Docket File

DEC 16 1977

Docket No. 50-305

Wisconsin Public Service Corporation
ATTN: Mr. E. W. James
Senior Vice President
Post Office Box 1200
Green Bay, Wisconsin 54305

Gentlemen:

As you know, we have been re-evaluating the acceptability of the calculational model used to evaluate the performance of the emergency core cooling system (ECCS) in Westinghouse designed two reactor coolant loop plants, such as your Kewaunee Nuclear Power Plant. Results of our evaluation of the model are presented in our Safety Evaluation Report on ECCS Evaluation Model for Westinghouse Two-Loop Plants, a copy of which is provided as Enclosure 1. This evaluation concludes that the presently approved model does not appropriately account for ECCS water injected above the core.

In addition, we have performed analyses to determine the immediate safety significance of this conclusion for operating two-loop facilities and the nature and timing of any corrective action that may be needed. A discussion of these analyses and our conclusions are presented in our Safety Evaluation Report on Continued Safe Operation of Westinghouse Designed Two-Loop Plants, a copy of which is provided as Enclosure 2. We have concluded that operation of your two-loop facility may continue safely for a limited period of time while we determine, after discussions with you, the proper application of the staff re-evaluation of the Westinghouse two-loop model to your plant.

Accordingly, interim bases for continued safe operation of your facility must be developed within the next 30 days, taking into account the apparent deficiencies in the Westinghouse two-loop models described in our Safety Evaluation Reports. We believe that such interim bases are likely to involve some additional operating limits to compensate for these model deficiencies. In addition, a permanent resolution of the problems giving rise to these model deficiencies should be developed and provided to us, along with a schedule for its implementation, as soon as possible, but not later than 60 days after this 30 day period.

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DEC 16 1977

Therefore, pursuant to 10 CFR 50.54(f) of the Commission's regulations, you are requested to propose, within 30 days from the date of this letter, appropriate bases, including any necessary operating limitations, to justify continued operation of your facility beyond this 30 day period. Any subsequent action that may be required will be based on our evaluation of your submittals. If you do not choose to propose alternative bases, the staff will prepare suitable operating limitations based on its reassessment.

We will be happy to meet with you or any of your representatives to discuss this matter. Please contact your NRC Project Manager if you wish such a meeting or if you have any questions.

Sincerely,

Original signed by

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Safety Evaluation Report on ECCS Evaluation Model for Westinghouse Two-Loop Plants
2. Safety Evaluation Report on Continued Safe Operation of Westinghouse Designed Two-Loop Plants

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

December 16, 1977

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Senior Vice President
Post Office Box 1200
Green Bay, Wisconsin 54305

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Sincerely,

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Safety Evaluation Report on ECCS Evaluation Model for Westinghouse Two-Loop Plants
2. Safety Evaluation Report on Continued Safe Operation of Westinghouse Designed Two-Loop Plants

cc w/enclosures:
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Analysis Branch
Division of Systems Safety
Office of Nuclear Reactor Regulation

SAFETY EVALUATION REPORT

ON

ECCS EVALUATION MODEL

FOR

WESTINGHOUSE TWO-LOOP PLANTS

November 1977

TABLE OF CONTENTS

I. INTRODUCTION

Statement of Problem

This report describes our concerns, with respect to the continued use by owners of the Westinghouse two-loop design of ECC systems which incorporate injection of ECC into the vessel upper plenum.* Our concern is that the interactive effects between the injected cold water and the reactor (core and fluid leaving the core) during the refill and reflood phases of a LOCA may not have been considered conservatively at the time the Appendix K analyses were done. There are six operating 2-loop plants (Prairie Island 1 and 2, Point Beach 1 and 2, Ginna, and Kewaunee). None are proposed or under construction.

The Emergency Core Cooling System (ECCS) for Westinghouse two-loop PWR's is shown in Figure 1. This system injects emergency core cooling water into the cold legs by means of accumulators and high head injection pumps. In addition, ECC water is injected directly into the reactor vessel upper plenum, by means of the low head injection pumps (and high head injection pumps in some plants). The upper plenum injection consists of two four inch injection pipes, 180° apart, which extend through the reactor vessel, the downcomer region and through the core barrel at locations 80° from the nearest hot leg and at approximately the same elevation as the hot legs. Table 1

*Hereafter referred to as upper plenum injection (UPI).

presents the design parameters for the Point Beach ECCS (as an example).

The original design of the two-loop plant ECCS included high head injection into the reactor hot legs, but this arrangement was changed such that the high head hot leg ECCS is rerouted to the upper plenum low head injection lines. This change was made in 1972 when Westinghouse decided that insufficient information existed about the possible steam-water interactions in the hot leg during a postulated loss-of-coolant accident (LOCA). The primary reason for having either hot leg or upper plenum injection on two-loop plants is so that no single failure associated with a postulated cold leg break could defeat the high or low pressure ECC injection. If the injection systems fed only into the cold legs, then a single failure could prevent either high or low head injection into the intact cold leg while the coolant injected into the broken cold leg could be lost out the break.

The evaluation model approved for the Appendix K analyses of Westinghouse two-loop plants includes a model assumption that the low head injection is delivered directly to the lower plenum through the cold leg injection location. This simplified treatment of upper plenum injection allows Westinghouse to use the same evaluation model for two-, three-, and four-loop plants.

Because of the simplifying treatment no accounting is made of the interaction of the cold water injected into the upper plenum (on the order of 100°F, at about 2000 GPM, from each of the two systems) with the steam exiting the core during refill and reflood. The current model also ignores the steam generated when water injected into the upper plenum falls and enters the core. Similarly, the model includes no accounting of heat transfer in the core or possible entrainment of the upper plenum water as it falls into the core.

During the last several years, new generic information has become available on liquid-vapor interaction, counter-current flow regimes, and core thermal response to ECC injected above the core. These developments have been closely followed by the NRC staff. During the past year the staff has also gained significant analysis experience related to the analytical modeling of ECC injected above the core in the course of our review of the Westinghouse upper head injection (UHI) model (reference 1). This analytical and experimental information (including the integral blowdown and reflood tests performed on Semiscale) indicated that the steam-water interactions associated with ECC injection above the core can play an important role in determining the course of a postulated LOCA. Ignoring these interactions in ECCS models is not always conservative. In light of this information, we began a study of the possible effects of upper plenum injection on a postulated LOCA for the operating two-loop plants and the possible treatment of these effects on their evaluation models.

On January 11, 1977, NRC staff discussed the new analytical and experimental information with Westinghouse and the two-loop plant licensees and pointed out those areas which might adversely affect a LOCA. On January 26, 1977, a similar meeting was held at which Westinghouse presented its interpretation of the new data and the applicability of the data to two-loop plant LOCA analyses. The Westinghouse conclusion presented at that meeting was that although the evaluation model did not realistically treat the injection of ECC water above the core, the model was nevertheless conservative as it also ignored the (allegedly) beneficial aspects of upper plenum injection (i.e., improved core heat transfer) which could outweigh the adverse effects (steam binding and retarded bottom reflood).

Subsequently, Northern States Power Company (for Prairie Island) submitted a report in support of the Appendix K model for two-loop plants (reference 2). This report documented the information presented by Westinghouse at the January 26, 1977 meeting. We reviewed this submittal and found that several important areas including the spatial distribution of injection water in the upper plenum, and steam generation in the core due to upper plenum injection, were not treated consistently with available experimental data.

Since then, in an attempt to quantify the effects of UPI, the staff has constructed a model to assess the impact on a postulated cold leg LOCA. The staff model is based on available data and includes: (1) condensation of steam in the upper plenum by the subcooled water injected into the upper plenum; (2) steam generation in the core resulting from upper plenum injection; (3) horizontal entrainment (carrying out) of water injected into the upper plenum when it is near the hot leg nozzles; (4) vertical entrainment of water by steam exiting the core. This model is discussed in detail in Section V of this report.

Summary of Results

The result of applying the staff's model to a two-loop plant cold leg LOCA shows a significant net steam generation from the vaporization of upper plenum injection water in the core and from the vaporization of upper plenum injection water which is entrained (carried over) into the steam generator. Since the total steaming rate during the reflood portion of a cold leg LOCA is determined by the steam relieving capability of the route from the upper plenum to the break via the hot legs (broken and unbroken), the increased steam generation due to upper plenum injection would cause a comparable decrease in the steam generation from the reflooding of the bottom of the core. This reduction in the reflood steaming rate is associated with a reduction in both the reflood rate and bottom quench front progression. Reduced reflood rate and delayed quenching result in higher calculated cladding temperatures. Our sensitivity studies indicate that use of conservative assumptions with respect to spatial distribution of the upper plenum injection water and with respect to entrained liquid carried into the hot legs could result in a calculated peak clad temperature several hundred degrees above the value calculated with the Westinghouse evaluation model for an initial full power condition. The sensitivity to these assumptions is substantially less at reduced power levels.

II. FACILITY DESCRIPTION

The two-loop plants which this report applies to are:
Northern States Power Company, Prairie Island 1 & 2,(1650 MWt);
Wisconsin Electric Power Company, Point Beach 1 & 2,(1520 MWt);
Wisconsin Public Service Corporation, Kewaunee, (1650 MWt);
Rochester Gas & Electric Corporation, R. E. Ginna, (1520 MWt).

Each of these plants has an operating license. The following figures and tables are presented as background information:

Figure 1 - Loop Configuration and ECCS Injection Locations

Figure 2 - Reactor Vessel Internals

Figure 3 - Reactor Core Cross-section (location of guide tubes shown)

Table 1 - ECCS Design Parameters

Table 2 - Reactor Thermal-Hydraulic Design Parameters

Table 3 - Reactor Vessel Design Data

This information is taken from the Point Beach Safety Analysis Report but generally applies to all Westinghouse two-loop plants. One noteable exception is the upper plenum injection flow rate which is approximately 10 percent higher in the 1650 MWt plants.

III. AVAILABLE RESEARCH

The substance of the licensee's contentions is that UPI fluid channels through the core with negligible interaction with the fuel rods. The literature review that follows is intended to cover:

a. is UPI delivered in such a manner as to provide a concentrated downward jet, or a dispersal "fog" flow spread out over upper plenum, or somewhere in between;

b. does UPI extract heat from core; and

c. does updraft of steam entrain UPI and carry it elsewhere?

None of these conditions are assumed in the current model.

Available Research on Upper Plenum Injection Flow Distribution

The low pressure injection water enters the upper plenum through a horizontal four-inch pipe which penetrates the reactor vessel, the downcomer and the core barrel. At the design flow rate of 240 lb/sec, the velocity of the water entering the upper plenum is 45 ft/sec in the horizontal direction. for the two-loop plants with the higher UPI flow rates the velocity approaches 60 ft/sec. The distribution of this water in the upper plenum and the associated fraction of the core which this water covers is extremely important in determining the effects on a postulated LOCA. If the water remains in a highly localized area, then both the interaction with the steam exiting the core and the steam generation in the core will be relatively small. If the water is distributed over a wide area of the core, then both the interaction with the steam exiting the core and the steam generation in the core will be relatively larger. Increased steam-water interaction (entrainment) and increased steam generation* are generally detrimental since they both lead to steam binding and retarded reflooding and quenching.

Two activities were undertaken to study the upper plenum injection flow distribution. The first was an analytical study to establish the flow regime and flow velocity. This was done to determine if the flow distribution was highly localized or widely dispersed. This analysis was not capable of determining the details of the flow distribution. The average droplet diameter of water injected into the upper plenum was calculated on the basis of the critical Weber number (reference 3). The stable droplet size for a fluid is determined by the balance between the forces attempting to break the droplet up (aerodynamic and mechanical forces) and the surface tension which acts to hold the droplet together. The Weber number is a dimensionless group which includes the surface tension to

*Heat transfer in the upper part of the core due to UPI effects does not help the peak cladding perspective; instead, the increased steaming retards the reflooding rate.

represent the constructive force and a momentum term (ρV^2) to represent the destructive forces:

$$We = \rho_g \, (V_g - V_f)^2 \, d/\sigma$$

where:

ρ_g = gas density; V_g = gas velocity; V_f = fluid velocity;

d= droplet diameter; and σ = surface tension.

Experiments with many gases and fluids indicate that the critical Weber number can generally be used to characterize the atomization process. The breakup of a liquid flow in a gas environment is also influenced by the injection nozzle design; and some nozzle designs can inhibit the breakup of the injection flow. Although the specific UPI nozzle design and piping bends have not been studied in detail, the presence of structures in the upper plenum tends to make the Weber number analysis applicable. The value of critical Weber number above which non-viscous, fluid droplets will break up tends to be in the range of 5 to 20. For this example a typical value of 12 has been chosen (see Reference 3). For the example in question:

$$12 = .06 \, \frac{lb}{ft^3} \, (0\text{-}60 \text{ ft/sec})^2 \, d/.16$$

$$d = .009 \text{ ft} = .11 \text{ inches} \qquad (\text{ at } 45 \text{ ft/sec } d{=}.19 \text{ inches})$$

This analysis indicates that even if there were no structures in the upper plenum, the aerodynamic forces, in this case interfacial friction, would be sufficient to break up the upper plenum injection flow into dispersed droplets with an average droplet size of .11 inches. The inclusion of structures in the upper plenum will accelerate the breakup and dispersal process. Although this analysis does not establish any details of the upper plenum injection flow distribution, it does indicate that a high liquid velocity (~ 45 ft/sec) results in a dispersed droplet flow which favors a widely dispersed rather than a highly localized flow distribution.

A review was begun to find applicable experimental data on flow into a geometry similar to the upper plenum in order to verify and quantify the above conclusions. Through the efforts of the Division of Reactor Safety Research we were able to obtain data from Kraftwerk Union (KWU) on air-water flow distribution in a KWU upper plenum geometry (reference 4). These tests were performed on a full scale mock-up of a 180° sector of a KWU upper plenum. Table 4 shows a comparison of the test geometry with a two-loop upper plenum. Water was injected into the upper plenum at various flow rates through an injection pipe located on the bottom of the hot leg. This arrangement is used in the KWU emergency core cooling systems. The air in the upper plenum was stagnant and no attempt was made to model possible steam-water interactions. These tests therefore establish the unperturbed upper plenum injection flow distribution which could be changed in a steam-water system.

Five tests were conducted by KWU with liquid injection velocities varying from 11 ft/sec to 29 ft/sec. For each test, static pressures were measured at over one hundred locations in the upper plenum by means of vertical Pitot tubes. The static pressures in the upper plenum are indicative of the amount of water delivered to each location. The results of these tests are summarized in Figure 4.

The results indicate that a substantial fraction of the upper plenum has water delivered to it. Figure 4 summarizes the results of all five tests by presenting the estimated percentage of the full upper plenum receiving water from one injection nozzle vs. injection velocity. The estimates of the percentage of the upper plenum receiving water is somewhat complicated by changes in free area (i.e., area outside the guide tubes and support columns) at different locations in the upper plenum. Upper and lower bounds on the results are shown on Figure 4 to account for this effect. An upper bound has been drawn on Figure 4 to indicate the trend with increasing injection velocity and to extrapolate the data to the range of values of injection velocities for the two-loop plants. On the basis of the available experimental and analytical information, we believe that a reasonable upper bound for an upper plenum injection flow distribution for two-loop plants is a uniform delivery to 50% of the upper plenum from one injection nozzle. This distribution will be used for each of the two injection nozzles in Section V of this report in which the overall effect of upper plenum injection is assessed. Also, this provides the basis for our conclusion that UPI will not reach the upper core region as a narrow jet of liquid.

Available Research on Flooding and Entrainment

Given that the data show rather wide dispersal of the UPI into small droplets, the next body of experience examined was the interaction of upflowing steam (from the core) with these droplets. If the droplets are carried up, and out the hot legs, this would be disadvantageous on two counts. First, this is the primary mass addition to the vessel for large breaks; if the UPI does not reach the lower plenum, the core reflooding rate will stop, then regress. Secondly, any liquid entrained in the upper plenum and carried to the steam generator will vaporize there. The flow rate of this steam will create additional pressure losses which further retards flooding rate. Thus it is important to consider the interactive processes.

Flooding and entrainment will be discussed together because the two phenomena are closely related. Flooding is the term applied to the phenomena encountered when the downward flow of water (or any liquid) is impeded by an upward flow of gas. The "flooding limit" (which is a function of gas velocity) refers to the maximum rate of liquid downflow allowed by the gas. At a sufficiently high gas flow rate, no liquid will be allowed to flow down. Entrainment is a related phenomenon in which the force exerted on a liquid by a gas is sufficiently great for the gas to carry off liquid droplets. Flooding could be important for two-loop plants since steam exiting the core could impede the progress of ECC water from the upper plenum to the core and lower plenum. Entrainment could also carry water into the steam generators via the hot legs. Entrained water in the steam generator would vaporize and increase steam binding.

The flooding phenomenon has been studied for several years and the staff has closely followed the experimental and analytical work in this area and has applied the results of this work to other reactor safety problems such as PWR accumulator bypass and BWR core spray flooding. Although the flooding phenomenon was looked at by the staff in connection with upper plenum injection, it was found that entrainment of upper plenum injection water into the steam generator was the overriding consideration in terms of upper plenum steam-water interactions. Therefore, the discussion of available research data will be limited to the entrainment phenomenon.

Extensive data on entrainment are available. Fifteen experimental and analytical studies were reviewed in determining an appropriate method to treat entrainment for two-loop plants. Reference 5 (Ross) presents the results of entrainment tests with steam and water in a three-inch test section. This reference also presents a review and summary of the work on entrainment by: Wallis and Steen; Kutateladze and Sorokin; Cousins and Hewitt; Van Rossum; Paleev and Filippovich; Wallis; Wicks and Dukler; Gill, et al.; and Simpson. Entrainment work by Dartmouth College (Porteous and Richter) (reference 6) and General Electric (Reference 4) were also reviewed. The three most useful studies relative to two-loop plants are discussed below.

