Q(K)-Q(K-1))+T(2,1,K
     1-1)
      RETURN
260   TWALL1=T(2,1,1)
      RETURN

270   IF (TWALL1.LE.T(2,1,1)) GO TO 300
      DO 280 K=2,30
      IF (TWALL1.LE.T(2,1,K)) GO TO 290
280   CONTINUE
      QWALL1=Q(30)
      RETURN
290   QWALL1=(Q(K)-Q(K-1))*(TWALL1-T(2,1,K-1))/(T(2,1,K)-T(2,1,K-1))+Q(K
     1-1)
      RETURN
300   QWALL1=Q(1)
      RETURN
      END
```

[redacted]

* T I D Y *

UNSTEADY STATE HEAT TRANSFER INTO FLOOR

```
C       UNSTEADY STATE HEAT TRANSFER INTO FLOOR
C
      SUBROUTINE BOND(QWALL1,TWALL1,NTIME,L)
C
      DIMENSION T(2,100,20),Q(20)
C
C
      NTEMP=20
      NLAY=100
      WWALL=91500.0
      CWALL=0.25
      AM=2.0
      IF (L.GT.0) GO TO 220
      IF (NTIME.GT.1) GO TO 130
      DO 110 K=1,NTEMP
      DO 100 J=1,NLAY
100   T(1,J,K)=100.0
110   CONTINUE
C
      DO 120 K=1,NTEMP
      T(2,100,K)=100.0
120   T(2,1,K)=100.0+100.0*(K-1)
      GO TO 160
130   DO 140 K=1,NTEMP
      DO 140 J=1,NLAY
      T(1,J,K)=T(2,J,K)
140   T(2,J,K)=0.0
C
      DO 150 K=1,NTEMP
      T(2,1,K)=T(1,1,K)
150   T(2,100,K)=100.0
160   NLAY2=NLAY-1
      DO 180 K=1,NTEMP
      DO 170 J=2,NLAY2
170   T(2,J,K)=(T(1,J-1,K)+T(1,J+1,K)+(T(1,J,K)*(AM-2.0)))/AM
180   CONTINUE
C
      DO 190 K=1,NTEMP
      Q(K)=0.0
190   CONTINUE
C
      DO 210 K=1,NTEMP
      DO 200 J=1,NLAY2
200   Q(K)=Q(K)+WWALL*CWALL*(T(2,J,K)-100.0)
210   CONTINUE
      RETURN
C
220   GO TO (230,270),L
230   IF (QWALL1.LE.Q(1)) GO TO 260
      DO 240 K=2,20
```

[redacted]

~~XXXXXXXXXXXXXXXXXX~~
~~XXXXSECRETXXXXXXXX~~
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~~XXXXXXXXXXXXXXXXXX~~ [redacted]

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*T I D Y *

UNSTEADY STATE HEAT TRANSFER INTO FLOOR

```
      IF (QWALL1.LE.Q(K)) GO TO 250
240   CONTINUE
      TWALL1=T(2,1,20)
      RETURN
250   TWALL1=(T(2,1,K)-T(2,1,K-1))*(QWALL1-Q(K-1))/(Q(K)-Q(K-1))+T(2,1,K
     1-1)
      RETURN
260   TWALL1=T(2,1,1)
      RETURN
C
270   IF (TWALL1.LE.T(2,1,1)) GO TO 300
      DO 280 K=2,20
      IF (TWALL1.LE.T(2,1,K)) GO TO 290
280   CONTINUE
      QWALL1=Q(20)
      RETURN
290   QWALL1=(Q(K)-Q(K-1))*(TWALL1-T(2,1,K-1))/(T(2,1,K)-T(2,1,K-1))+Q(K
     1-1)
      RETURN
300   QWALL1=Q(1)
      RETURN
      END
```


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[redacted] XXXXXXXXXXXXXXX
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XXXXXXXXXXXXXXXXXX

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```
C       COLLISION PROBABILITY IN A SLAB
C
      SUBROUTINE ESCAPE(BMU,PC)
C
      DIMENSION PP(40)
      PP(1)=0.0484
      PP(2)=0.0831
      PP(3)=0.1127
      PP(4)=0.1390
      PP(5)=0.1629
      PP(6)=0.1849
      PP(7)=0.2054
      PP(8)=0.2246
      PP(9)=0.2427
      PP(10)=0.2597
      PP(11)=0.3932
      PP(12)=0.4859
      PP(13)=0.5554
      PP(14)=0.6097
      PP(15)=0.6533
      PP(16)=0.6890
      PP(17)=0.7187
      PP(18)=0.7437
      PP(19)=0.7651
      PP(20)=0.8757
      PP(21)=0.9167
      PP(22)=0.9375
      PP(23)=0.9500
      PP(24)=0.9580
      PP(25)=0.9650
      PP(26)=0.9690
      PP(27)=0.9720
      PP(28)=0.9740
      IF (BMU.LE.0.1) K=BMU*100.0
      IF (BMU.LE.1.0.AND.BMU.GT.0.1) K=BMU*10.0+9.0
      IF (BMU.LE.10.0.AND.BMU.GT.1.0) K=BMU+18.0
      IF (BMU.GE.10.0) PC=0.974
      IF (BMU.GE.10.0) RETURN
      IF (BMU.LE.0.1) A=BMU*100.0
      IF (BMU.LE.1.0.AND.BMU.GT.0.1) A=BMU*10.0+9.0
      IF (BMU.LT.10.0.AND.BMU.GT.1.0) A=BMU+18.0
      B=K
      PC=PP(K)+(A-B)*(PP(K+1)-PP(K))
      RETURN
      END
```

[redacted]

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[redacted] XXXXXXXXXXXXXXX
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XXXXXXXXXXXXXXXXXX
XXXXXXXXXXXXXXXXXX

TIDY

COLLISION PROBABILITY IN A CYLINDER

```
C       COLLISION PROBABILITY IN A CYLINDER
C
      SUBROUTINE GETOUT(AMU,PCC)
C
      DIMENSION PP(40)
      PP(1)=0.115
      PP(2)=0.207
      PP(3)=0.284
      PP(4)=0.348
      PP(5)=0.404
      PP(6)=0.452
      PP(7)=0.494
      PP(8)=0.531
      PP(9)=0.564
      PP(10)=0.592
      PP(11)=0.764
      PP(12)=0.837
      PP(13)=0.877
      PP(14)=0.901
      PP(15)=0.917
      PP(16)=0.929
      PP(17)=0.938
      PP(18)=0.945
      PP(19)=0.950
      PP(20)=0.975
      PP(21)=0.983
      PP(22)=0.988
      PP(23)=0.990
      PP(24)=0.992
      PP(25)=0.993
      PP(26)=0.994
      PP(27)=0.994
      PP(28)=0.995
      IF (AMU.LE.1.0) K=AMU*10.0
      IF (AMU.LE.10.0.AND.AMU.GT.1.0) K=AMU+9.0
      IF (AMU.LE.100.0.AND.AMU.GT.10.0) K=AMU*0.1+18.0
      IF (AMU.GT.100.0) PCC=0.995
      IF (AMU.GT.100.0) RETURN
      IF (AMU.LE.1.0) A=AMU*10.0
      IF (AMU.LE.10.0.AND.AMU.GT.1.0) A=AMU+9.0
      IF (AMU.LE.100.0.AND.AMU.GT.10.0) A=AMU*0.1+18.0
      B=K
      PCC=PP(K)+(A-B)*(PP(K+1)-PP(K))
      RETURN
      END
```

[redacted]

[redacted] ~~SECRET~~

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(Blank)

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APPENDIX B

LAVA CODE OUTPUT

Units of BTU and degree Fahrenheit are used in the LAVA code. Units of pcu (pound-centigrade unit) and degree centigrade are used in the text of this report.

[redacted]

INPUT INFORMATION

MELTING TEMPERATURE OF FUEL	1220.0
MASS OF AIR FROM -40	5120.0
TEMPERATURE OF AMBIENT SYSTEM	100.0
EFFECTIVE MASS OF CORE	15000.0
EFFECTIVE MASS OF EFFLUENT PIPING AND PUMPS	150000.0
EFFECTIVE AREA OF EFFLUENT PIPING AND PUMPS	1000.0
EMISSIVITY OF EFFLUENT PIPING AND PUMPS	0.5
EMISSIVITY OF CONCRETE WALLS AT -40	0.8
EFFECTIVE SURFACE AREA OF CONCRETE WALLS	30000.0
THERMAL CONDUCTIVITY OF CONCRETE	0.5
COEFFICIENT FOR CONVECTION EQUATION	0.2
RELATIVE HEAT GENERATION	0.6

OUTPUT INFORMATION

TIME	FUEL TEMP	AVG PIPE TEMP	CONCRETE WALL TEMP	AIR TEMP	PIPE SURFACE TEMP	INTERFACE TEMP	HEAT IN
1	520.3	249.2	102.8	100.1	100.0	525.1	3700000.0
2	629.3	434.0	101.8	103.2	289.0	632.9	3300000.0
3	675.4	505.8	101.5	104.3	355.8	678.8	3080000.0
4	740.4	575.1	101.4	106.0	439.4	742.8	2900000.0
5	793.5	639.2	101.4	107.5	509.4	795.3	2730000.0

EX-1176

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WILMINGTON, DELAWARE 19898

EXPLOSIVES DEPARTMENT

DPST-70-433-TL

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October 29, 1970

Mr. N. Stetson, Manager (35)
Savannah River Operations Office
U. S. Atomic Energy Commission
Aiken, South Carolina 29801

Dear Mr. Stetson:

Attached is a copy of DPST-70-433, "Analysis of Postulated Core Meltdown of an SRP Reactor - Final Report." This document presents the calculated consequence of a loss-of-coolant accident in a Savannah River Plant reactor in which either: (1) the emergency cooling system fails to function or (2) actuation of the system is delayed. Preliminary calculations were contained in an earlier document, DPST-67-639.

This report is the second of the four reports requested by DRL-ACRS in September 1969, to complete the studies of the conformance of SRP reactors to licensing criteria. We expect to transmit the two remaining reports together with a summary report in early November 1970.

While this report might be considered a "Safety Analysis," we would not expect it to fall in the category of an SAR which is subject to formal approval by the AEC under the procedure now being discussed by our staffs.