The Ross data for vertical entrainment was chosen because the tests were with steam and water and because it includes data for dispersed droplet flow. In addition, the correlation of entrainment fraction vs. momentum flux ($\bar{\rho}\ V_g^2$) was found to be useful for application to the two-loop plant. Figure 5 is a plot of the entrainment fraction vs. modified momentum flux. The data for two different geometries are included in Figure 5 and that the correlation with modified momentum flux appears to be geometry independent. The onset of entrainment occurs at a value of:

$$\bar{\rho}\ V_g^2 = 40$$

where: $\bar{\rho} = \rho_g\ (1+E\ .\ W_f/Wg)$

V_g = gas velocity

E = entrained fraction

W_f = entering liquid mass flow rate

W_g = entering vapor mass flow rate

At 30 psia, the onset of entrainment occurs at:

$$V_g = (40/\rho_g)^{1/2}$$

$$V_g = (40/.073)^{1/2}$$

$$V_g = 23.0 \text{ ft/sec}$$

Entrainment increases rapidly as the gas velocities are increased above the critical velocity for the onset of entrainment. This trend is seen in all of the data. A linear fit to the data on Figure 5 was used for entrainment fractions up to .3.

A second method of determining the critical velocity for the onset of entrainment was studied. This method is based on a force balance between the gravitational force and interfacial friction and has been used in several areas.

Gravitational Force = Frictional Force

$\rho_f \pi d^3 g/6g_c = C_D \rho_g Vg^2 \pi d^2/2g_c 4$

Where C_D = droplet drag coefficient, typically 0.4

The droplet diameter can be determined based on the Weber number, so that the critical velocity for entrainment can be written as:

$$V_{gc} = \left[\frac{4}{3} \quad \frac{\rho_f}{\rho_g}2 \quad \sigma \frac{We}{CD} \quad gg_c\right]^{1/4}$$

at 30 psia, and a Weber number, We = 12:

V_{gc} = 36.8 ft/sec

The values of critical velocity for entrainment of 23.0 ft/sec and 36.8 ft/sec are both relatively low compared to some of the other available data. This is primarily due to the droplet flow regime.

The third study used for the modeling of a two-loop plant during a postulated LOCA was the Dartmouth College (Porteous and Richter) study on horizontal entrainment in a scale model upper plenum. These tests were done in air and water and the entrainment fraction was measured as a function of air velocity. The water was introduced into the upper plenum from above (the tests were primarily intended to model the Westinghouse upper head injection system) and entrainment occurred when droplets were stripped from a thick film of water. The onset of entrainment occurred at approximately 47 ft/sec. As expected, the value is somewhat higher than the tests done with dispersed droplets. These tests lead to three important conclusions: first, the mechanism for horizontal entrainment is essentially the same as for vertical entrainment; second, the inclusion of structures to model guide tubes and support columns does not necessarily result in de-entrainment and might even increase entrainment slightly; third, introducing prewetted air (air already carrying some entrained water) into the model upper plenum resulted in the equal or greater entrainment. The first conclusion allows horizontal entrainment to be treated with the same kind of model developed for vertical entrainment, and the second conclusion eliminates the need for a complex entrainment/de-entrainment model for the upper plenum.* The third conclusion allows the steam generated from the bottom reflood, which is already carrying a significant amount of water, to be treated the same (for entrainment purposes) as the steam generated in the core by vaporization of the upper plenum injection water.

Having studied the above information, it has been concluded that sufficient analytical and experimental information exists to establish a conservative model for entrainment of the upper plenum injection water for a two-loop plant. The details of the staff's treatment of upper plenum entrainment based on the above data is discussed in Section V of this report.

Available Research on Heat Transfer; Steam Generation and Fuel Rod Quench Characteristics

Following a discussion of distribution of UPI, and interaction of UPI with up-flowing steam, we considered the interaction with the heated core. If that fraction of the liquid that falls downward

*The NRC is cooperating with the Federal Republic of Germany to develop a realistic upper plenum simulator. These results will not be available for several years.

through the core is heated and vaporized there, then two things happen. There is earlier quenching of the upper parts of the core, perhaps preferentially where the UPI water is delivered. Although this is generally beneficial, it may not result in a reduction in PCT. In addition, this core heat transfer is another source of vapor to augment the upper plenum entrainment and carryout process. On the other hand, the energy extracted by upper core quenching would have been removed in the old model also (bottom flooding). Double accounting is not needed; rather, it is the time-sequence that is changing. For these reasons we examined the new information on core heat transfer.

As previously stated, the amount of steam generated in the core due to vaporization of upper plenum injection water is significant in determining the effects of a postulated LOCA. In order to establish how much steam is generated in the core, we reviewed several sources of data for heat transfer coefficients and fuel rod quench characteristics for top injection tests.

The FLECHT SET Phase A top injection tests (reference 7) were reviewed. These tests were performed by injecting subcooled water into the upper plenum of a test vessel which contained a 10 x 10 array of electrically heated rods 12 feet long, in PWR geometry. Seven successful tests were run with various initial rod temperatures, various injection water temperatures, various rod powers and two different flow rates. In six of the tests, steam was vented from both ends of the test section and, in one test, steam was only vented from the top of the test section. Figure 6 presents the results for two of the FLECHT SET Phase A tests (tests 5703 and 6007). Test 5703 is typical of the tests with the bottom vent opened. As indicated by the figure, the top injection water is able to remove the simulated decay heat and cool the rod at a rate of approximately 1°F/sec. At the end of 630 seconds the rod temperature was 1200°F. The heat transfer coefficient for this test was 10 BTU/hr-ft^2-°F. For test 6007, the bottom vent was closed and water was therefore allowed to accumulate in the bottom of the test bundle. This resulted in better heat transfer and the fuel rods were cooled at approximately 2°F/sec. This corresponds to a heat transfer coefficient of approximately 15 BTU/hr-ft^2-°F. The six foot elevation (midplane) quenched at about 620 seconds for this test. Test 6205 (with bottom venting) was run with an increased top injection flow rate and the results indicate somewhat improved heat transfer (~ 15 BTU/hr-ft^2-°F) and earlier quench of the bundle midplane (280 seconds). Since the tests with the bottom vent opened did not measure the steam flow to the atmosphere, there are no accurate measurements of steam production in these tests. The difference in the mass of water injected during the test and the mass of water collected following the test gives an indication of the total steam production. In general, 10% to 20% of the injected water was converted to steam for the tests with an injection flow of 15 GPM. No mass balance was available for the 35 GPM test. This data has been included in the heat transfer model developed by the staff.

The Westinghouse upper head injection low pressure, refill heat transfer tests (reference 8) were also reviewed. These data are Westinghouse proprietary and show quench times and quench superheat for top injection heat transfer tests. These tests were run at the G-2 Test facility (shown in detail in reference 8) which includes a 19 x 19 array of 336 electrically heated rods and 25 unheated rods. The heated length of the rods is 164 inches. The rod size, pitch and space grid design are typical of a Westinghouse 17 x 17 fuel assembly design. Tests were performed with various initial rod temperatures; injection water subcoolings; pressures; and injection flow rates. Each test was run for approximately 60 seconds and the quench wall superheat (that is, the wall temperature minus the saturation temperature at the time of quench) was measured for those locations which quenched and the percent of quenched rods at each of 12 axial locations was also reported. The following conclusions can be drawn from the results of these tests:

1. The low power sections of the rod quenched first (i.e., the top and bottom) and the quench front progressed steadily in both directions;

2. In the test period of 60 seconds only a few locations quenched - usually between 8% and 24% of the total bundle;

3. The amount of quenching and therefore the heat transfer increased with increasing injection flow;

4. The amount of quenching and therefore the heat transfer increased with increasing injection water temperature; and

5. The amount of quenching and therefore the heat transfer increased with increasing pressure.

The first four conclusions are consistent with the results of the seven FLECHT SET Phase A top injection tests. The effects of pressure was not seen earlier because the FLECHT SET Phase A tests did not include significant pressure variation. These tests therefore confirm the trends seen in the FLECHT SET Phase A tests and extend the range of flow rates and injection water temperature. The results of these tests and the FLECHT SET test are plotted on Figure 7 as quench time for the bundle midplane vs. top injection flow rate. For the G-2 tests, the midplane quench times were estimated from the rate of quench observed during the 60 second test period. The cooldown rates in these tests are similar to those observed in the FLECHT Set Phase A top W proprietary injection tests, that is 1°F/sec to 2°F/sec. This indicates that the injection water is removing only slightly more than decay heat.

The upper plenum injection tests performed on semiscale (S-05-3, S-05-4, S-05-7) references 9-14 were reviewed relative to the fuel heat transfer and quench performance. The integral systems effects observed in these tests will be discussed in the section on available research on system simulation (III-4). Semiscale is a two-loop PWR model including 36 electrically-heated rods 5.5 feet long in a PWR geometry. This facility has been used to study the integral (blowdown, refill and reflood) performance of model PWR under simulated LOCA conditions. Tests S-05-3 and S-05-4 were part of the alternative injection study for double-ended cold leg breaks and included upper plenum accumulator injection in addition to cold leg injection. Test S-05-7 was specifically run at our request for application to Westinghouse two-loop plants. Test S-05-7 therefore attempted to match two-loop plant parameters and included upper plenum, low pressure pumped injection. The heat transfer coefficient at each thermocouple location was calculated from the measured clad and fluid temperatures. These tests confirm the previously established behavior under transient conditions. The heat transfer coefficients prior to quench were approximately 10 to 15 BTU/hr-ft^2°F for test S-05-7 and slightly higher values were observed for tests S-05-4 and S-05-3, thus reconfirming the trend of increasing heat transfer with increasing top injection flow rate.

Based on the number of independent tests reviewed and the consistency of the results, we concluded that a reasonable but conservative model for fuel rod heat transfer and quench can be developed by the staff and by Westinghouse or the two-loop plant licensees for application to two-loop plants.

<u>Available Research on System Simulation</u>

In addition to the separate effects tests discussed above (flow distribution, flooding and entrainment and heat transfer), integral tests with upper plenum injection have been reviewed for their applicability to two-loop plants. Three semiscale upper plenum injection tests were reviewed - S-05-3, S-05-4, and S-05-7. The KWU-PKL loop combined injection tests (reference 15) were also reviewed.

Semiscale tests S05-3 and S-05-4 were both upper plenum injection tests with 16 GPM and 8 GPM, respectively, injected by the vessel accumulator with a pressure setting of 300 PSIA. In each case, the injection period was from approximately 20 seconds to approximately 150 seconds. Because of the need for parametric variation to understand the two-loop, upper plenum injection problems, a third semiscale test was run with low pressure pumped injection into the upper plenum at an injection rate more typical of two-loop plants. In each case the results were similar: UPI water entered the core while cold leg injection was underway but before recovery of the bottom of the core. Steam was generated in the core by vaporization of the UPI water. This steam was drawn to the cold leg injection water which was the low pressure point in the system due to the subcooling of the cold leg ECC water. As a result, the UPI test generally showed increased ECC bypass and little or no reflooding of the core from below. Figures 8 and 10 illustrate the difference in bottom reflooding with and without UPI. Figure 8 shows the density just below the core for test S-04-6, the base case without UPI. After 50 seconds the density remains relatively high with oscillations about an average value which varies from 30 lb/ft^3 to 60 lb/ft^3. Figure 9 shows that the density at the same location in test S-05-3 oscillates about an average value which varies from 5 lb/ft^3 to 25 lb/ft^3 while UPI injection continues. At 164 seconds UPI terminated in this test and Figure 9 shows an almost instantaneous increase from 10 lb/ft^3 and a continuing trend of increasing density thereafter. The general trend of poor bottom reflooding persists throughout each semiscale test. This is also shown in Figure 10 which illustrates the volumetric flow rate at the core inlet. This figure clearly indicates negative flow at the core inlet until the end of UPI at which time the volumetric flow rate oscillates around zero.

The core quenching characteristics of the UPI tests were also different from the base cases and again show the trend of top to bottom UPI flow and poor bottom reflooding. In each test the core quenched from the top down. In test S-05-4, which injected at the highest UPI flow rate, all of the core locations quenched at the same time or an earlier time than in the base case (S-04-6). For tests S-053 and S-05-7, the top core locations quenched earlier than the base cases (S-04-6 and S-05-6); but the lower elevations which were not directly below the UPI injection point did not quench until later than in the base cases. In fact, some locations (20 to 25-inch elevations) for test S-05-7 never quenched during the test, which was terminated at 300 seconds after rupture.

The results of the semiscale tests can be summarized as follows: UPI resulted in significant net steam generation; good heat transfer was observed in those regions near the injection location; and little or no bottom reflood was observed.

The KWU-PKL loop is a one-loop Pressurized Water Reactor Simulation, operated by Kraftwerk Union, containing 340 electrically-heated rods. This facility has been used to study the refill and reflood performance of cold leg injection and combined injection i.e., upper plenum and cold leg injection ECCS during a simulated LOCA reference (15).

Two series of KWU -PKL loop tests were reviewed by the staff. Both series studied combined injection vs. cold leg injection only. The top injection flow rates in both series of tests was significantly greater than the two-loop upper plenum injection flow rate on a per bundle basis. In this range of injection rates, the increased flow results in a reduction in the net steam generation associated with upper plenum injection, since the injection of additional subcooled water causes a significant increase in steam condensation without causing a significant increase in core heat transfer. As a result, the reflood rates and the peak clad temperatures from the KWU-PKL tests are not representative of two-loop plants. The first series of tests was conducted with an experimental facility containing a single external downcomer. These tests were not published but discussions among DSS, Reactor Safety Research (RSR) and Kraftwerk Union indicate that the results were similar to the semiscale results, that is, persistent down flow through the core and prolonged ECC bypass. The second series of tests was conducted with two external downcomers (see Figure 11) in an attempt to model a realistic view of the steam and water flow patterns during refill. One of the two downcomers was designed to allow steam flow out of the lower plenum to the cold leg break. The other downcomer could allow downward flow of cold leg injection water. Whether this picture of separated steam and water flow is correct for a full size PWR is not known. There is no large scale data to support this view and small scale data up to 1/5 scale generally indicate symmetric delivery (or nondelivery) of cold leg injection water. Some indication of asymmetry exists in the LOFT L1-4 data but better instrumentation is required to confirm and quantify it. The asymmetry in LOFT L1-4 was not nearly as extreme as in the PKL-KWU tests.

Although the two downcomer KWU-PKL loop tests showed accelerated core quenching from both above and below with corresponding lower values of peak clad temperature, it was concluded that the data may not be typical of two loop plant ECCS systems behavior because:

(1) These tests injected significantly more water from above the core on a per bundle basis in comparison to the two loop plants; and

(2) The two downcomer arrangement allowed more downcomer penetration than has been measured in other available ECC bypass tests.

The staff concludes that the Semiscale tests and other small scale data provide a suitably conservative basis for use in appraising the Westinghouse two loop PWR evaluation models.

IV. Views of the Two-Loop Plant Licensees

The subject of two-loop plant ECCS performance during a postulated LOCA has been discussed with the two-loop plant licensees and with Westinghouse. Their views have been expressed at a meeting with the staff on January 26, 1977 and in a subsequent submittal on the subject by Northern States Power Company (reference 2). The licensees' views appear to be consistent with the views presented by Westinghouse on this subject. These views can be summarized as follows:

1. The evaluation model is physically unrealistic in that the low pressure injection water in the model is added to the cold leg injection rather than to the upper plenum;

2. The effects associated with adding subcooled water to the upper plenum are small relative to those phenomena controlling the refill and reflood process;

3. Both conservative and nonconservative aspects of upper plenum injection are not included in the evaluation model and the net effect leads to an overall conservative model.

4. Model development to include the effects of upper plenum injection is not needed. Based on the model development experience

with the upper head injecton system, this course of action could require two years or more; and

5. Modification of the plant ECCS to eliminate upper plenum injection would take two years to implement and single failure considerations make the safety of this approach questionable.

The key element in viewing the effects of upper plenum injection is the upper plenum injection flow distribution. The flow distribution is the controlling factor relative to steam generation, steam condensation and upper plenum injection water entrainment. The two-loop plant designer (Westinghouse) and the plant licensees description of the upper plenum injection flow distribution during blowdown, refill and reflood is shown in Figures 12 through 18.

These figures are from reference 2 and are an "artist's conception" of a highly localized injection flow which has little or no interaction with steam generated in the core either by bottom reflood or by vaporization of upper plenum injection water. The conclusions by Westinghouse and the licensees concerning the effectiveness of upper plenum injection depend on this view of a highly localized injection flow.

V. Regulatory Analysis

The inclusion of upper plenum injection significantly increases the complexity of analyzing a postulated LOCA. Sufficient analytical and experimental information exists to reasonably model the important separate effects of: upper plenum flow distribution; heat transfer and quench; and, flooding and entrainment.

Our review of the available information on upper plenum flow distributions indicates that the view of upper plenum injection as a highly localized flow (as presented by Westinghouse and the two-loop plant licensees) is not correct and is non-conservative. We conclude that the data supports the concept of a widely dispersed droplet flow in the upper plenum. For this condition, an integral model which treats the interactions among the controlling phenomena (i.e., heat transfer and quench; steam generation; entrainment and flooding; reflood rate) is required to establish the important assumptions; to identify the proper sensitivities to parameter changes; and to assess the overall effect on peak clad temperature. For these reasons we have constructed a simple, quasi-static model to study the bounding effects of upper plenum injection, and the sensitivity to various assumptions. In this model the reflood rate is calculated by adjusting the current evaluation model calculation of reflood rate to account for the additional steam generation associated with UPI.