Very truly yours,

ATOMIC ENERGY DIVISION

J. W. Croach

J. W. Croach, Director
Technical Division

WSD:bw
Attachment: DPST-70-433

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Mr. N. Stetson, Manager (35)
Savannah River Operations Office
U. S. Atomic Energy Commission
Aiken, South Carolina 29801

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Very truly yours,

ATOMIC ENERGY DIVISION

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Technical Division

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EXTERNAL RELEASE OF TECHNICAL INFORMATION

Description of Material No. DPST-70-433 Date September 23, 1983

Title: ANALYSIS OF POSTULATED CORE MELTDOWN OF AN SRP REACTOR

Author(s) William S. Durant & Robert J. Brown

Date of Material

( ) Classified DP Report	( ) Classified Paper
( ) Unclassified DP Report	(XXX) Unclassified Paper
( ) Classified Letter	( ) Classified Abstract or Summary
( ) Unclassified Letter	( ) Unclassified Abstract or Summary

Technical Content

Approved by T. V. Crawford Date 10/24/83
T. V. Crawford

Classification

Approved by T. V. Crawford Date 10/24/83
T. V. Crawford

Approved by SWO'Rear / CJ Banick Date 10/3/83
AED Classification Officer

Authority

CG-UF-3, Topic 2.2

Category if DP Report

Approved by N/A Date
Supervisor, I&PS

Released by:

A. F. Westerdahl 10/4/83 AW

J. M. Gaver 10/12/83 JG

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DPST-70-433 (Deleted Version)	DE-AC09-76SR00001	UC-80

4. Title
ANALYSIS OF POSTULATED CORE MELTDOWN OF AN SRP REACTOR

5. Type of Document ("x" one)
- ☒ a. Scientific and technical report
- ☐ b. Conference paper: Title of conference ________
  ________ Date of conference ________
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Signature C J Banick Date 6/27/84

OSR 25-5
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INCORPORATED
ATOMIC ENERGY DIVISION
SAVANNAH RIVER PLANT
AIKEN, SOUTH CAROLINA 29808-0001
(TWX 810-771-2670 TEL 803-725-6211 WU AUGUSTA GA)

CC: S. W. O'Rear

September 23, 1983

Mr. A. F. Westerdahl, Chief
Patent Branch
U. S. Department of Energy
Aiken, South Carolina 29808

Dear Mr. Westerdahl:

REQUEST FOR PATENT REVIEW

Please review for patent matter the documents described in the attached list.

The documents are proposed for release to the general public. Following necessary approvals, copies will be sent to DOE/TIC and also to public reading rooms containing documents cited in the Draft Environmental Impact Statement Pertaining to the Proposed Restart of the L-Reactor at SRP.

If any technical clarification is needed please call C. J. Banick whose Document Review is attached.

Please telephone your comments to the Records Management office (Ext. 2606) and notify me by signing and returning to C. J. Banick the original of this letter. A copy is provided for your file.

If you decide to pursue a patent on any development covered, I shall be happy to supply additional information required such as appropriate references and the names of persons responsible for the development.

Very truly yours,

S. W. O'Rear, Chief Supervisor
Records Management

By: C J Banick
C. J. Banick

The above item is approved for release.

A F Westerdahl 10/11/83

A. F. Westerdahl
Chief, Patent Branch
DOE-SR

Date

SOME REFERENCES THAT ARE CITED IN THE L-REACTOR RESTART EIS

	DOCUMENT NUMBER	TITLE	AUTHOR(S)
1.	DPE-2383	Earthquake Criteria For The Savannah River Plant - March 1968	T. V. Crawford
2.	DPE-3465	Danger Of Soil Liquefaction Caused By Earthquake At The Savannah River Plant	F. H. Herzog
3.	DPST-69-566	Seismic Resistance Of The Savannah River Reactors	J. A. List
4.	DPST-70-433	Analysis Of Postulated Core Melt-down Of An SRP Reactor	W. S. Durant R. J. Brown
5.	DPST-70-435	Radioactivity In The Environs of Steel Creek - August 20, 1970	W. L. Marter
6.	DPST-70-463	Analysis Of The Savannah River Reactor - Emergency Core Cooling System	J. W. Joseph, Jr. R. C. Thornberry
7.	DPSP-70-1457	Source Rod Failure - K Area	W. M. Olliff
8.	DPST-76-461	Tornadic Resistance of 105 Building Disassembly Areas	W. W. F. Yau C. W. Zeh
9.	DPST-79-509	Energy Recovery With Hydraulic Turbines	J. L. Jarriel J. B. Price
10.	DPST-79-531	Use of Cooling Ponds and Hydraulic Turbines to Save SRP Energy Consumption	J. B. Price
11.	WILLJO-80-RSR	Report On The Savannah River Plant Hydroelectric Development	Tudor Engineering Co. Denver, Colorado
12.	DPST-80-250	Preliminary Safety Analysis Defense Waste Processing Facility Reference Case Volume 1 of 4	W. R. Stevens
13.	DPST-81-481	Analysis of Water Diversion Paths To By-Pass Clay Pipe	H. P. Olson

OSR 25-5
(REV 3-83)

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SAVANNAH RIVER PLANT
AIKEN, SOUTH CAROLINA 29808-0001
(TWX 810-771-2670, TEL 803-725-6211, WU AUGUSTA, GA.)

CC: A. F. Westerdahl, DOE-SR
S. W. O'Rear

September 23, 1983

TO: S. W. O'REAR

FROM: C. J. BANICK CJB

DOCUMENT REVIEW

Document: See Attached List

Title:

Author(s)

Contractual Origin: DE-AC09-SR7600001

Present Classification: Unclassified Paper

References:

No items were noted that, in my opinion, should be called to the attention of the DOE for patent consideration.

CC: J. M. Gaver, DOE-SR
E. B. Sheldon, SRP
W. C. Reinig
L. Hibbard
S. Mirshak - J. A. Porter, SRL
D A. Ward - M. R. Buckner
T. V. Crawford
A. L. Boni
J. C. Corey
File (DPST-70-433)

September 23, 1983

TO DISTRIBUTION

The documents described in the attached list are proposed for release to the general public. Following necessary approvals, copies will be sent to DOE/TIC and also to public reading rooms containing documents cited in the Draft Environmental Impact Statement Pertaining to the Proposed Restart of the L-Reactor at SRP.