The elements of the staff model and their interactions are shown in Figure 19. The procedure used in this model is summarized in the following steps, where each step is performed as a function of initial power level:

1. Establish an upper plenum flow rate, subcooling and flow distribution.

2. Determine the decay heat and stored energy in the fuel.

3. Determine the heat transfer and quench rate from the upper plenum injection flow.

4. Determine the amount of vaporization of upper plenum injection water by the heat added to the water and the initial subcooling of the water.

5. Determine the entrainment of upper plenum injection water by calculating - the momentum flux of steam and water exiting the core and then finding the corresponding entrainment fraction from a correlation of entrainment fraction vs. modified momentum flux.

6. Determine the net reflood rate based on the core steam generation, entrainment and the steam relieving capability of the flow path from the upper plenum to the cold leg break.

7. Determine peak clad temperature from the FLECHT bottom reflood sensitivities of peak clad temperature vs. reflood rate.

By applying the staff model, for steam generation and effective reflood rate at many time points during the reflood, an approximation to a dynamic reflood model can be achieved. The staff model is typically used at time increments of approximately two seconds.

At this point, a penalty in terms of increased peak clad temperature is known as a function of power level. In order to determine the sensitivity of the model to changes in peaking factor, power level and other assumptions, the following steps are undertaken which offset the increased peak clad temperature by giving credit for: a reduced peaking factor, Fq (2.0 vs. 2.32); a decay heat curve of ANS x 1.0 rather than ANS x 1.2; new research data on Zirc-Water reactions (reference 16); and reduced power level, as appropriate.

At present the relationship between peaking factors necessary to Appendix K calculations and the best-estimate range of peaking factors (without load following) is:

Facility	Technical Specification Fq and EM Peak Clad Temperature	Best-Estimate Fq for Steady State Operation
Point Beach 1-2	2.32 (=1965°F)	1.55-1.82
Prairie Island 1-2	2.32 (2187°F)	1.55-1.90
Kewaunee	2.25 (2172°F)	1.55-1.90
Ginna	2.32 (1957°F)	1.55-2.00

As seen from this table, the two-loop plants can be operated at significantly lower peak linear heat rates compared to the peak linear heat rate used in the present LOCA analysis.

The following section describes the individual elements of the staff model and identifies those areas of conservatism which could be changed on the basis of additional experimental or analytical work.

Upper Plenum Injection Flow Distribution Model

Our model assumes that the upper plenum injection flow from each injection nozzle is uniformly distributed to one half of the upper plenum and therefore to one-half of the core. This is based on the KWU data described above and depicted in Figure 4. Based on the staff model the worst case has been determined to be injected from both nozzles covering the whole core. Although the data indicate that the assumption that one-half of the core is covered by injection water from a single nozzle is slightly conservative, an additional conservatism exists in that the distribution of water from the two injection nozzles is assumed to be uniform over the entire core. The data indicate that more water is delivered in the center of the region covered by the water and that the amount near the upper plenum periphery is significantly less. A nonuniform delivery of upper plenum injection water in the model could be advantageous in reducing the amount of entrainment into the steam generator. In order to modify the assumption of uniform delivery to one-half of the core from each injection nozzle, additional experimental data on a two-loop upper plenum geometry would be required or a model based on the physical phenomena (i.e., not empirical) which could bound the KWU data would be required.

Decay Heat and Stored Energy in the Fuel Model

The decay heat used in the calculation is based on the ANS Standard; both the nominal value and the nominal value plus 20%, i.e., in accord with Appendix K to 10 CFR Part 50 were studied. The initial stored energy in the fuel is assumed to be a linear function of power level. The full power initial stored energy is the value used by Westinghouse in the Point Beach evaluation model.

Heat Transfer and Fuel Rod Quench Model

The heat transfer and fuel rod quench model is based on the previously described data from FLECHT Set Phase A, upper head injection low pressure quench data and semiscale tests S-05-3, S-05-4 and S-05-7. The heat transfer model assumes a quench front progression from the top and bottom of the core. The quench front progression rate is assumed to be a function of upper plenum injection rate. Figure 20 shows the quench rate model and the data which form the basis for this model. Since rapid quench increases the amount of steam generation and therefore impedes bottom reflood, the model is based on a lower bound of the applicable data. That is a conservative application of the data.

The heat transfer model takes the following form. In the unquenched portion of the core, the heat transfer is modeled as a factor times decay heat, that is, a factor of 1.0 to account for removal of decay heat and an additional factor of .3 to account for the cooldown of the fuel. The factor of .3 is based on a cooldown rate of 2°F/sec which is an upper bound for the data. In the portion of the core which has been quenched, only decay heat is removed. This heat transfer and quench model only applies to the region of the core which has upper plenum injection water delivered to it. This model is not applied to the hot channel.

Steam Generation Model

The steam generation resulting from the vaporization of upper plenum injection water in the core is calculated by an energy balance. The energy removed from the fuel is first used to raise the temperature of the injection water to saturation temperature and any additional energy is used to vaporize a portion of the injection water.

Entrainment Model

The staff model for entrainment contain three addictive parts. One part accounts for the entrainment of bottom reflood water. This model uses the Westinghouse calculated values. The second part of the model accounts for vertical entrainment caused by vaporization of upper plenum injection water in the core. The third part of the model accounts for horizontal entrainment of upper plenum injection water delivered near the hot legs.

The model used by the staff to account for entrainment caused by vaporization of upper plenum injection water is based on the Ross work described in Section III of this report. The onset of entrainment is calculated to occur at a modified momentum flux of 40(lb/ft-sec^2)

$$\bar{\rho}\ Vg^2 = 40$$

$$\text{where } \bar{\rho} = \rho g\ (1 + E\ Wf/Wg)$$

At higher values of modified momentum flux the entrainment flow is approximated as:

$$W_{ent} = 1.67 \times Wg - 80$$

This correlation is based on a best fit of the Ross data at small values of entrainment fraction.

The model used by the staff to account for the entrainment of upper plenum injection water which is delivered near the hot legs is based on the Dartmouth College data described in Section III of this report. Complete horizontal entrainment of the droplets in the upper plenum is assumed to take place for those regions with horizontal velocities equal to or greater than 60 ft/sec. The horizontal velocity profile of the upper plenum is based on the air flow tests from the Stade Nuclear Power Plant (reference 17). The data from these tests are shown in Figure 21. Figure 22 presents the same data in the form of "Fraction of Upper Plenum Area" vs. "Air Velocity."

On the basis of this data, only 1.6% of the area in the upper plenum would experience velocities above the critical velocity for entrainment. Therefore, on the basis of a uniform distribution of upper plenum injection water, 1.6% of the 240 lbs/sec injected from each low pressure injection system will be entrained by this mechanism.

The treatment of entrainment in the staff model is relatively simplistic, the treatment of entrainment as three separate phenomena is somewhat arbitrary. Large-scale integral tests would provide a much better data base for this model. Large-scale entrainment/de-entrainment tests are presently planned by Kraftwerk Union and the German Government. U.S. involvement in these tests includes instrumentation development and analytical modeling. The results of these tests, in the early 1980's could help significantly in understanding and modeling the entrainment phenomena associated with upper plenum injection.

<u>Total Steam Generation Due to Upper Plenum Injection</u>

The total steam generation due to upper plenum injection consists of the sum of the following components: horizontal entrainment of upper plenum injection water (all water entrainment into the steam generator is assumed to be vaporized); vaporization of upper plenum injection water in the core; and, vertical entrainment of upper plenum injection water by the core steam (due to upper plenum injection vaporization) exiting the core. Figure 23 presents each of these components and the total steam generation as a function of initial core power level at a point in time near the beginning of reflood.

Net Reflood Rate Model

The staff reflood model is based on a simple momentum balance and assumes that, for a given water level in the core during reflood, the steaming rate at the break is a fixed value such that:

ΔP, Elevation head (downcomer - core) = ΔP, flow resistance (hot leg to break)

$$\frac{\Delta h\ \rho (PSI)}{144} = \frac{KW^2\ Total (PSI)}{A^2 2 g_c \rho\ 144}$$

and that for a given reflood rate, the amount of reflood steam and water carried to the steam generators is:

Carryover of Reflood Steam + Water = Reflood Rate (in/sec) x 12 x density x core area x CRF

Since the total steaming rate is fixed at a given point during reflood, the bottom reflood rate is directly reduced when steam is generated by the upper plenum injection process. This means that each pound per second of steam generated by the upper plenum injection must be compensated for by a reduction of one pound per second from the bottom reflood.

During the reflood process, the reflood rate decreases as the level in the core increases. This is due to the associated decreasing difference between the downcomer level and the core level and a decreasing differential pressure. The reflood rate for any given transient is, therefore, not a constant. For the purpose of calculating an effective reflood rate due to additional steam generation, the reflood rates will be characterized as an unperturbed reflood rate and a perturbation due to UPI. The unperturbed reflood rate is from the base case, which is the evaluation model calculation and includes no steam generation due to upper plenum injection.

Change in Peak Clad Temperature Due to Upper Plenum Injection

The staff model assesses a peak clad temperature penalty associated with the calculated reduction in bottom reflood rate. Inherent in this treatment is the assumption that no credit is given for increased heat transfer in the hot channel due to top injection. This assumption is used because of the large uncertainty which exists relative to the distribution of water in the upper plenum. The increase in peak clad temperature with decreasing reflood rate is taken from the FLECHT reflood rate studies (Figure 3-26) in reference 18. Figure 24 presents the results of those studies for the case which showed the greatest sensitivity to decreasing reflood rate. The FLECHT experiments were performed with stainless steel heater rods. The increase in clad temperature associated with the decreased reflood rate does not include any Zirconium water reaction. The use of this data therefore only accounts for the additional temperature rise associated with a reduced heat transfer coefficient and a prolonged exposure with steam cooling only. The use of this curve is appropriate since the staff model will not be used to calculate peak clad temperatures above 2200°F but will only be used to determine the reduction in peaking factor and/or power level required to maintain a peak clad temperature less than or equal to 2200°F.

Result of Staff Model

The model described in the preceding section was applied to a typical Westinghouse two-loop plant. The calculated increase in peak clad temperature due to vaporization and entrainment is based on the hypothetical case assuming that the Westinghouse evaluation model results in a calculated peak clad temperature of 2200°F with a peaking factor, Fq, of 2.32. The results of these calculations are presented in this section.

The staff model assesses a penalty on peak clad temperature to account for the generation of steam due to upper plenum injection. Figure 25 presents the peak clad temperature penalty as a function of initial power level. At approximately 50% power, the heat being removed from the core can no longer be absorbed completely by the initial subcooling of the upper plenum injection water and the result of steam generation in the core can be seen in the figure. At 92% power, entrainment of upper plenum injection water begins and the penalty on peak clad temperature increases rapidly. This rapid increase is a reflection of the rapid increase in entrainment which occurs after the onset of entrainment.

In order to determine the effect on power level of this penalty, the following assessments were made. The decrease in the calculated peak clad temperature associated with the following were estimated:

(1) operation at reduced power level; (2) operation with a peaking factor less than 2.32; and (3) use of new research data on decay heat and zirconium-water reactions. The decrease in peak clad temperature associated with these assumptions is shown in Figure 25. The circles on Figure 25 indicate the power level at which the increase in peak clad temperature due to UPI is exactly balanced by the decrease in peak clad temperature. The results are summarized below:

Case	Assumptions
1	Appendix K, decay heat and Zirc-Water reaction assumptions and a peak to average power of 2.32.
2	Appendix K, decay heat and Zirc-Water reaction assumptions and a peak to average power of 2.0.
3	Appendix K, decay heat and Zirc-Water reaction assumptions and a peak to average power of 1.8.
4	Credit for new decay heat and Zirc-Water reaction research and a peak to average power of 2.0.

In each case the peak clad temperature was assumed to decrease at a rate of 15°F per percent power reduction. The resulting power levels at which the decreased peak clad temperature equals the upper plenum injection penalty on peak clad temperature are:

77 Percent for Appendix K, decay heat and Zirc-Water reaction assumptions and Fq = 2.32

83 Percent for Appendix K, decay heat and Zirc-Water reaction assumptions and Fq = 2.0

87 Percent for Appendix K, decay heat and Zirc-Water reaction assumptions and Fq = 1.8

102 Percent for New Research data on decay heat and Zirc-Water reaction and Fq = 2.0.

The staff model has been useful in studying the sensitivity of the calculated results of a postulated cold leg LOCA to various modeling and input assumptions.

The following have been identified as important items in determining the effects of upper plenum injection as a postulated, large break, cold leg LOCA:

1. Upper Plenum Injection Flow Distribution
2. Upper Plenum Injection Flow Rate
3. Upper Plenum Injection Subcooling
4. Initial Core Power Level
5. Decay Heat Assumptions
6. Top Injection Heat Transfer and Quench Models
7. Entrainment and Flooding Models
8. Dynamic Modeling of the Effects of UPI on Reflood.

VI. CONCLUSIONS

We conclude that the thermal and hydraulic effects of upper plenum injection are significant in determining the course of a postulated LOCA transient and the associated peak clad temperature. A model which does not adequately represent upper plenum injection is unrealistic (and therefore, does not show the correct sensitivities to break size, break location, etc.) and may be nonconservative.

We conclude that the key element in determining the effectiveness of the two-loop plant upper plenum injection system is the distribution of the injected water in the upper plenum. The description of the upper plenum injection distribution provided by Northern States Power Company in reference 2 states that, "...the water which is injected stays in a highly localized area of the upper plenum." We conclude that this assertion is incorrect and that it has led to an incorrect assessment of the impact of upper plenum injection on a postulated LOCA. We further conclude that the ECCS evaluation model as applied to two-loop plants is non-conservative in the assumption that emergency core cooling water injected into the upper plenum reaches the lower plenum refill and reflood water without adverse effect on the core reflooding rate. Therefore the model does not meet the General Standards for Acceptability required by 10 CFR 50 Appendix K, paragraph II-5:

"Elements of evaluation models reviewed will include technical adequacy of the calculational methods, including compliance with required features of Section I of this Appendix K and provision of a level of safety and margin of conservatism comparable to other acceptable evaluation models, taking into account significant differences in the reactors to which they apply."

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TABLE 1
TWO-LOOP-PLANT
ECCS DESIGN PARAMETERS

Accumulator Design

Number of Accumulators	2
Design Pressure (psig)	800
Design Temperature (°F)	300
Total Volume, ft^3	1750
Minimum Water Volume, ft^3	1000
Minimum Pressure (psig)	700
Injection Locations	Cold Leg

High Pressure Pumped Injection

Number of Pumps	2
Design Pressure (psig)	1750
Design Temperature (°F)	300
Design Flow Rate (gpm)	700
Maximum Flow Rate (gpm)	1230
Injection Locations	Cold Leg and Upper Plenum

Low Pressure Pumped Injection

Number of Pumps	2
Design Pressure (psig)	600
Design Temperature (°F)	400
Design Flow (gpm)	1560
Injection Location	Upper Plenum
Pump Flow at Reduced Pressure (30 psia), (gpm)	1800

TABLE 2

Point Beach 1 + 2

THERMAL AND HYDRAULIC DESIGN PARAMETERS

Total Primary Heat Output, MWt	1524
Total Reactor Coolant Pump Heat Output, MWt	5.5
Total Core Heat Output, MWt	1518.5
Total Heat Output, Btu/hr	5198×10^6
Heat Generated in Fuel, %	97.4
Maximum Thermal Overpower, %	12
Nominal System Pressure, psia	2250
Hot Channel Factors	
Heat Flux	
Nuclear, F_q^N	2.72
Engineering, F_q^E	1.03
Total	2.80
Enthalpy Rise	
Nuclear, $F_{\Delta H}^N$	1.58
Engineering, $F_{\Delta H}^E$	1.02
Coolant Flow	
Total Flow Rate, lbs/hr	66.7×10^6
Average Velocity Along Fuel Rods, ft/sec	15.0
Average Mass Velocity, lb/hr-ft^2	2.37×10^6
Core Flow Area, ft^2	28.2
Coolant Temperature, °F	
Nominal Inlet	552.5
Average Rise in Vessel	57.6
Average Rise in Core	60.0
Average in Core	582.5
Average in Vessel	581.3
Nominal Outlet of Hot Channel	648.9
Heat Transfer	
Active Heat Transfer Surface Area, ft^2	28,715
Average Heat Flux, Btu/hr-ft^2	175,800
Maximum Heat Flux, Btu/hr-ft^2	491,000
Maximum Thermal Output, kw/ft	16.0
Maximum Clad Surface Temperature at Maximum Power (100% power), °F	657
Radially Averaged Clad Temperature At Maximum Power, (100% power), °F	709
Fuel Central Temperatures, °F	
Maximum at 100% Power	3750
Maximum at 112% Power	4000
DNB Ratio	
Minimum DNB Ratio at nominal operating conditions	2.11
Pressure Drop, psi	
Across Core	26
Across Vessel, including nozzles	43