The classification reviewer has determined this document to be unclassified.

If there are comments about its release, notify the Records Management office as soon as possible (Ext. 2606).

TO: Senior Information Specialist, I&PS, SRL, 773-A DATE: 10/12/83

DOCUMENT NO.: DPST-70-433, Analysis of Postulated Core Meltdown Of An SRP Reactor

RECEIVED OEA/SR ON: 9 / 26 / 83

The above referenced document has been reviewed by SR staff and is:

/ / Approved for release as written.

/ x / Approved for release subject to the following changes:

Document contains Section 148 information.
Bill Wisenbacker has deleted version

/ / Not Approved for release.

ADDITIONAL COMMENTS:

Deleted version placed in DOE-SR Reading Room, Aiken, SC. RES 10/14/83

for M R Rearden
Robert C. Webb, Deputy Director
Office of External Affairs

Attachment

[redacted]

6/28

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- do we have a folder on this?
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[redacted]

6	852.2	700.2	101.5	109.0	576.1	853.4	2650000.0
7	904.7	758.8	101.7	110.5	637.9	905.2	2560000.0
8	957.8	815.3	101.9	111.9	698.6	957.9	2480000.0
9	1007.4	869.6	102.2	113.4	756.3	1007.8	2400000.0
10	1056.7	921.9	102.6	114.8	811.6	1056.6	2340000.0
11	1104.8	972.5	103.0	116.3	865.0	1103.3	2280000.0
12	1150.4	1021.5	103.5	117.9	916.7	1149.3	2220000.0
13	1196.7	1068.8	104.0	119.1	965.8	1196.3	2190000.0
14	1220.0	1112.9	104.7	120.8	1020.4	1216.6	2160000.0
15	1220.0	1148.0	105.4	122.4	1080.0	1215.2	2110000.0
16	1220.0	1170.0	106.1	124.0	1126.2	1212.5	2050000.0
17	1220.0	1182.5	106.9	124.8	1154.2	1209.3	2020000.0
18	1298.1	1196.9	107.7	124.5	1146.2	1285.7	2000000.0

AT THIS CONDITION THE PIPE COLLAPSES

INPUT INFORMATION

FLOOR AREA AT -40	5000.0
MELTING TEMPERATURE OF LAVA	1900.0
LINEAR ABSORPTION COEF OF FUEL	0.20000
LINEAR ABSORPTION COEF OF STAINLESS STEEL	0.57000
DENSITY OF FUEL	200.00000
DENSITY OF STAINLESS STEEL	480.00000
EMISSIVITY OF DEBRIS	0.5

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MASS OF STEEL IN LAVA 85050.7

OUTPUT INFORMATION

TIME	AVG LAVA TEMP	LAVA SURF TEMP	LAVA AREA	WALL TEMP	AIR TEMP	HEAT IN	HEAT OXID	HEAT CONV	HEAT COND	HEAT RAD
19	1904.4	1901.9	120.0	109.2	105.8	1960000.0	1813.1	8630.9	912792.6	47517.2
20	1903.1	1900.4	280.0	110.3	116.1	1940000.0	4359.3	20205.6	1048918.0	111118.9
21	1903.6	1901.2	420.0	112.0	123.6	1920000.0	6697.0	30195.9	943722.7	166655.6
22	1906.3	1904.2	540.0	114.2	129.3	1900000.0	8822.1	38669.1	835768.1	214937.9
23	1903.4	1901.5	660.0	116.9	135.2	1880000.0	11010.0	47054.0	852427.1	262745.8
24	1903.8	1902.1	760.0	119.9	140.1	1860000.0	12869.3	53915.6	739335.3	301982.9
25	1904.0	1902.4	850.0	123.2	144.7	1840000.0	14630.2	60098.7	687435.9	337964.5
26	1904.3	1902.8	930.0	126.7	148.8	1820000.0	16240.2	65551.9	634553.8	369966.6
27	1905.1	1903.8	1000.0	130.4	152.9	1800000.0	17698.5	70357.9	579717.8	398236.2
28	1902.9	1901.6	1070.0	134.3	156.6	1790000.0	19140.7	75017.4	587203.5	425606.6
29	1901.8	1900.5	1130.0	138.4	159.9	1770000.0	20355.9	78907.7	529405.8	448227.7
30	1902.5	1901.3	1180.0	142.5	162.8	1760000.0	21430.8	82180.7	470683.1	467857.8
31	1904.9	1903.8	1220.0	146.6	165.3	1740000.0	22386.2	84884.2	411687.7	484989.3
32	1901.2	1900.1	1270.0	150.8	168.4	1730000.0	23475.7	88138.0	479115.8	504317.6
33	1903.8	1902.8	1300.0	155.0	170.3	1710000.0	24125.9	89999.6	354274.3	515684.5
34	1904.9	1903.9	1330.0	159.3	172.5	1700000.0	24910.9	92051.4	356631.9	529218.6
35	1904.2	1903.2	1360.0	163.5	174.6	1630000.0	25617.5	93993.7	358958.4	541310.4
36	1902.3	1901.4	1390.0	167.7	176.5	1670000.0	26245.9	95819.6	359604.4	551983.4
37	1903.2	1902.2	1410.0	171.9	178.0	1650000.0	26691.2	97092.2	296114.3	559493.6
38	1903.0	1902.1	1430.0	176.0	179.5	1640000.0	27184.0	98366.1	296554.3	567701.1
39	1901.7	1900.8	1450.0	180.1	180.8	1620000.0	27561.2	99501.4	295908.1	573957.8

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40	1903.4	1902.5	1460.0	184.2	181.6	1600000.0	27866.7	100166.4	232139.1	578945.3
41	1903.9	1903.1	1470.0	188.2	182.6	1580000.0	28176.4	100861.3	231945.1	583984.7
42	1903.9	1903.1	1480.0	192.2	183.7	1570000.0	28434.6	101501.2	231362.6	588163.4
43	1903.3	1902.5	1490.0	196.1	184.7	1560000.0	28655.7	102079.4	230053.7	591727.7
44	1902.2	1901.4	1500.0	199.9	185.8	1550000.0	28843.8	102609.6	228671.7	594746.9
45	1904.9	1904.1	1500.0	203.7	186.5	1540000.0	28896.5	102590.4	163537.9	595512.2
46	1902.9	1902.1	1510.0	207.5	187.8	1530000.0	29158.9	103229.1	225985.8	599719.2
47	1904.7	1903.9	1510.0	211.1	188.5	1520000.0	29154.8	103145.6	161224.6	599569.8
48	1901.9	1901.1	1520.0	214.8	189.7	1510000.0	29364.1	103705.5	222620.9	602898.6
49	1903.1	1902.3	1520.0	218.3	190.4	1500000.0	29315.3	103566.1	157207.8	602026.1
50	1903.5	1902.7	1520.0	221.8	191.2	1480000.0	29367.5	103555.1	155664.6	602782.6
51	1903.7	1902.9	1520.0	225.2	192.0	1470000.0	29388.2	103521.5	154403.3	603030.4
52	1903.6	1902.9	1520.0	228.6	192.8	1460000.0	29392.3	103460.3	152172.9	603010.1
53	1903.4	1902.6	1520.0	231.9	193.6	1450000.0	29381.5	103379.8	150318.4	602749.1
54	1902.9	1902.1	1520.0	235.2	194.4	1440000.0	29356.8	103284.3	148478.9	602265.8
55	1902.2	1901.5	1520.0	238.4	195.2	1430000.0	29319.6	103173.9	146225.6	601589.3
56	1901.5	1900.7	1520.0	241.5	195.9	1420000.0	29224.2	102990.2	143324.2	599962.4
57	1900.6	1899.8	1520.0	244.6	196.7	1410000.0	29201.4	102901.3	141283.0	599510.1
58	1900.0	1899.3	1520.0	247.6	197.5	1400000.0	29172.5	102803.7	139689.4	598957.1
59	1900.0	1896.2	1520.0	250.5	198.1	1390000.0	28961.1	102470.9	135895.7	595456.5
60	1900.0	1896.2	1520.0	253.4	198.9	1380000.0	28962.9	102418.9	133871.7	595401.8
61	1900.0	1896.2	1520.0	256.3	199.7	1370000.0	28964.6	102364.9	133144.2	595347.7
62	1900.0	1896.3	1520.0	259.1	200.5	1360000.0	28966.3	102301.2	132285.9	595293.8
63	1900.0	1896.3	1520.0	261.9	201.2	1350000.0	28968.0	102240.6	131336.3	595249.6
64	1900.0	1896.3	1520.0	264.6	202.0	1340000.0	28969.8	102182.0	130490.5	595196.0

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65	1900.0	1896.4	1520.0	267.3	202.7	1330000.0	28971.5	102122.9	129672.1	595133.3
66	1900.0	1896.4	1520.0	270.0	203.5	1320000.0	28973.2	102064.1	128684.3	595089.6
67	1900.0	1896.4	1520.0	272.6	204.2	1310000.0	28974.9	102006.0	128037.1	595036.7
68	1900.0	1896.4	1520.0	275.2	205.0	1300000.0	28976.7	101948.1	126886.6	594984.0
69	1900.0	1895.5	1520.0	277.8	205.7	1290000.0	28978.4	101890.8	126166.2	594932.5
70	1900.0	1896.5	1520.0	280.3	206.4	1280000.0	28980.1	101833.8	125516.0	594880.2
71	1900.0	1896.5	1520.0	282.8	207.2	1280000.0	28980.1	101775.1	124541.5	594801.1
72	1900.0	1896.5	1520.0	285.2	207.9	1270000.0	28981.8	101719.2	123402.1	594749.6
73	1900.0	1896.5	1520.0	287.6	208.6	1270000.0	28981.8	101661.6	122796.6	594670.6
74	1900.0	1896.6	1520.0	290.0	209.3	1260000.0	28983.5	101606.3	122008.7	594619.4
75	1900.0	1896.6	1520.0	292.4	210.0	1260000.0	28983.5	101549.4	121100.1	594541.1
76	1900.0	1896.6	1520.0	294.8	210.7	1250000.0	28985.3	101495.2	120148.7	594490.1
77	1900.0	1896.6	1520.0	297.1	211.4	1250000.0	28985.3	101439.1	119582.6	594412.2
78	1900.0	1896.6	1520.0	299.4	212.1	1240000.0	28987.0	101385.5	118669.9	594361.6
79	1900.0	1896.6	1520.0	301.6	212.7	1240000.0	28987.0	101330.3	117922.2	594284.0
80	1900.0	1896.6	1520.0	303.8	213.4	1230000.0	28988.7	101277.6	117256.0	594234.1
81	1900.0	1896.6	1520.0	306.1	214.1	1230000.0	28988.7	101223.2	116440.9	594157.1
82	1900.0	1896.7	1520.0	308.2	214.8	1220000.0	28990.4	101171.4	115691.1	594107.2
83	1900.0	1896.7	1520.0	310.4	215.4	1220000.0	28990.4	101117.7	114860.9	594030.4
84	1900.0	1896.7	1520.0	312.5	216.1	1210000.0	28992.1	101066.6	114164.4	593981.1
85	1900.0	1896.7	1520.0	314.7	216.7	1210000.0	28992.1	101013.7	113553.4	593905.1
86	1900.0	1896.7	1520.0	316.8	217.4	1200000.0	28993.9	100963.5	112697.0	593856.0
87	1900.0	1896.7	1520.0	318.8	218.0	1200000.0	28993.9	100911.4	112170.1	593780.0
88	1900.0	1896.7	1520.0	320.9	218.7	1190000.0	28995.6	100861.9	111278.6	593731.2
89	1900.0	1896.7	1520.0	322.9	219.3	1190000.0	28995.6	100810.6	110786.0	593655.7