TABLE 3

REACTOR VESSEL DESIGN DATA

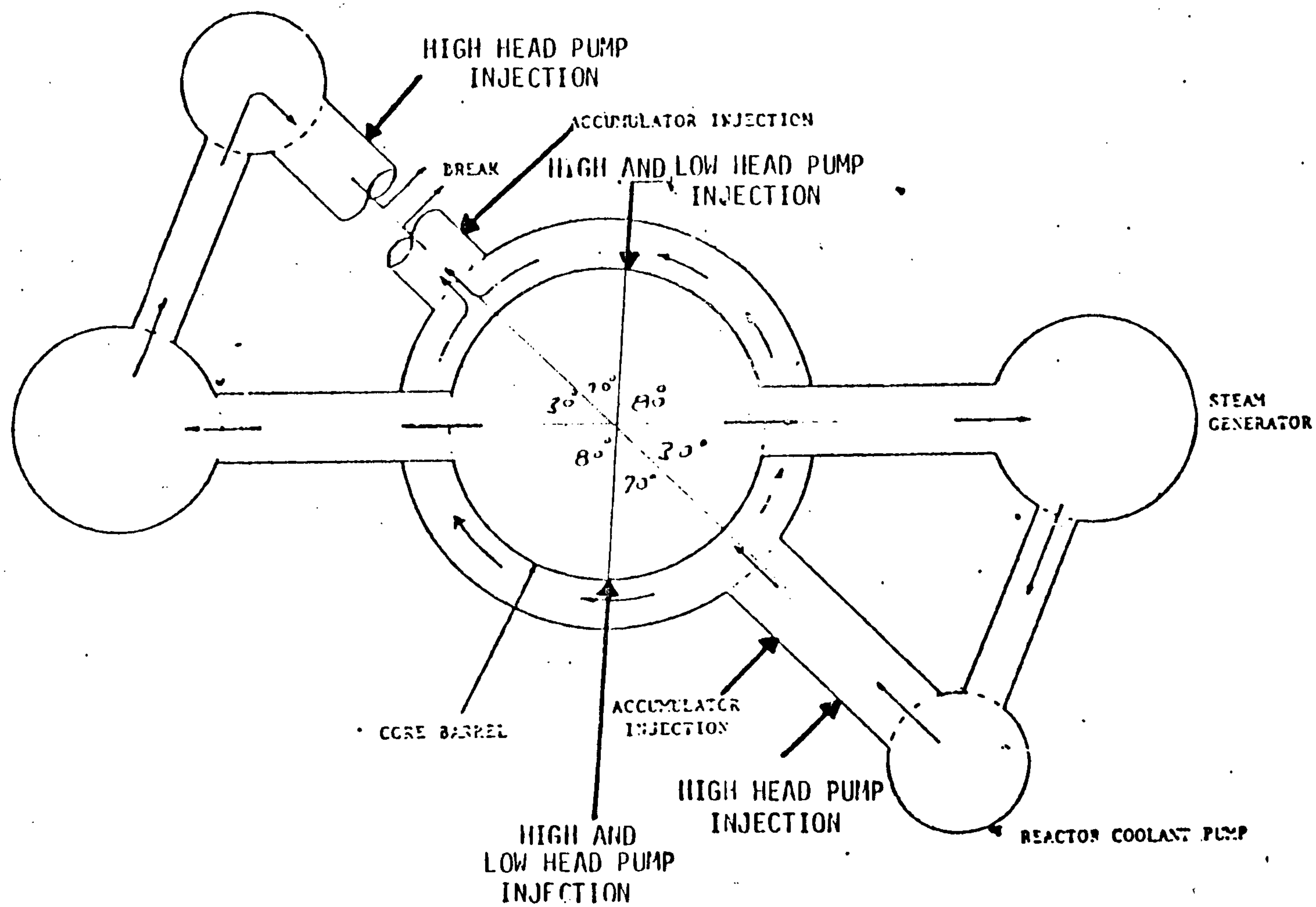
Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure, psig	3110
Design Temperature, °F	650
Overall Height of Vessel and Closure Head, ft-in. (Bottom Head O.D. to top of Control Rod Mechanism Housing)	39.0
Water Volume (with core and internals in place), ft^3	2473
Thickness of Insulation, min., in.	3
Number of Reactor Closure Head Studs	48
Diameter of Reactor Closure Head Studs, in.	6
Flange, ID, in.	123.8
Flange, OD, in	157.3
ID at Shell, in.	132
Inlet Nozzle ID, in.	27.47
Outlet Nozzle ID, in.	28.97
Clad Thickness, min., in.	0.156
Lower Head Thickness, min., in.	4.125
Vessel Belt-Line Thickness, min., in.	6.5
Closure Head Thickness, in.	5.375
Reactor Coolant Inlet Temperature, °F	552.5
Reactor Coolant Outlet Temperature, °F	610.1
Reactor Coolant Flow, lb/hr	66.7×10^6
Safety Injection Nozzle, number/size, in	2/4

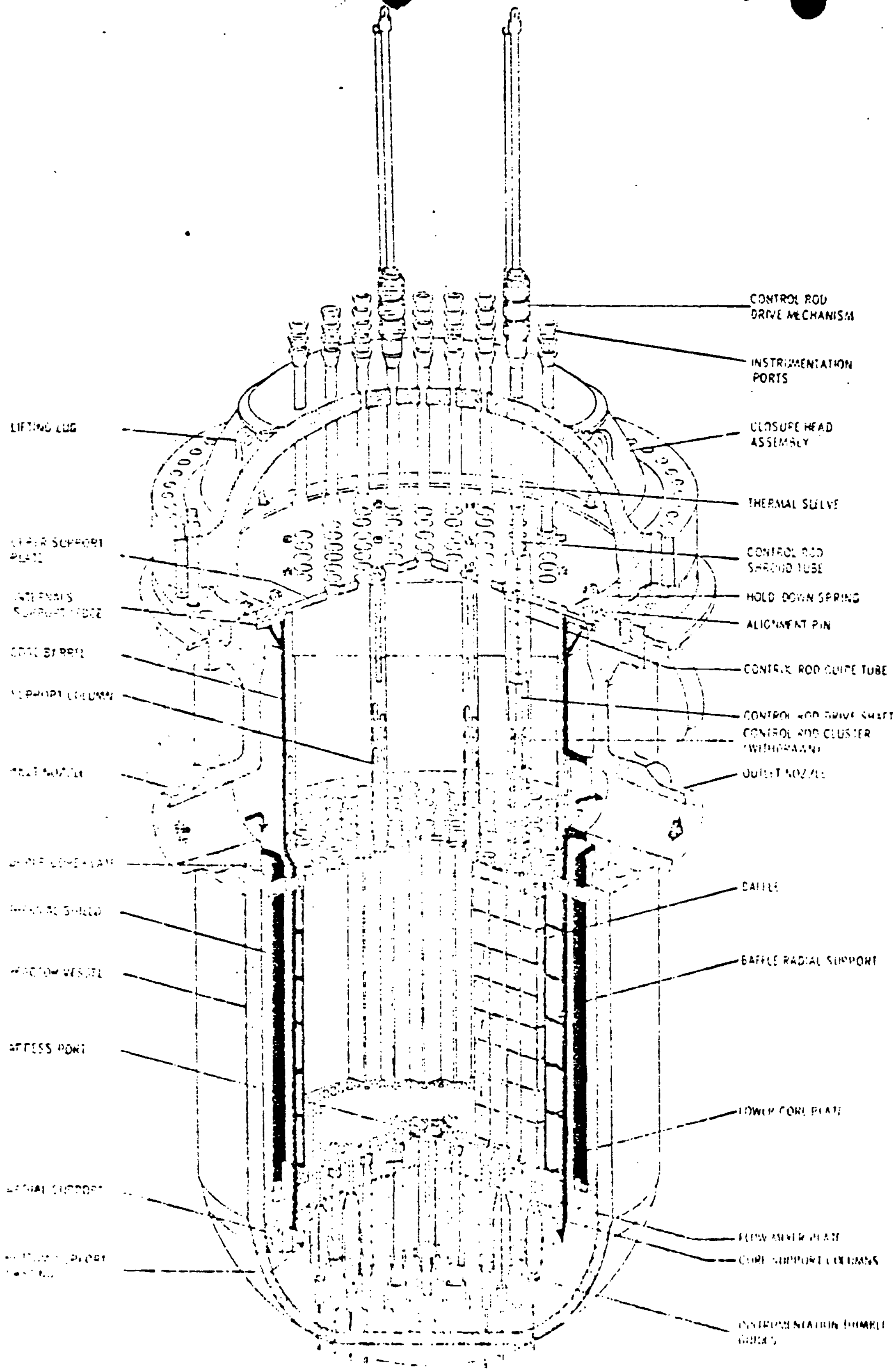
TABLE 4

COMPARISON OF 2-LOOP PLANTS AND KWU TESTS

	2-Loop Plants	KWU Tests
Upper Plenum Diameter	9.1 ft	13.7 ft
Sector Size	360°	180°
Number of Control Rod Guide Tube Locations (for full core)	61 (slotted)	66 (without slots)
Number of support columns (for full core)	12 (slotted)	12 (without slots)
Injection Velocity	45 ft/sec to 60 ft/sec	11 ft/sec 14 ft/sec 18 ft/sec 23 ft/sec 29 ft/sec
Injection Pipe Diameter	4 inches	9 inches

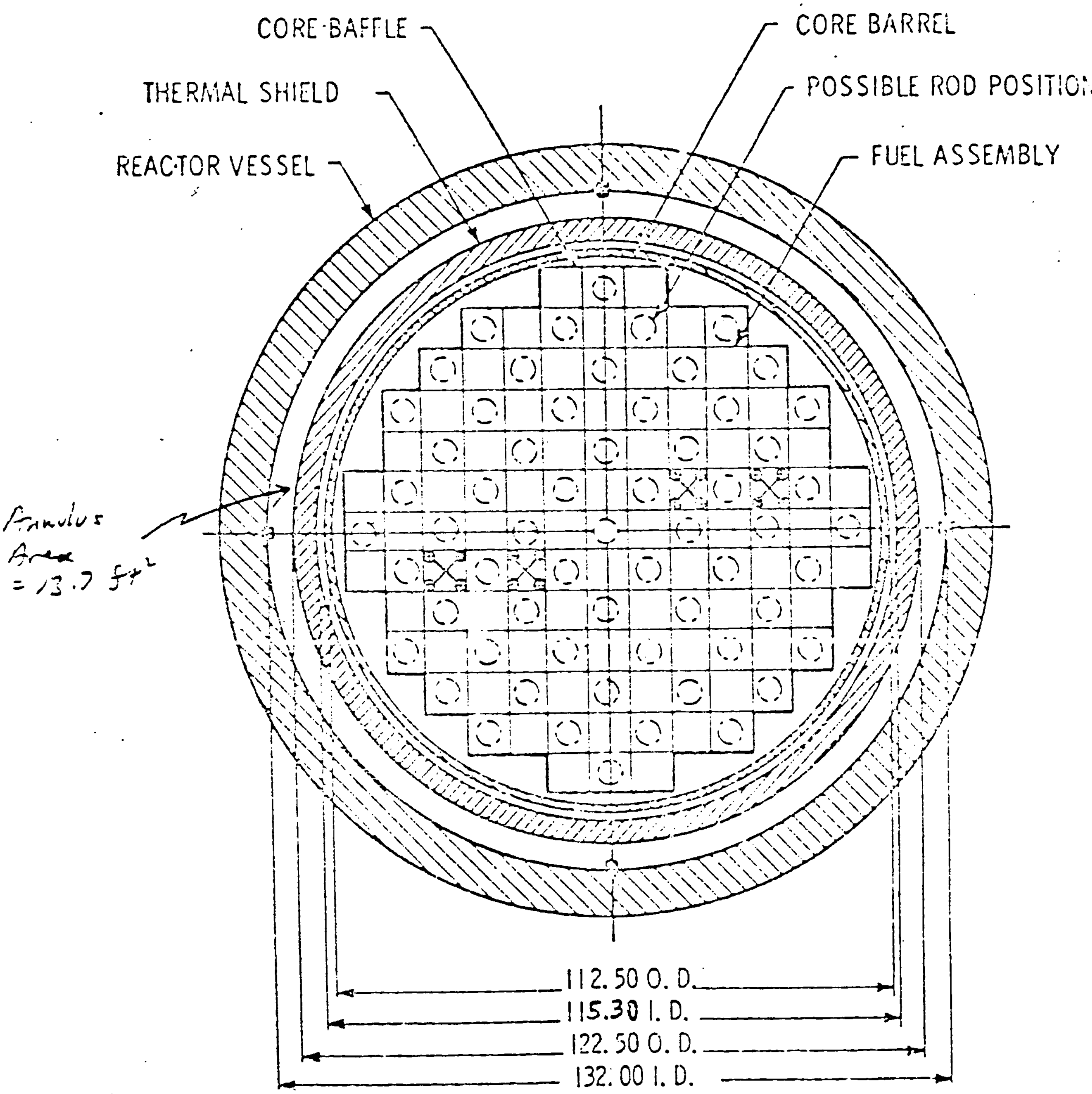
Figure 1
Two-Loop Plant ECCS





REACTOR VESSEL INTERNALS

Figure 2



PLAN VIEW OF CORE 121 FUEL ASSEMBLIES

primary
secondary
-- primary and secondary source assembly

-- secondary source assembly

REACTOR CORE CROSS SECTION

Figure 3

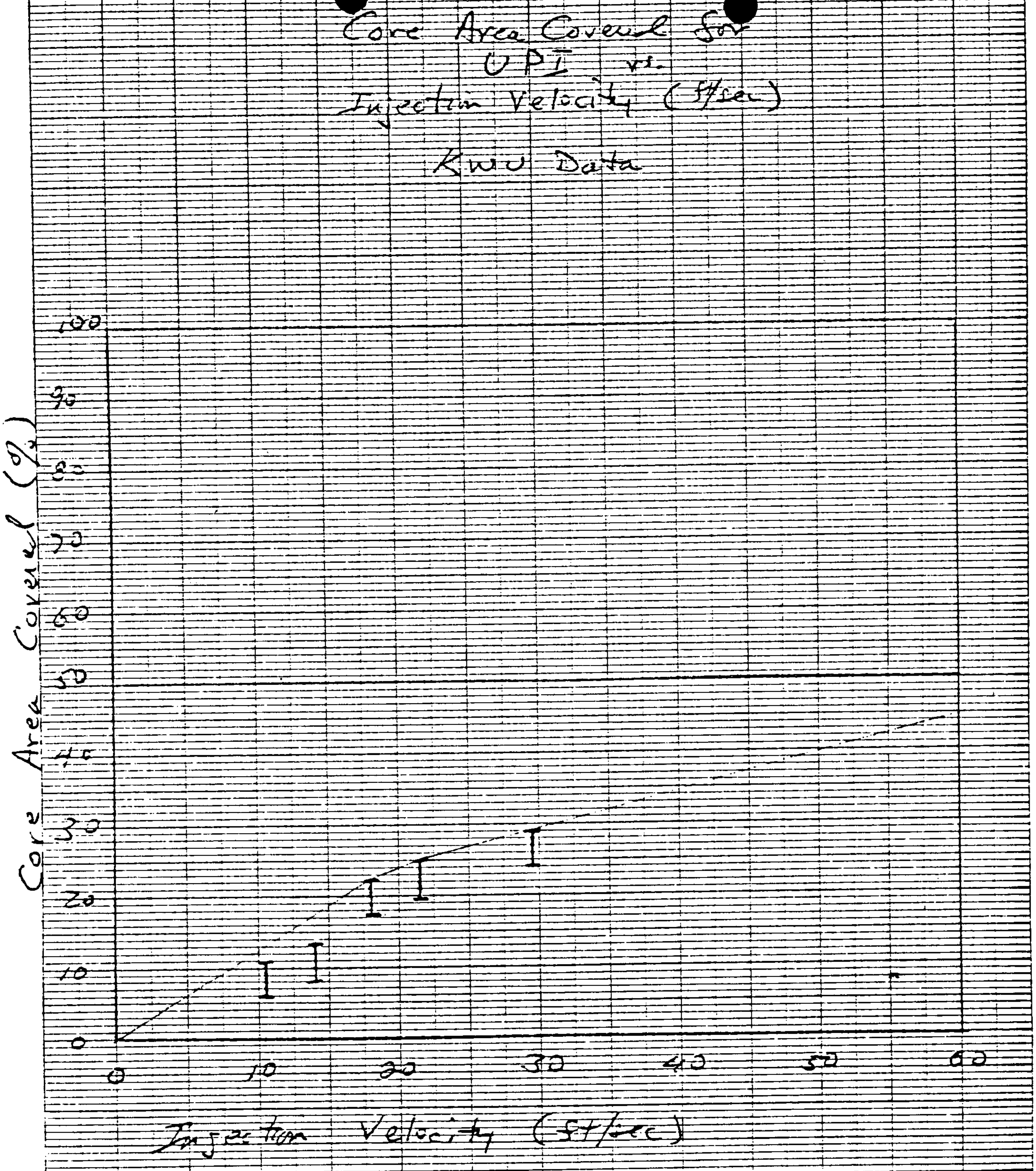
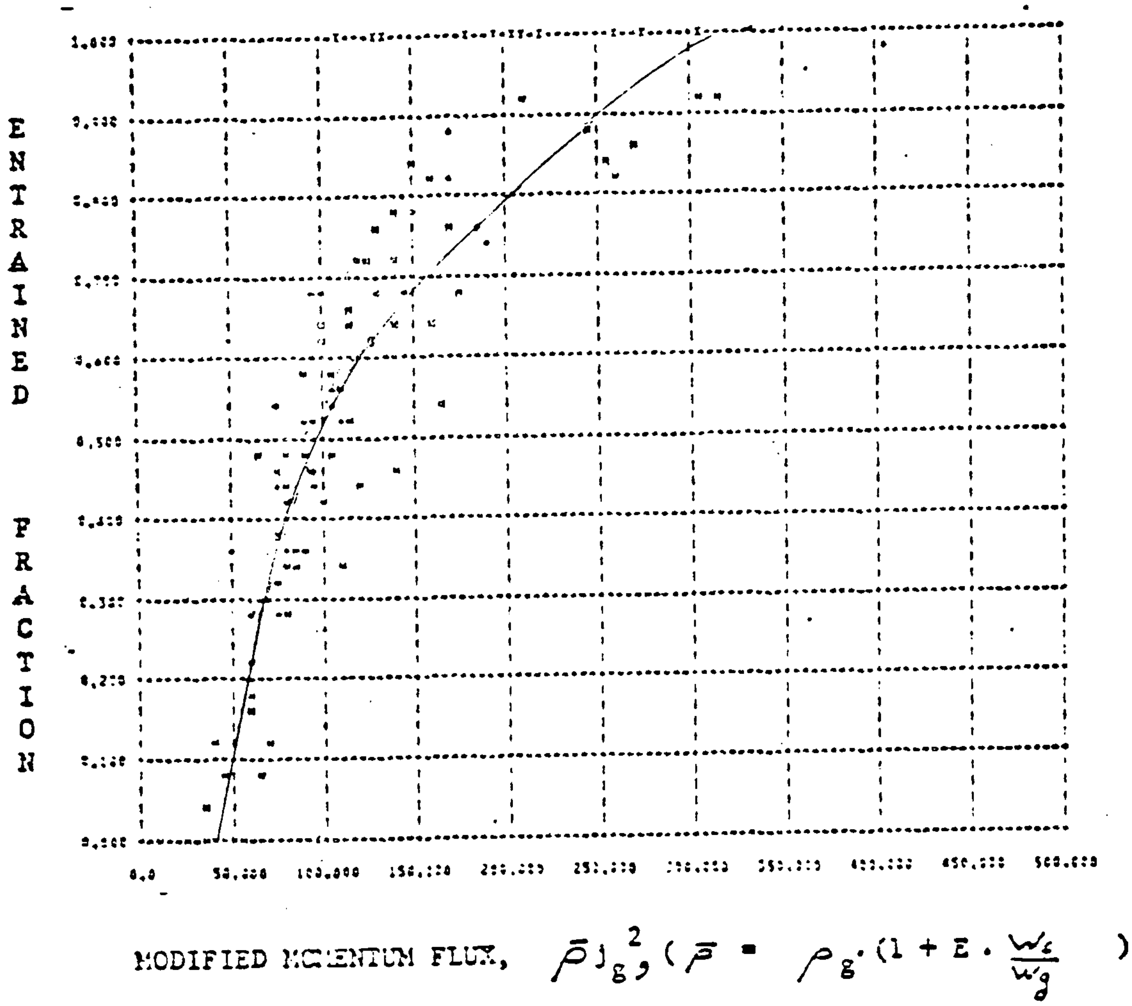
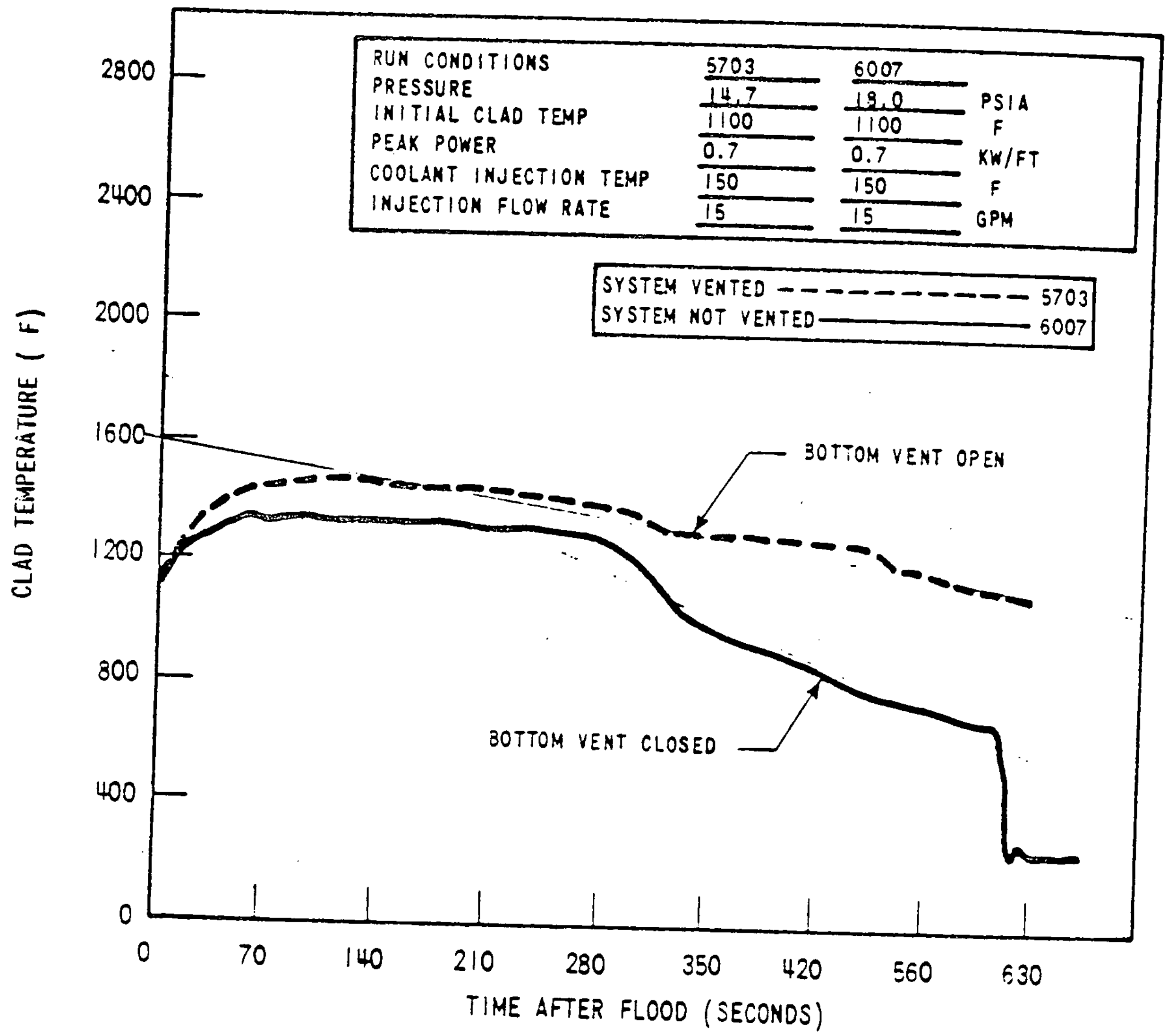

Core Area Covered for
UPI vs.
Injection Velocity (ft/sec)
KWU Data
Core Area Covered (%)
Injection Velocity (ft/sec)
0
10
20
30
40
50
60
70
80
90
100
0
10
20
30
40
50
60

Figure 4



ENTRAINED FRACTION VERSUS $\bar{\rho} \cdot j_g^2$

Figure 5

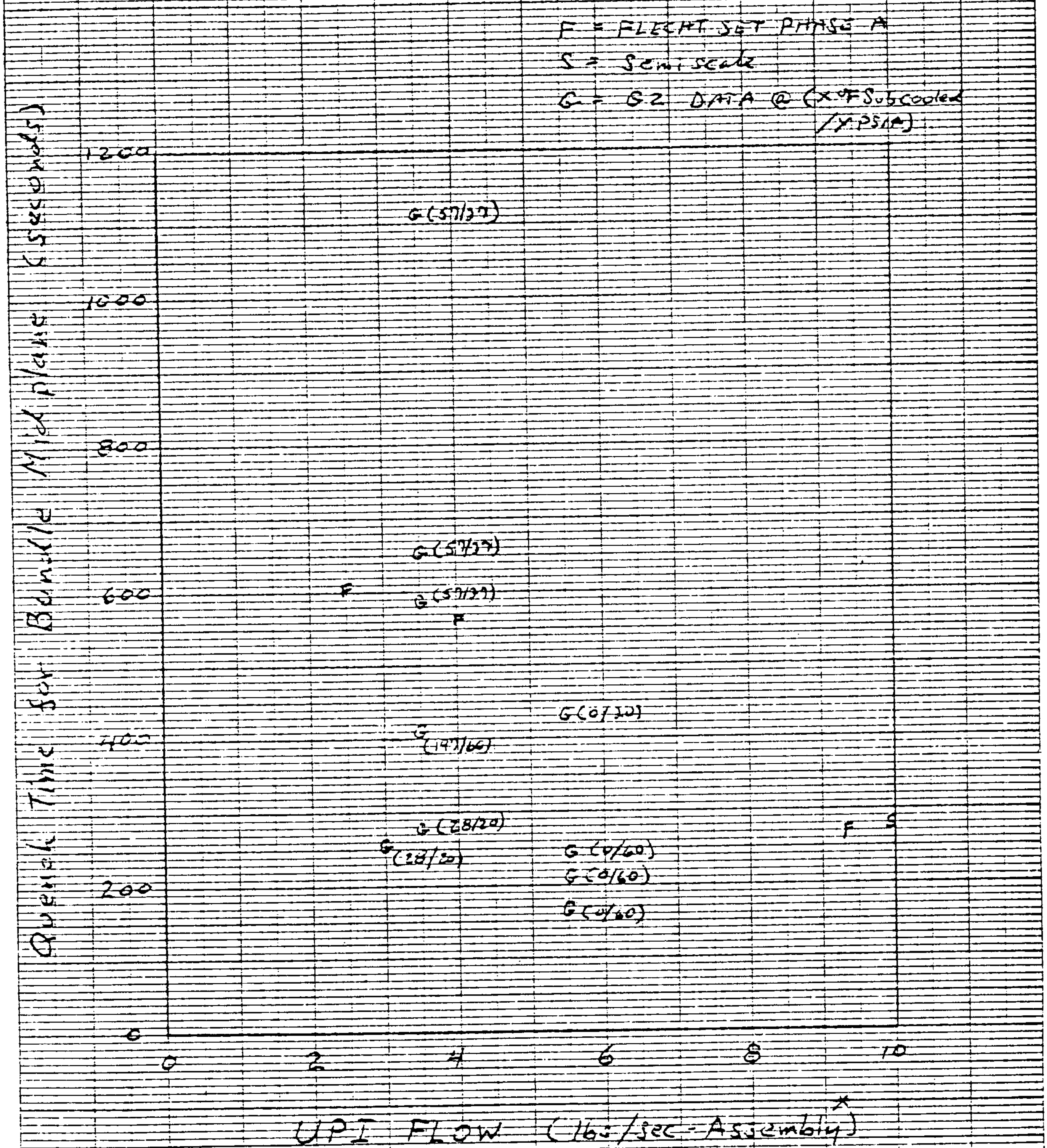


Effect of Bundle Venting on Rod 5G, Six-foot Temperature

FLECHT SET PHASE A
TOP INJECTION TESTS

Figure 6

Quench Time vs UPI Flow



* an Assembly is defined as 179 heated rods

Figure 7

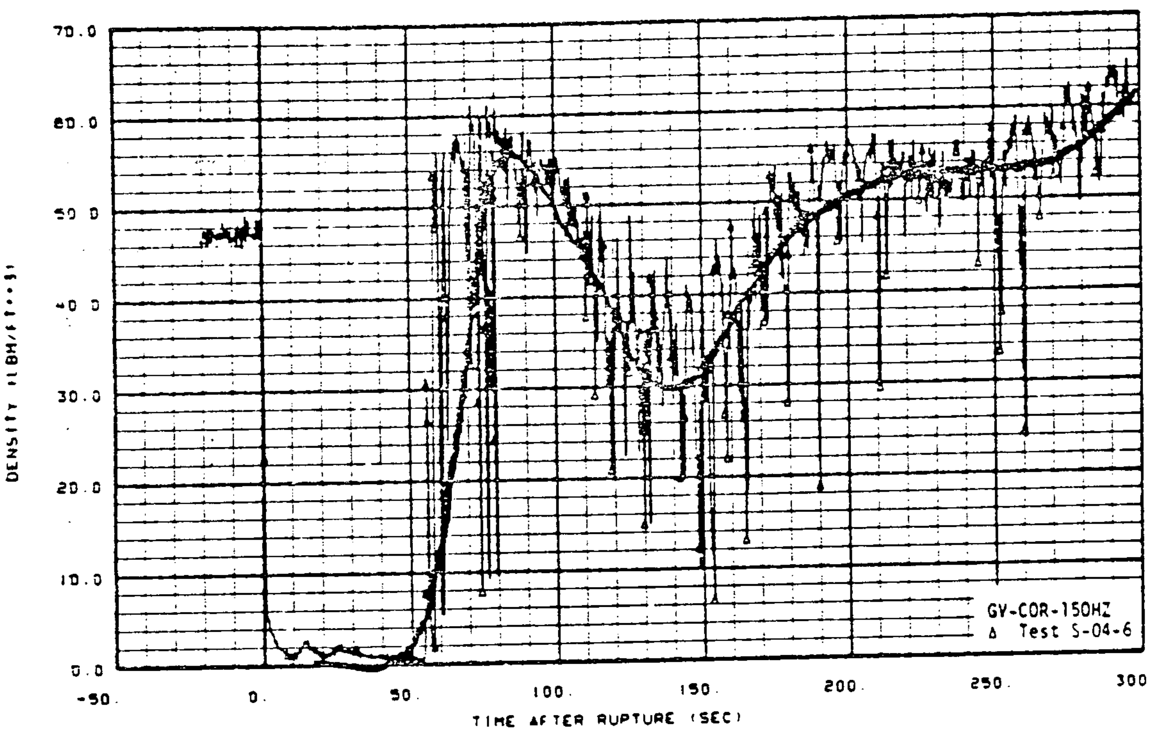


Fig. 501 Density in vessel, Test S-04-6 (GV-COR-150HZ), from -20 to 300 seconds.

BASE CASE

Figure 8

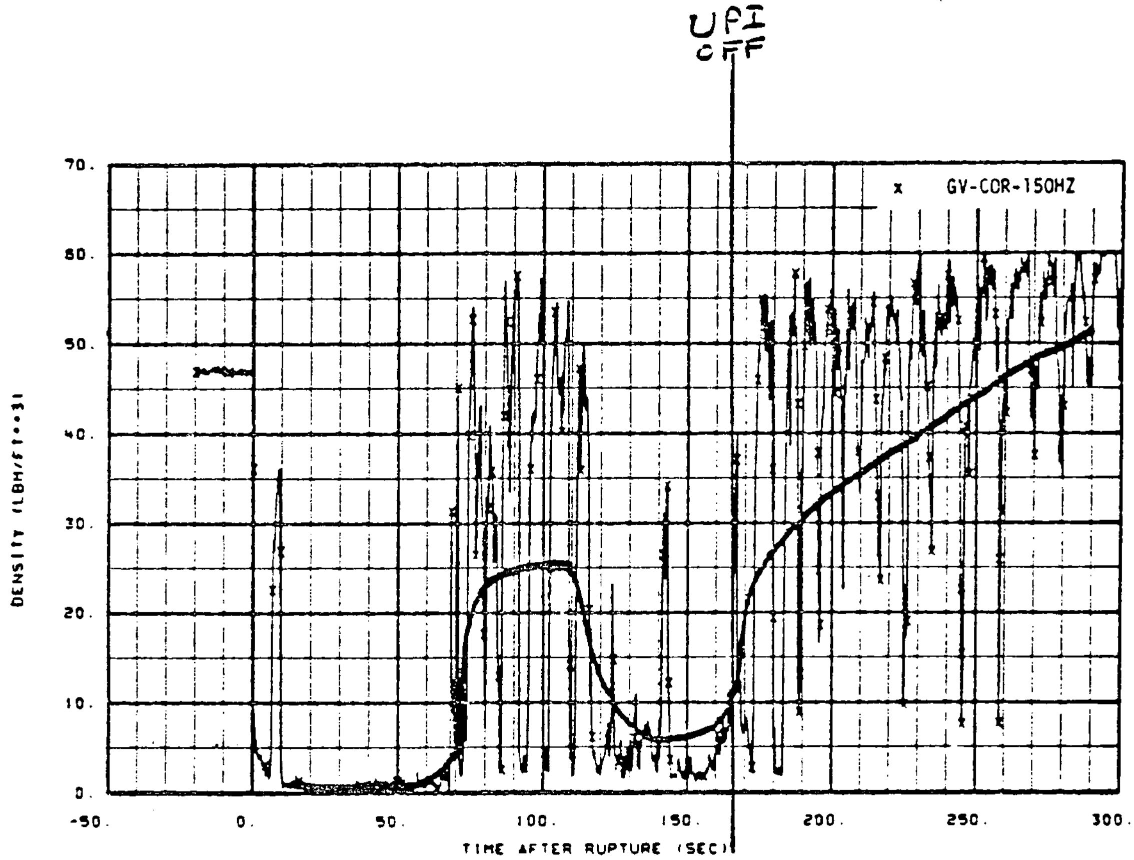
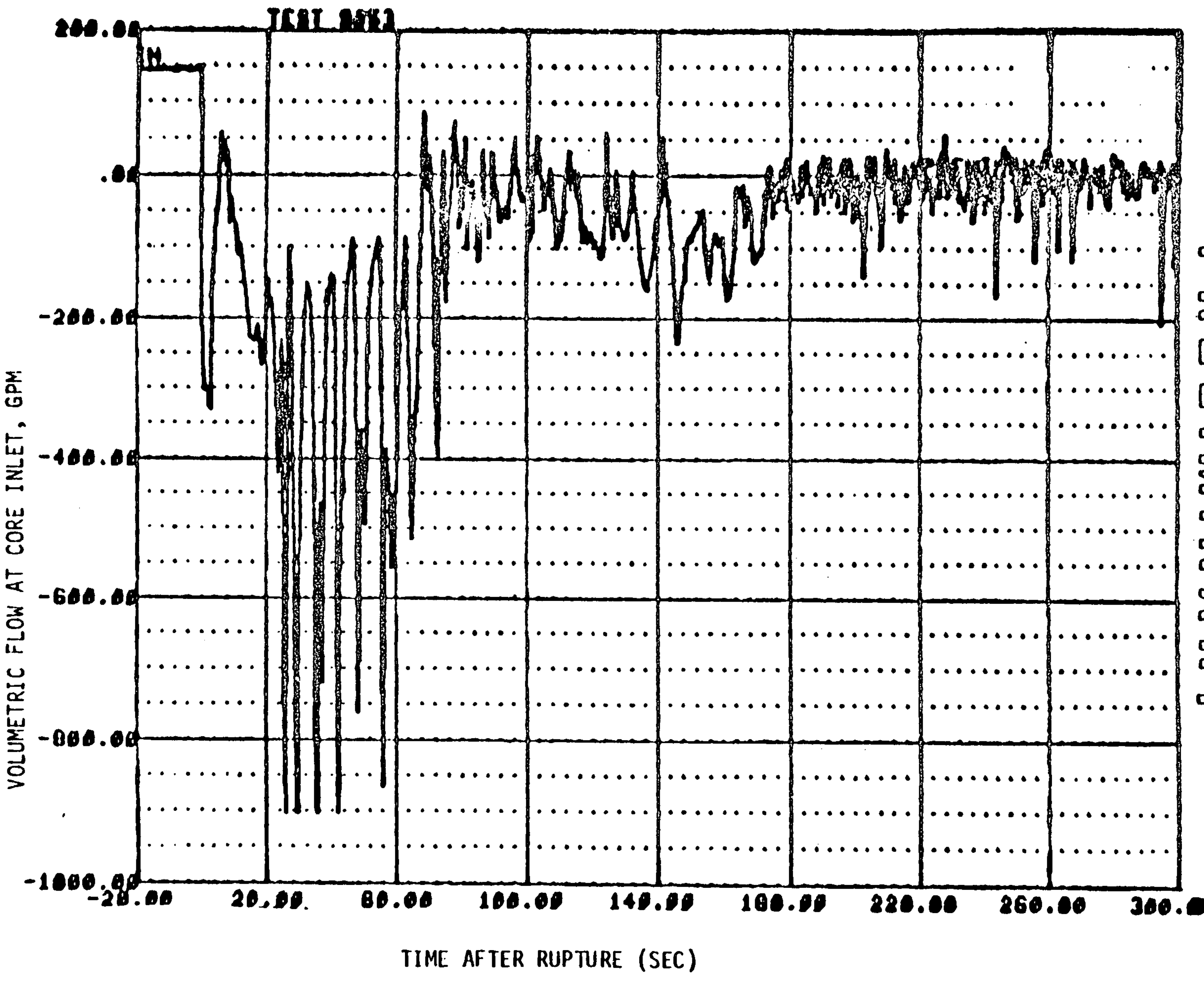