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90	1900.0	1896.8	1520.0	324.9	219.9	1180000.0	28997.3	100761.7	110093.2	593607.7
91	1900.0	1896.8	1520.0	326.9	220.6	1180000.0	28997.3	100711.2	109423.8	593532.2
92	1900.0	1896.8	1520.0	328.9	221.2	1170000.0	28999.0	100663.1	108761.5	593484.5
93	1900.0	1896.8	1520.0	330.8	221.8	1170000.0	28999.0	100613.3	108116.0	593409.3
94	1900.0	1896.8	1520.0	332.7	222.4	1160000.0	29000.8	100565.9	107474.2	593362.1
95	1900.0	1896.8	1520.0	334.7	223.0	1150000.0	29000.8	100516.7	106851.7	593287.7
96	1900.0	1896.9	1520.0	336.6	223.6	1150000.0	29002.5	100470.2	106235.4	593240.2
97	1900.0	1896.9	1520.0	338.4	224.2	1150000.0	29002.5	100421.6	105634.2	593166.1
98	1900.0	1896.9	1520.0	340.3	224.8	1150000.0	29002.5	100373.6	105011.4	593092.0
99	1900.0	1896.9	1520.0	342.1	225.4	1140000.0	29004.2	100328.0	104450.3	593045.6
100	1900.0	1896.9	1520.0	344.0	226.0	1140000.0	29004.2	100280.4	103860.7	592971.4
101	1900.0	1896.9	1520.0	345.8	226.6	1140000.0	29004.2	100233.2	103293.1	592897.8
102	1900.0	1896.9	1520.0	347.6	227.2	1130000.0	29006.0	100188.7	102731.6	592851.7
103	1900.0	1896.9	1520.0	349.3	227.7	1130000.0	29006.0	100142.1	102198.8	592778.6
104	1900.0	1896.9	1520.0	351.1	228.3	1130000.0	29006.0	100095.8	101629.8	592705.1
105	1900.0	1896.9	1520.0	352.9	228.9	1120000.0	29007.7	100052.2	101126.8	592668.7
106	1900.0	1896.9	1520.0	354.6	229.4	1120000.0	29007.7	100006.5	100589.8	592596.0
107	1900.0	1896.9	1520.0	356.3	230.0	1120000.0	29007.7	99961.1	100073.7	592522.9
108	1900.0	1897.0	1520.0	358.0	230.6	1110000.0	29009.4	99918.4	99570.2	592477.8
109	1900.0	1897.0	1520.0	359.7	231.1	1110000.0	29009.4	99873.6	99084.9	592405.6
110	1900.0	1897.0	1520.0	361.4	231.7	1110000.0	29009.4	99829.1	98563.0	592333.6
111	1900.0	1897.0	1520.0	363.0	232.2	1100000.0	29011.1	99787.2	98105.5	592288.[illegible]
112	1900.0	1897.0	1520.0	364.7	232.8	1100000.0	29011.1	99743.3	97623.9	592216.[illegible]
113	1900.0	1897.0	1520.0	366.3	233.3	1100000.0	29011.1	99699.6	97144.1	592144.7
114	1900.0	1897.0	1520.0	368.0	233.8	1090000.0	29012.9	99658.6	96687.5	592100.5

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115	1900.0	1897.0	1520.0	369.6	234.4	1090000.0	29012.9	99615.5	96246.5	592028.7
116	1900.0	1897.0	1520.0	371.2	234.9	1090000.0	29012.9	99572.6	95769.1	591957.3
117	1900.0	1897.0	1520.0	372.8	235.4	1080000.0	29014.6	99532.4	95347.2	591913.8
118	1900.0	1897.0	1520.0	374.3	236.0	1080000.0	29014.6	99490.1	94906.6	591842.4
119	1900.0	1897.0	1520.0	375.9	236.5	1080000.0	29014.6	99448.0	94467.4	591771.4
120	1900.0	1897.1	1520.0	377.4	237.0	1070000.0	29016.3	99408.6	94048.3	591728.0
121	1900.0	1897.1	1520.0	379.0	237.5	1070000.0	29016.3	99367.0	93642.7	591657.3
122	1900.0	1897.1	1520.0	380.5	238.0	1070000.0	29016.3	99325.7	93200.6	591586.6
123	1900.0	1897.1	1520.0	382.0	238.5	1060000.0	29018.0	99287.0	92814.2	591544.2
124	1900.0	1897.1	1520.0	383.5	239.0	1060000.0	29018.0	99246.1	92410.4	591473.6
125	1900.0	1897.1	1520.0	385.0	239.5	1060000.0	29018.0	99205.6	91999.4	591403.7
126	1900.0	1897.1	1520.0	386.5	240.0	1050000.0	29019.8	99167.6	91617.5	591361.3
127	1900.0	1897.1	1520.0	388.0	240.5	1050000.0	29019.8	99127.5	91237.2	591291.2
128	1900.0	1897.2	1520.0	389.4	241.0	1040000.0	29021.5	99089.8	90861.0	591248.7
129	1900.0	1897.2	1520.0	390.9	241.5	1040000.0	29021.5	99050.2	90481.6	591179.4
130	1900.0	1897.2	1520.0	392.3	242.0	1040000.0	29021.5	99010.9	90096.1	591109.8
131	1900.0	1897.2	1520.0	393.7	242.5	1030000.0	29023.2	98974.1	89741.4	591068.0
132	1900.0	1897.2	1520.0	395.2	243.0	1030000.0	29023.2	98935.1	89375.9	590998.9
133	1900.0	1897.2	1520.0	396.6	243.5	1020000.0	29024.9	98898.6	89035.7	590957.6
134	1900.0	1897.2	1520.0	398.0	244.0	1020000.0	29024.9	98860.1	88670.4	590888.7
135	1900.0	1897.2	1520.0	399.4	244.4	1020000.0	29024.9	98821.9	88306.8	590820.0
136	1900.0	1897.2	1520.0	400.7	244.9	1010000.0	29026.7	98786.1	87972.0	590778.9
137	1900.0	1897.2	1520.0	402.1	245.4	1010000.0	29026.7	98748.3	87632.1	590710.6
138	1900.0	1897.3	1520.0	403.5	245.8	1000000.0	29028.3	98712.9	87298.3	590669.3
139	1900.0	1897.3	1520.0	404.8	246.3	1000000.0	29028.3	98675.6	86964.8	590601.3