Fig. 283 Density in vessel (GV-COR-150HZ), from -20 to 300 sec.

UPI CASE

Figure 9

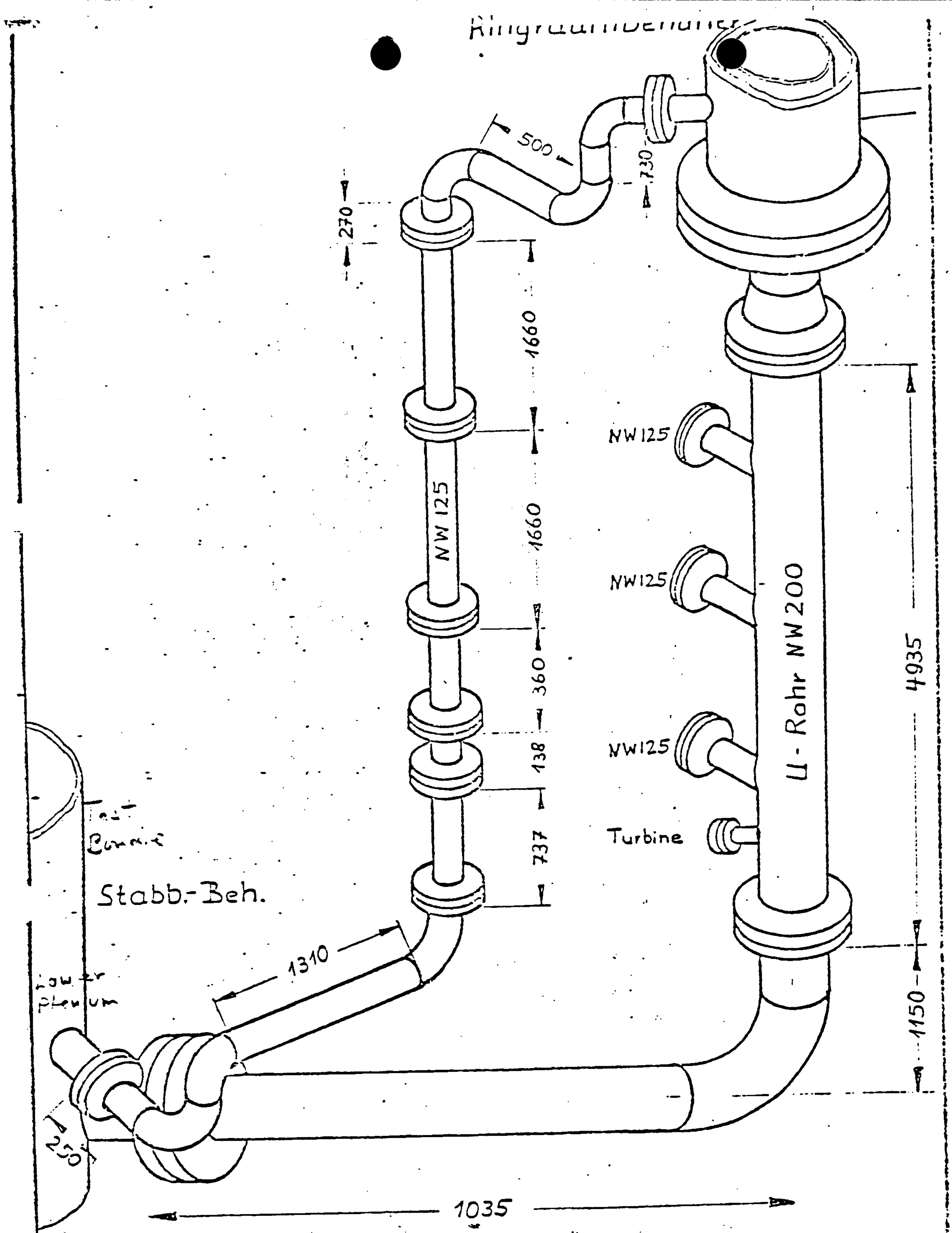


PRELIMINARY

Figure 10

PRELIMINARY

Volumetric Flow Rate at Core Inlet (Test S-05-3)

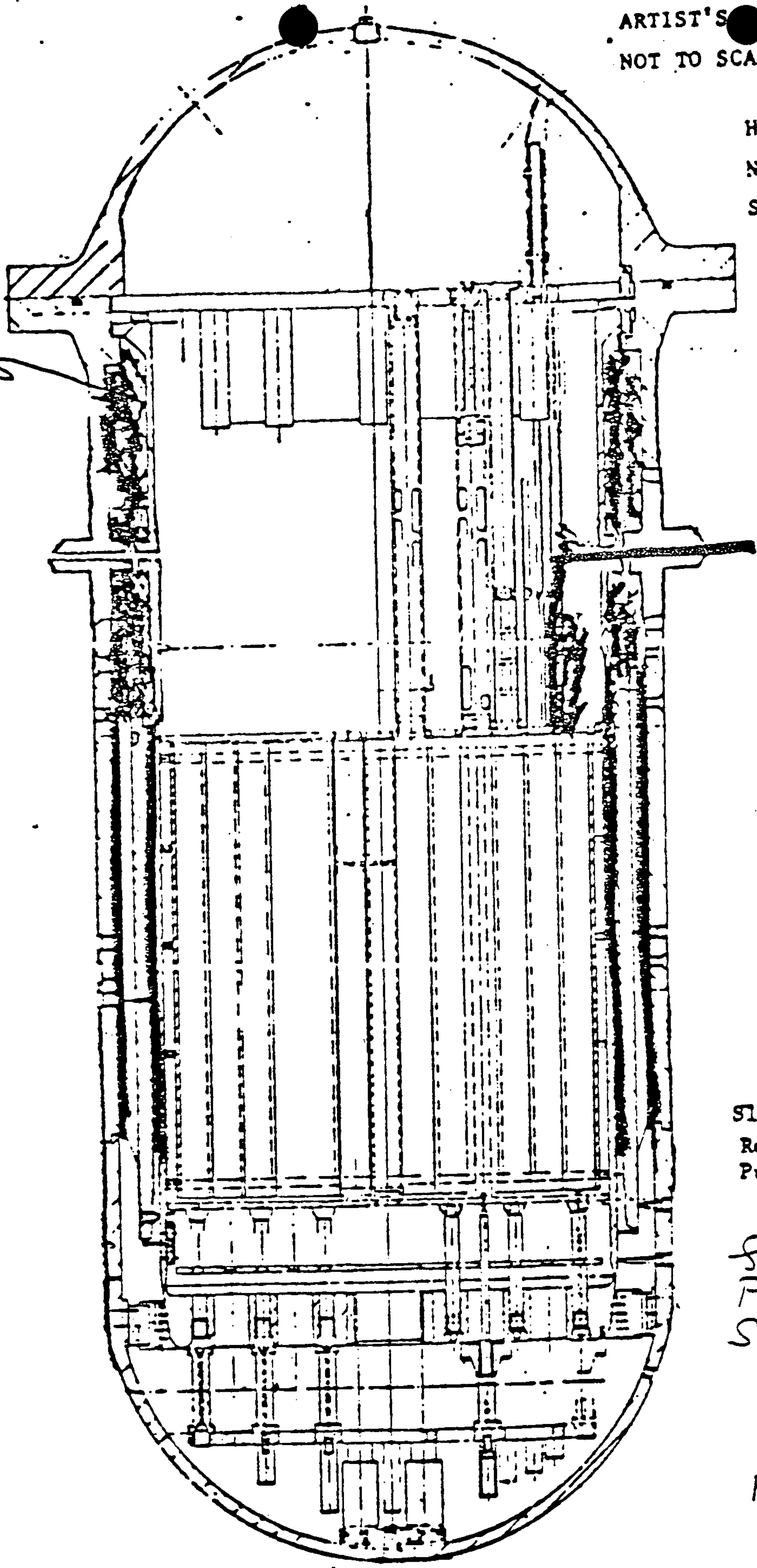


PKL Downcomer

PKL-Anlage

ARTIST'S CONCEPTION
NOT TO SCALE

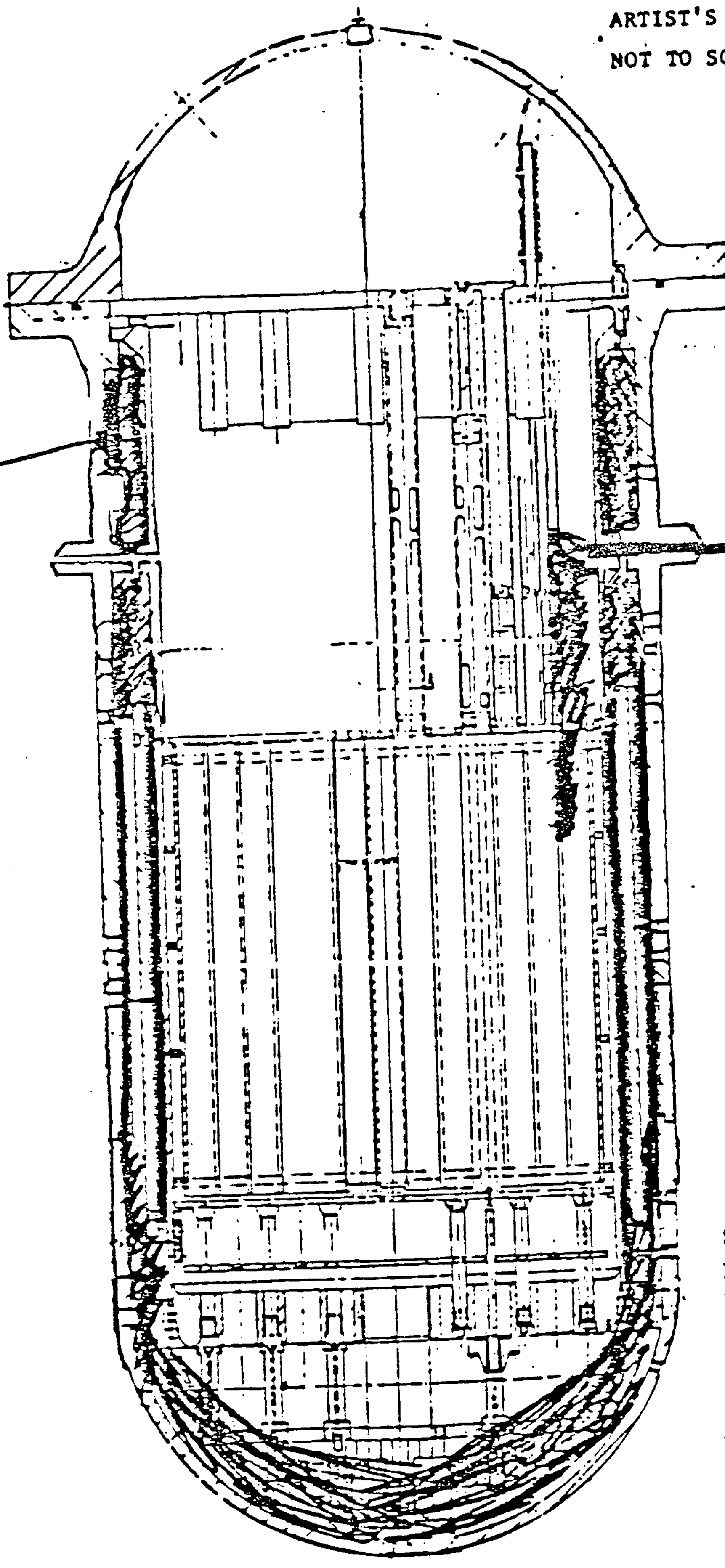
HOT AND COLD LEG
NOZZLES ARE NOT
SHOWN



Slide 1-2
Refill. Upper Plenum
Pump Started.