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140	1900.0	1897.3	1520.0	406.2	246.8	1000000.0	29028.3	98638.4	86617.9	590533.1
141	1900.0	1897.3	1520.0	407.5	247.2	990000.0	29030.1	98603.7	86306.7	590493.1
142	1900.0	1897.3	1520.0	408.8	247.7	990000.0	29030.1	98566.9	85977.9	590425.1
143	1900.0	1897.3	1520.0	410.2	248.1	980000.0	29031.8	98532.4	85673.5	590385.1
144	1900.0	1897.3	1520.0	411.5	248.6	980000.0	29031.8	98496.1	85351.7	590317.6
145	1900.0	1897.3	1520.0	412.8	249.0	980000.0	29031.8	98459.9	85023.9	590250.1
146	1900.0	1897.3	1520.0	414.1	249.5	970000.0	29033.5	98426.2	84726.6	590210.4
147	1900.0	1897.3	1520.0	415.4	249.9	970000.0	29033.5	98390.4	84420.5	590143.2
148	1900.0	1897.4	1520.0	416.6	250.4	960000.0	29035.3	98356.9	84128.2	590103.4
149	1900.0	1897.4	1520.0	417.9	250.8	960000.0	29035.3	98321.5	83827.9	590036.7
150	1900.0	1897.4	1520.0	419.2	251.3	960000.0	29035.3	98286.3	83512.8	589970.1
151	1900.0	1897.4	1520.0	420.4	251.7	960000.0	29035.3	98251.4	83202.2	589903.4
152	1900.0	1897.4	1520.0	421.7	252.1	950000.0	29037.0	98218.7	82930.2	589864.2
153	1900.0	1897.4	1520.0	422.9	252.6	950000.0	29037.0	98184.1	82642.9	589797.8
154	1900.0	1897.4	1520.0	424.1	253.0	950000.0	29037.0	98149.6	82335.6	589731.4
155	1900.0	1897.4	1520.0	425.4	253.4	950000.0	29037.0	98115.3	82044.6	589665.6
156	1900.0	1897.4	1520.0	426.6	253.9	940000.0	29038.7	98083.4	81786.1	589626.9
157	1900.0	1897.4	1520.0	427.8	254.3	940000.0	29038.7	98049.3	81509.6	589561.3
158	1900.0	1897.4	1520.0	429.0	254.7	940000.0	29038.7	98015.6	81220.7	589495.4
159	1900.0	1897.4	1520.0	430.2	255.1	940000.0	29038.7	97982.0	80933.2	589429.6
160	1898.4	1895.8	1520.0	431.4	255.5	930000.0	28998.5	97897.7	79923.1	588712.2
161	1885.9	1883.4	1520.0	432.5	255.5	930000.0	28559.1	97307.8	71260.5	581510.4
162	1875.3	1872.8	1520.0	433.6	255.3	930000.0	27853.7	96381.3	54985.2	569890.4
163	1866.3	1863.8	1520.0	434.7	255.3	930000.0	27265.8	95613.5	40222.1	560112.0
164	1858.1	1855.6	1520.0	435.7	255.4	920000.0	26763.1	94942.2	29590.4	551667.7

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165	1851.0	1848.5	1520.0	436.6	255.5	920000.0	26322.3	94333.2	21159.3	544214.4
166	1844.7	1842.2	1520.0	437.6	255.5	920000.0	25941.1	93799.6	14833.3	537715.4
167	1839.0	1836.5	1520.0	438.5	255.7	920000.0	25606.1	93324.4	10840.1	531974.1
168	1833.5	1831.0	1520.0	439.4	255.8	910000.0	25298.8	92879.9	8045.6	526670.9
169	1828.5	1826.0	1520.0	440.2	255.9	910000.0	25171.3	92458.1	5070.3	521681.3
170	1823.9	1821.4	1520.0	441.1	256.0	910000.0	25140.9	92071.5	4523.4	517133.6
171	1819.6	1817.1	1520.0	441.9	256.1	910000.0	25113.0	91714.3	3579.0	512970.4
172	1815.2	1812.8	1520.0	442.7	256.2	900000.0	25086.1	91369.1	3502.7	508975.2
173	1811.1	1808.7	1520.0	443.5	256.4	900000.0	25059.6	91028.5	3889.4	505050.4
174	1807.3	1804.8	1520.0	444.2	256.5	900000.0	25034.7	90708.4	3884.4	501382.6
175	1803.6	1801.1	1520.0	445.0	256.6	900000.0	25011.2	90406.4	4431.6	497936.0
176	1799.7	1797.3	1520.0	445.7	256.7	890000.0	24988.0	90106.2	5579.2	494536.5
177	1796.1	1793.7	1520.0	446.4	256.8	890000.0	24964.6	89804.7	6030.5	491128.7
178	1792.7	1790.2	1520.0	447.1	257.0	890000.0	24942.5	89520.3	6266.8	487928.2
179	1789.4	1786.9	1520.0	447.7	257.1	890000.0	24921.6	89250.2	7101.3	484900.0
180	1786.2	1783.8	1520.0	448.4	257.2	890000.0	24901.5	88990.7	8257.5	482010.6
181	1782.8	1780.4	1520.0	449.1	257.3	880000.0	24881.4	88730.2	9040.1	479116.5
182	1779.6	1777.2	1520.0	449.7	257.4	880000.0	24860.9	88465.7	9960.0	476187.2
183	1776.5	1774.1	1520.0	450.3	257.5	880000.0	24841.2	88212.1	11216.2	473386.9
184	1773.5	1771.1	1520.0	450.9	257.6	880000.0	24822.3	87967.6	12279.7	470691.4
185	1770.6	1768.2	1520.0	451.5	257.7	880000.0	24804.1	87732.4	13173.6	468121.3
186	1767.5	1765.1	1520.0	452.1	257.8	870000.0	24785.8	87495.4	13839.7	465528.7
187	1764.6	1762.2	1520.0	452.7	257.9	870000.0	24757.0	87252.8	14772.3	462885.1
188	1761.7	1759.3	1520.0	453.2	258.0	870000.0	24749.0	87021.0	15500.8	460364.5
189	175[illegible].0	1756.6	1520.0	453.8	258.[illegible]	870000.0	24731.8	86798.9	16238.3	457958.[illegible]

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190	1756.4	1754.0	1520.0	454.3	258.2	870000.0	24715.3	86585.6	16940.3	455657.6
191	1753.5	1751.2	1520.0	454.8	258.3	860000.0	24698.7	86369.7	17569.2	453340.0
192	1750.8	1748.4	1520.0	455.4	258.4	860000.0	24681.4	86147.2	18400.4	450949.1
193	1748.1	1745.8	1520.0	455.9	258.4	860000.0	24654.9	85934.2	19123.1	448662.6
194	1745.6	1743.3	1520.0	456.4	258.5	860000.0	24649.0	85729.7	19882.4	446478.6
195	1743.2	1740.8	1520.0	456.8	258.6	860000.0	24633.8	85532.7	20657.6	444381.8
196	1740.5	1738.2	1520.0	457.3	258.7	850000.0	24618.2	85331.9	21224.8	442253.3
197	1738.0	1735.6	1520.0	457.8	258.8	850000.0	24602.2	85125.1	21534.2	440059.4
198	1735.5	1733.2	1520.0	458.2	258.8	850000.0	24586.9	84928.7	21878.2	437978.[illegible]
199	1733.2	1730.9	1520.0	458.7	258.9	850000.0	24572.4	84741.4	22365.2	436004.[illegible]
200	1731.0	1728.7	1520.0	459.1	259.0	850000.0	24558.4	84561.4	22899.6	434114.[illegible]
201	1728.5	1726.2	1520.0	459.6	259.1	840000.0	24544.2	84377.7	23300.2	432191.2
202	1726.2	1723.9	1520.0	460.0	259.1	840000.0	24529.4	84187.3	23479.7	430202.1
203	1723.9	1721.6	1520.0	460.4	259.2	840000.0	24515.4	84007.1	23732.4	428316.5
204	1721.8	1719.5	1520.0	460.8	259.3	840000.0	24502.1	83835.1	24108.5	426520.2
205	1719.4	1717.1	1520.0	461.2	259.3	830000.0	24488.4	83659.1	24376.7	424697.6
206	1717.2	1714.9	1520.0	461.6	259.4	830000.0	24474.2	83476.7	24403.0	422802.6
207	1715.0	1712.7	1520.0	462.0	259.4	830000.0	24460.8	83304.6	24550.1	421022.6
208	1713.0	1710.7	1520.0	462.4	259.5	830000.0	24448.0	83140.6	24871.2	419322.9
209	1710.7	1708.4	1520.0	462.7	259.5	820000.0	24434.9	82972.3	25074.9	417590.1
210	1708.5	1706.3	1520.0	463.1	259.6	820000.0	24421.3	82797.5	25036.4	415792.7
211	1706.4	1704.2	1520.0	463.5	259.6	820000.0	24408.4	82632.7	25082.8	414094.4
212	1704.5	1702.2	1520.0	463.8	259.7	820000.0	24396.2	82476.0	25337.8	412487.7
213	1702.3	1700.0	1520.0	464.2	259.7	810000.0	24383.6	82314.7	25523.4	410841.6

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OSR 3-17

DON'T SAY IT - WRITE IT

TO D. A. WARD LOCATION 773A DATE 2/18/82

FROM BETTY JOHNSON PHONE NO. 2147 LOCATION 773A-TIS

We have a request from Regina Harris, DOE/SR for DPST-70-433.

Please initial DAW.

Preventive Safety - Prevents Injuries