from
Prairie Island
Submittal

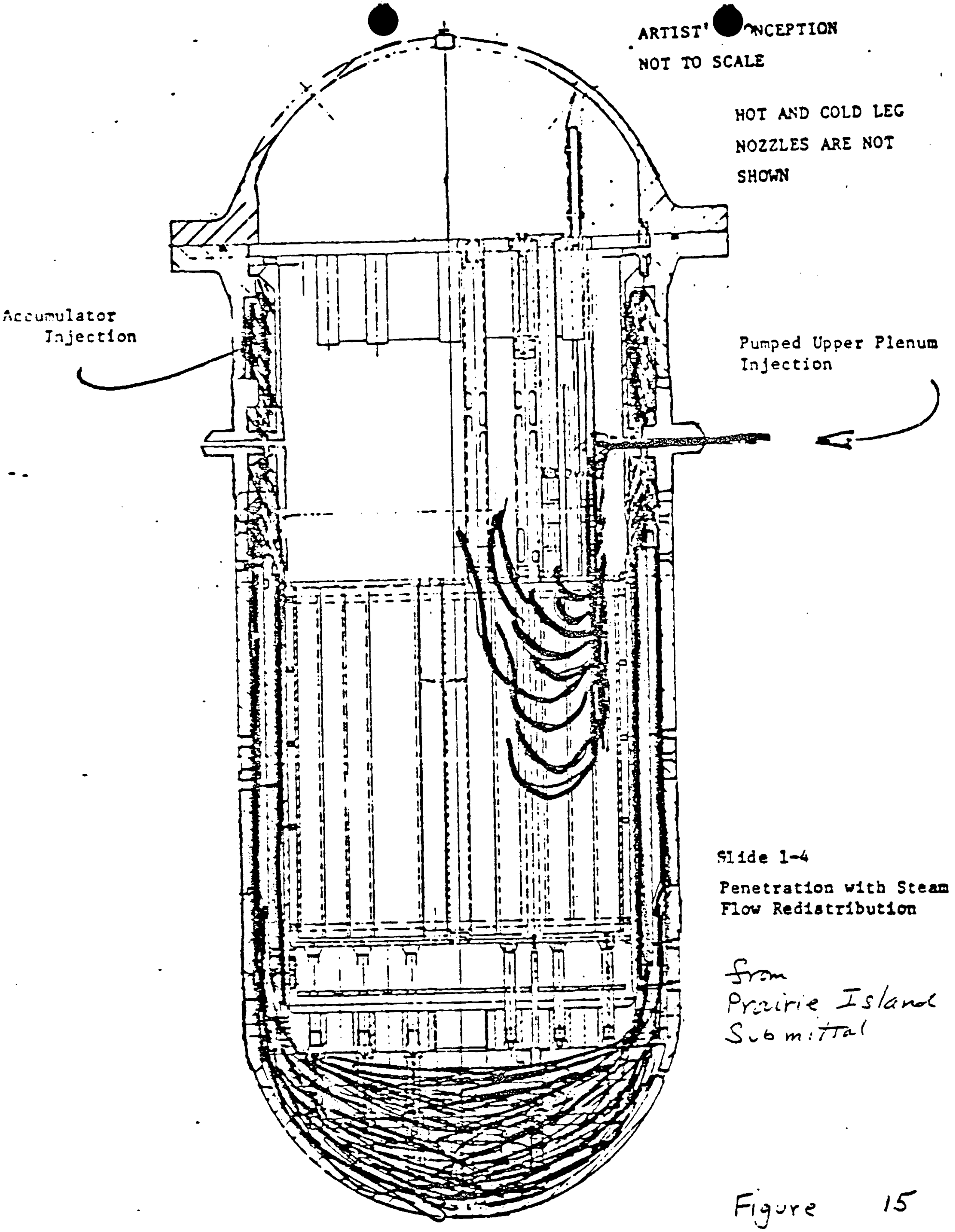
Figure 13



Slide 1-3
Penetration Prior to
Bottom of Core Recovery

from
Prairie Island.
Submittal

Figure 14

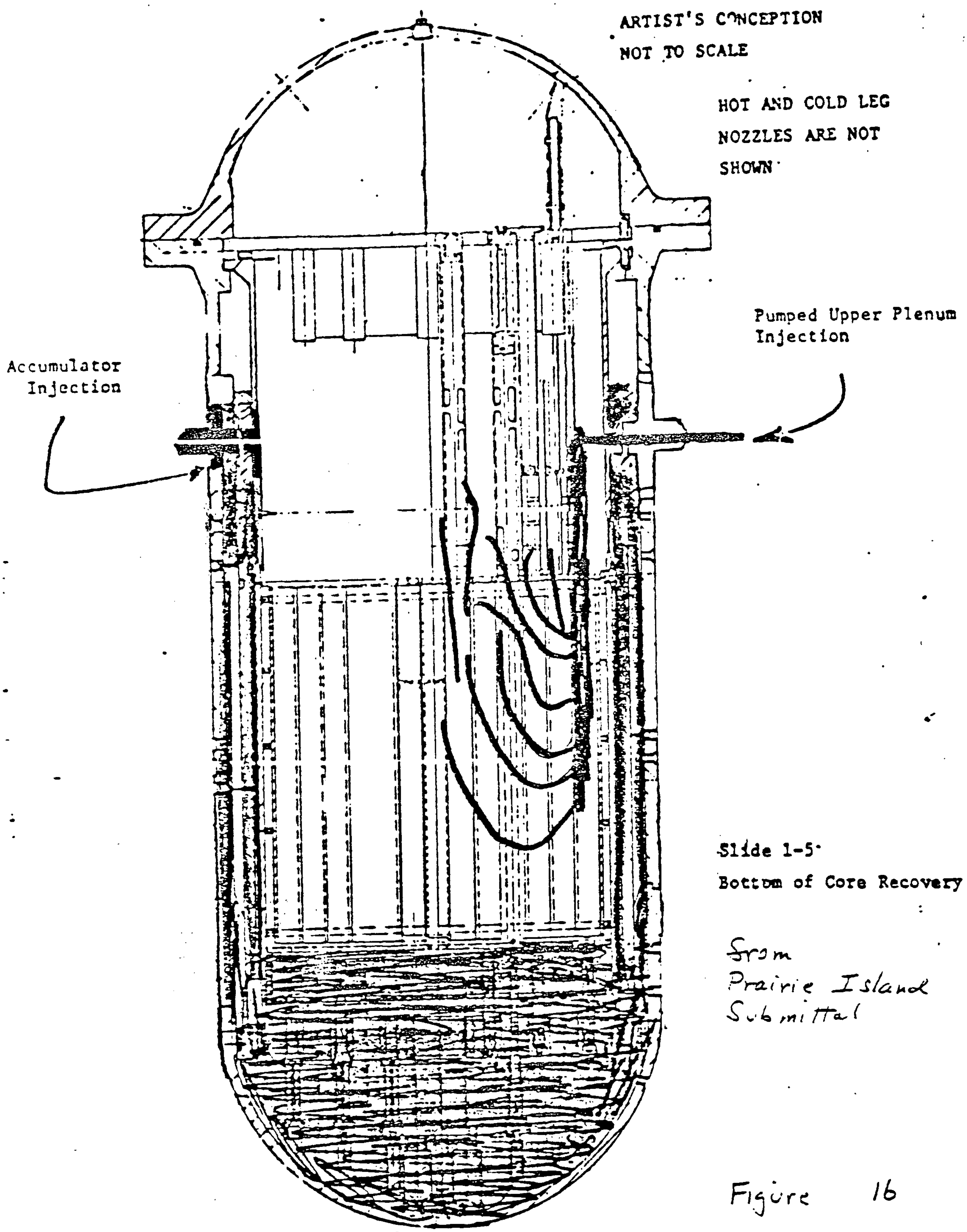


Slide 1-4

Penetration with Steam
Flow Redistribution

from
Prairie Island
Submittal

Figure 15



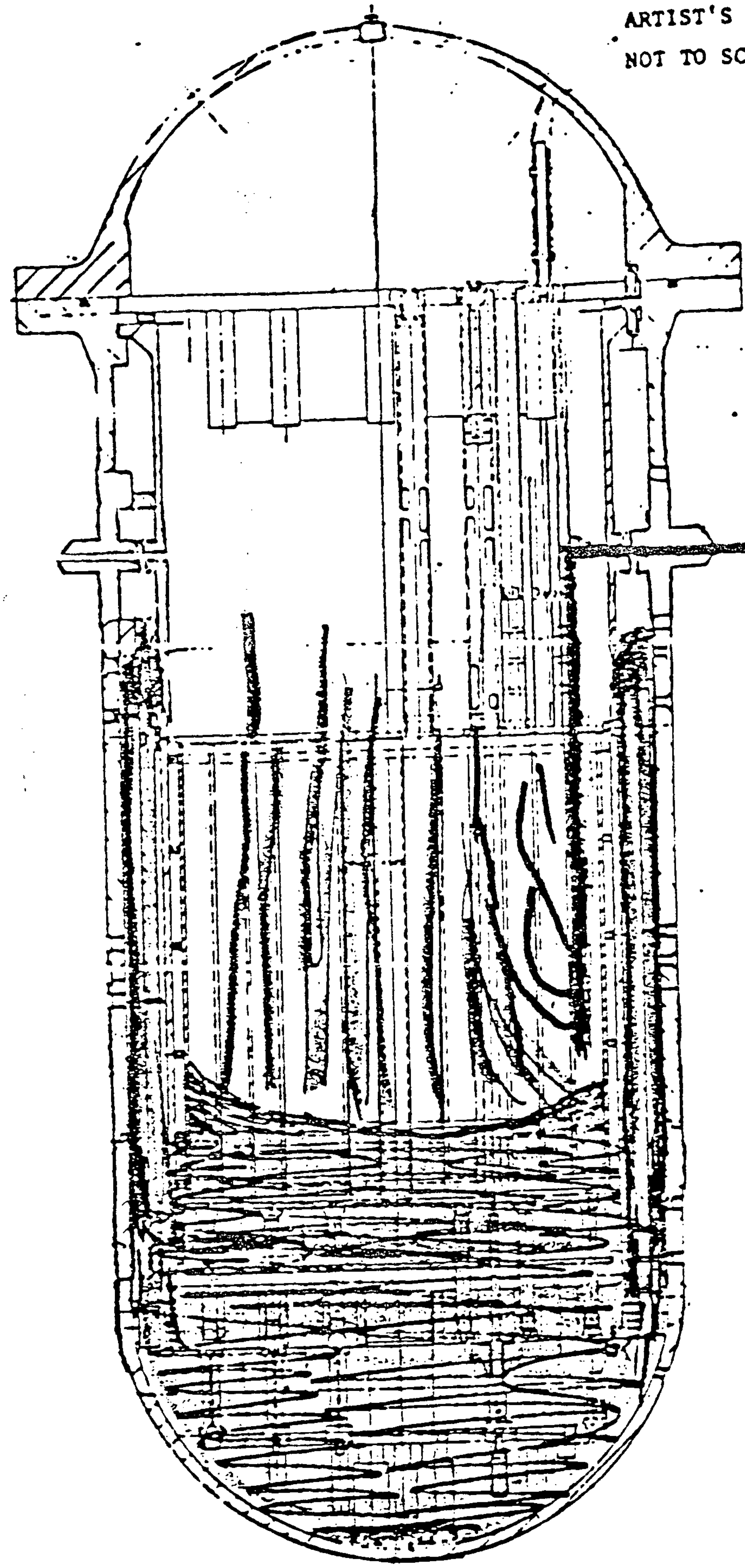
Slide 1-5
Bottom of Core Recovery

From
Prairie Island
Submittal

Figure 16

ARTIST'S CONCEPTION
NOT TO SCALE

HOT AND COLD LEG
NOZZLES ARE NOT
SHOWN

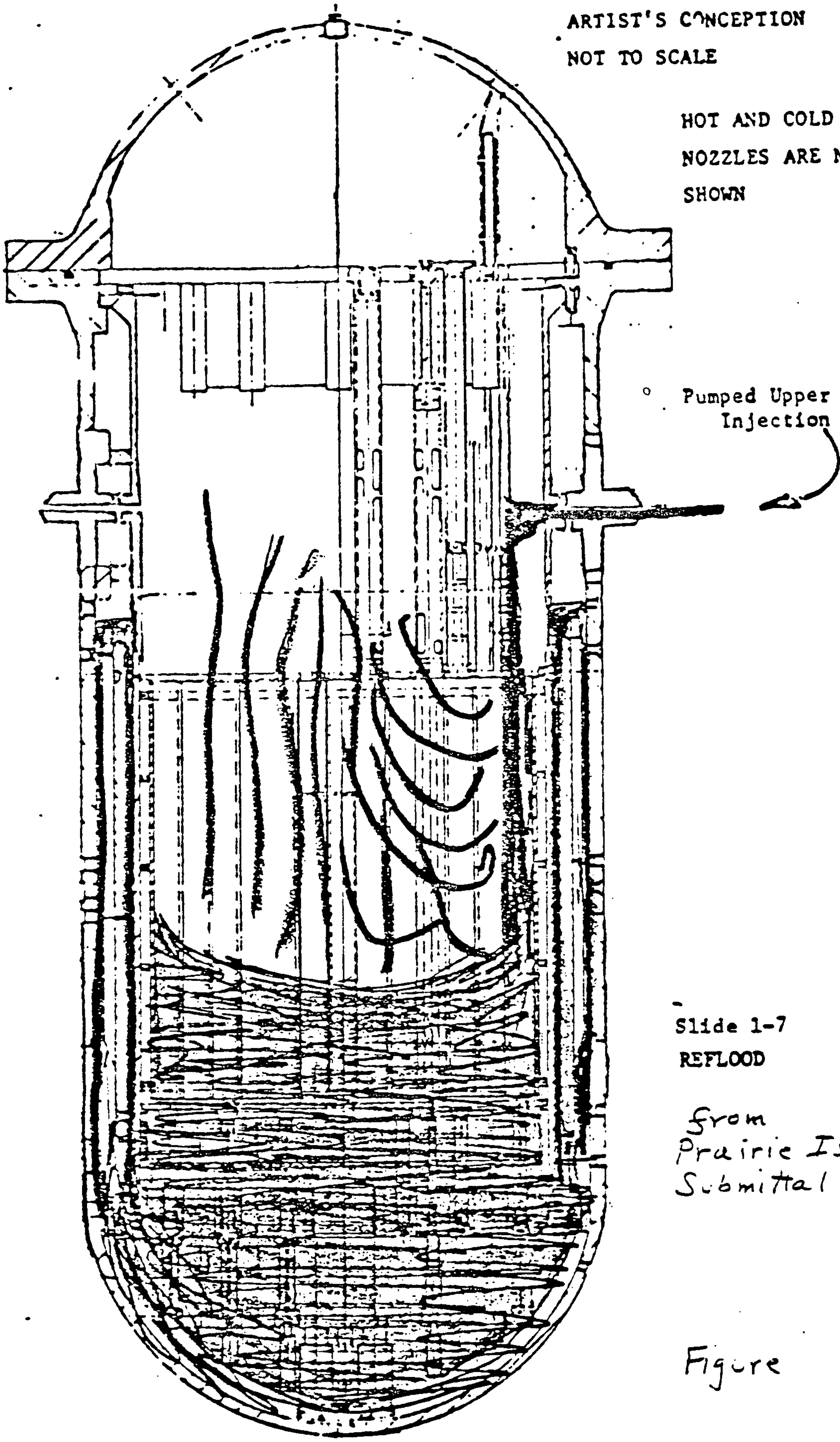


Pumped Upper Plenum
Injection

Slide 1-6-
REFLOOD

From
Prairie Island
Submittal

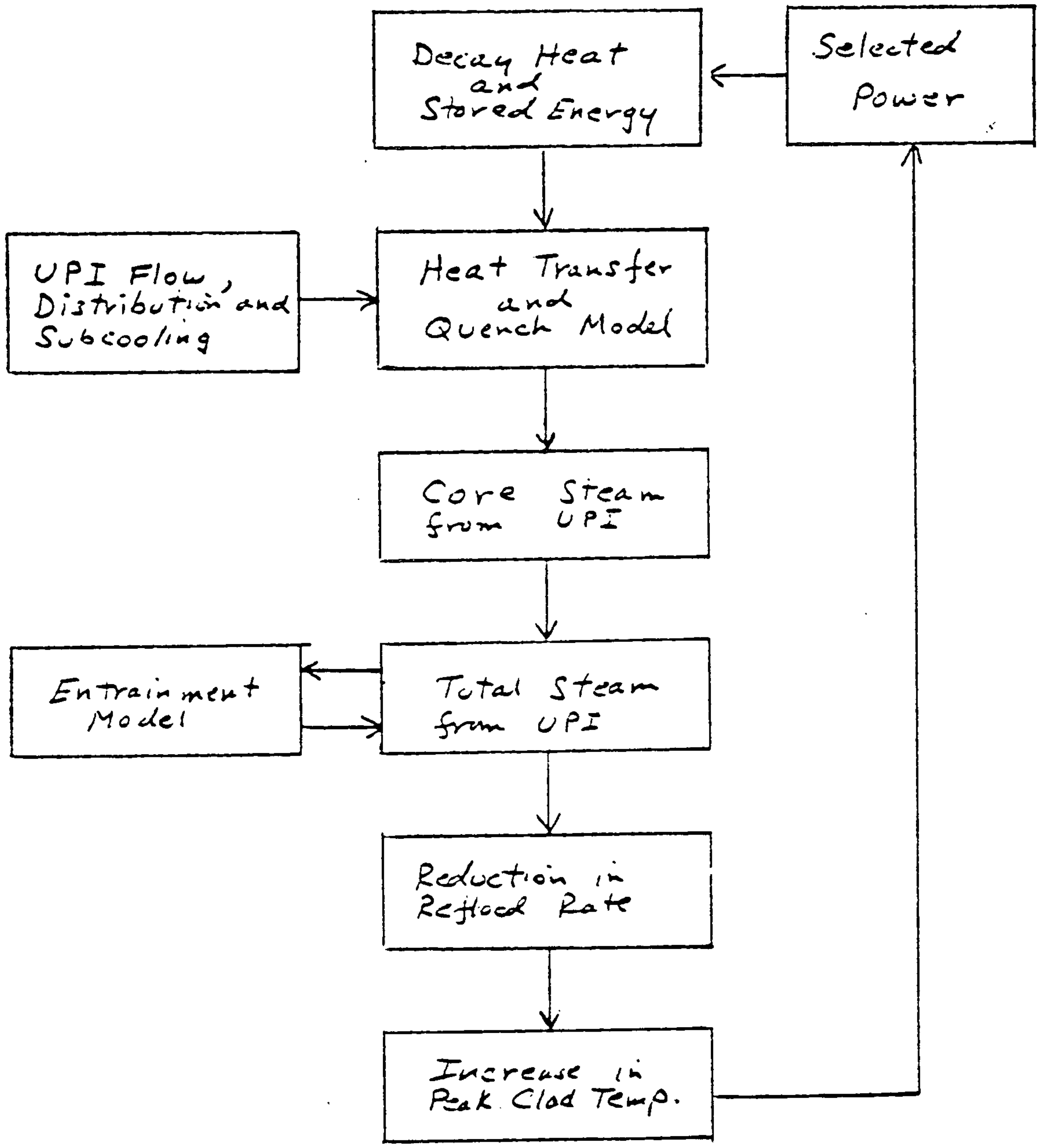
Figure 17



Slide 1-7
REFLOOD

from
Prairie Island
Submittal

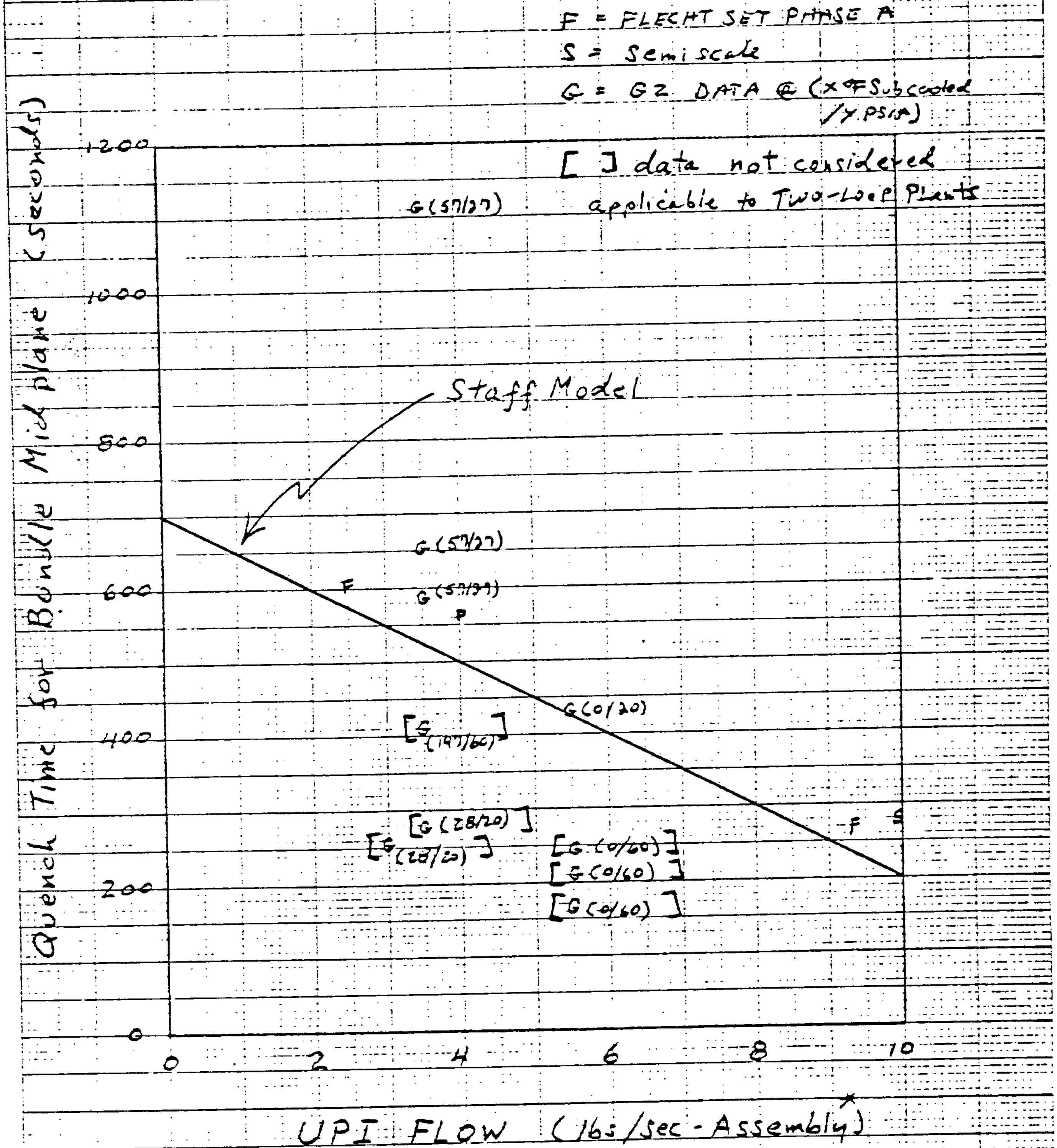
Figure 18



Staff Model to Account for the Effects of Upper Plenum Injection

Figure 19

Quench Time vs UPI Flow



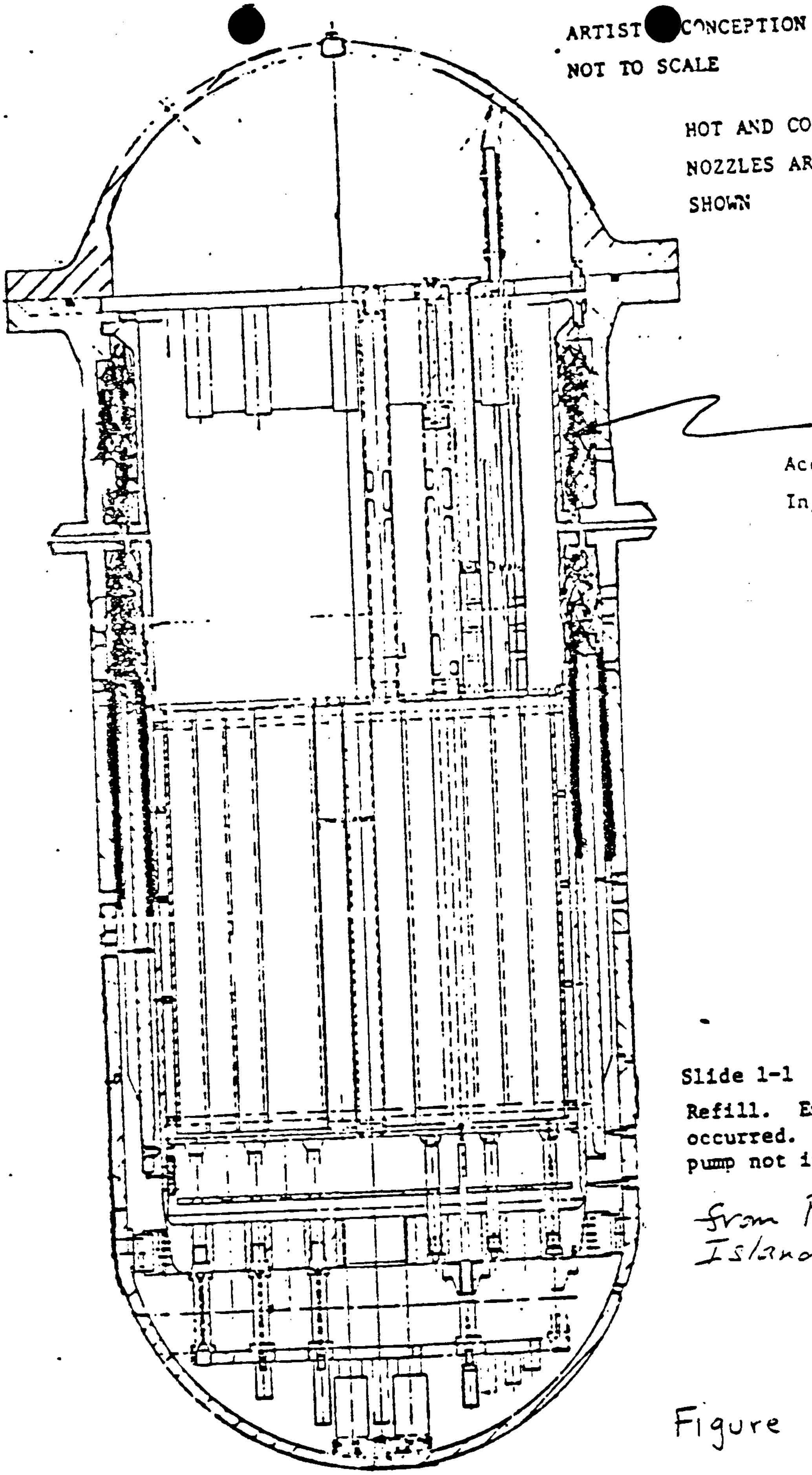
* an Assembly is defined as 179 heated rods

Figure 20

ARTIST CONCEPTION

NOT TO SCALE

HOT AND COLD LEG NOZZLES ARE NOT SHOWN

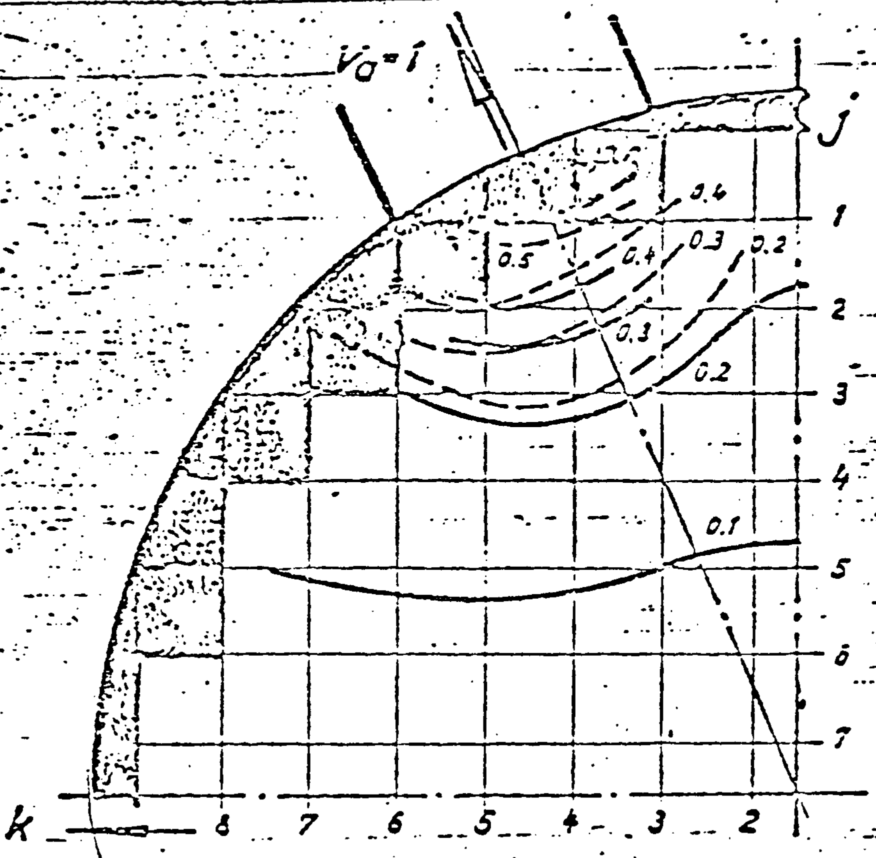


Slide 1-1

Refill. End of Bypass has occurred. Upper plenum pump not initiated.

from Prairie Island Submittal

Figure 12



and
Horizontalebenen $i = 4$ und $z = 170$

Horizontal planes

Average velocity in the outlet
Mittlere Geschwindigkeit im Auslauf $v_a = 1$

Zahlen an Isotachen geben die horizontalen Geschwindigkeitskomponenten
...mbers at isotachs indicate the horizontal velocity components

- - - - - Elektrolytischer Trog — electrolytic trough
——— Strömungsmodell — flow model

Figure 21

ISOTACHS IN THE UPPER COLLECTOR SPACE OF THE STADE NUCLEAR POWER PLANT
Comparison of Results: Electrolytic Trough and Air (Flow) Model

Isotachen im oberen Sammelraum KKS
Vergleich der Ergebnisse: Elektrolytischer Trog und Luftmodell

8/17/71
CMH

Fraction of the Upper Plenum Area
with air velocity ≥ given velocity
vs
air velocity

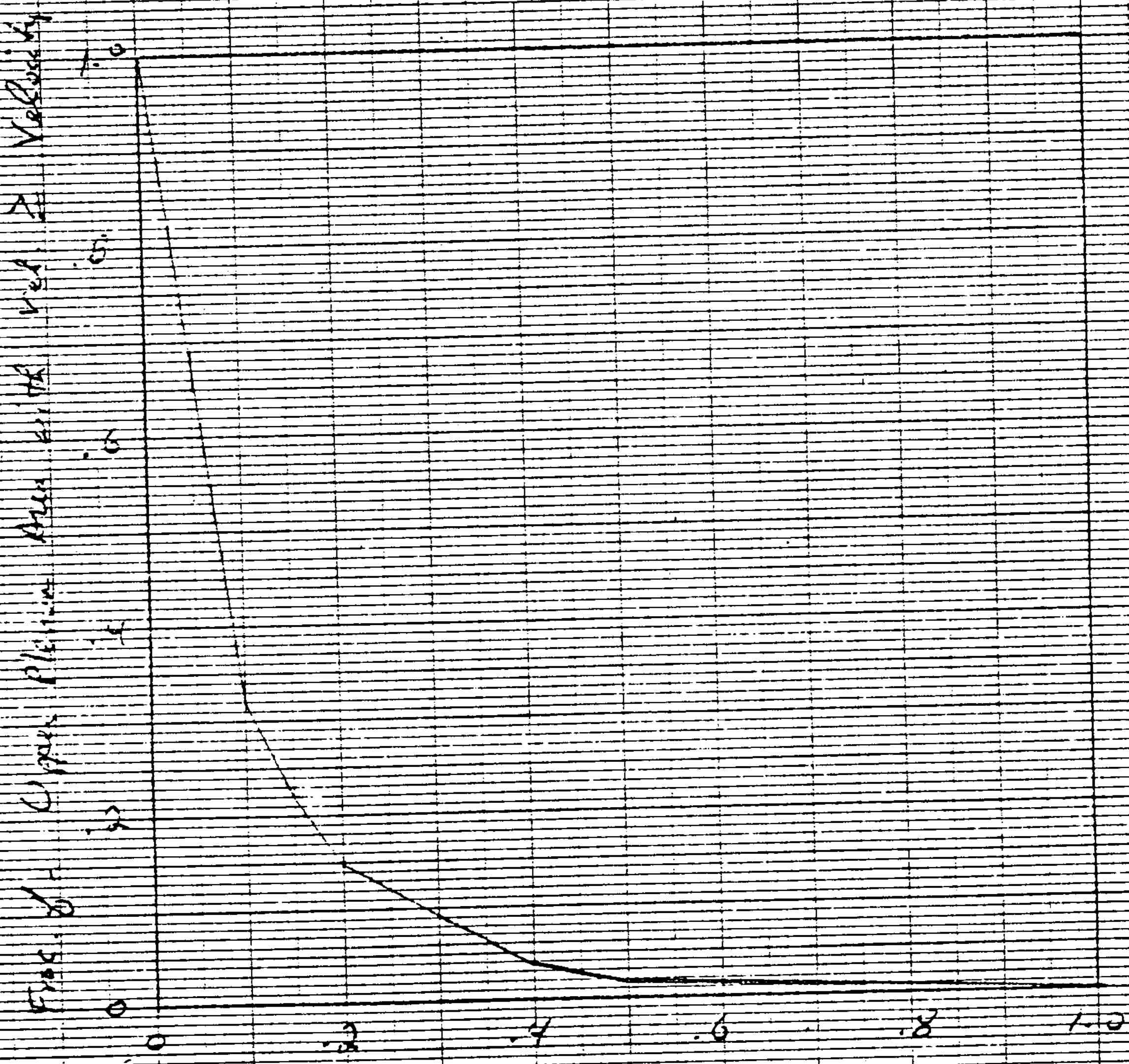


Figure 22

46 1320

K+E 10 X 10 TO ½ INCH
KEUFFEL & ESSER CO.

461510

UPI Steam Generation
vs
Power Level
at the Beginning of Reflood

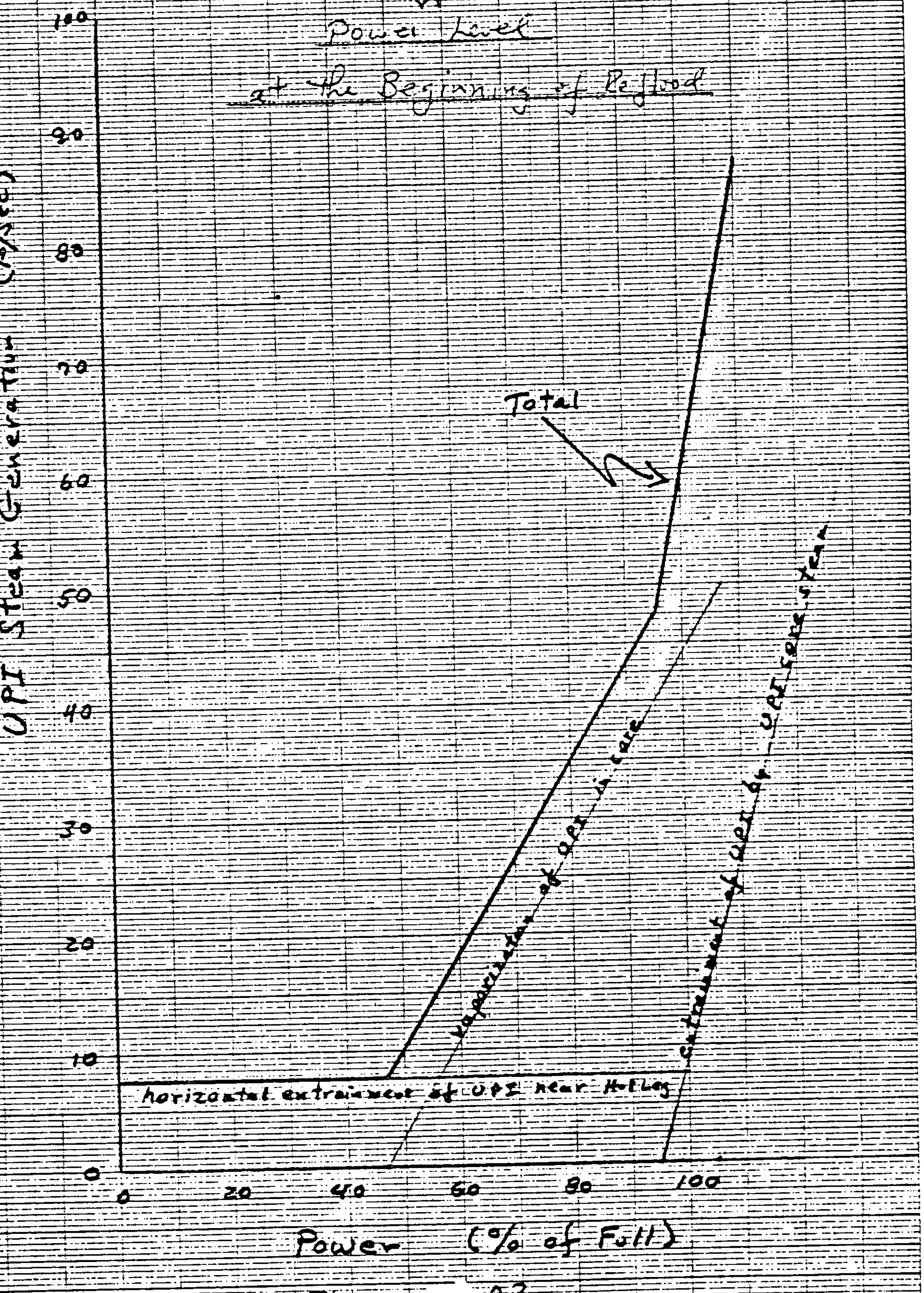


Figure 23

Change in Peak Clad Temperature
vs
Reflood Rate

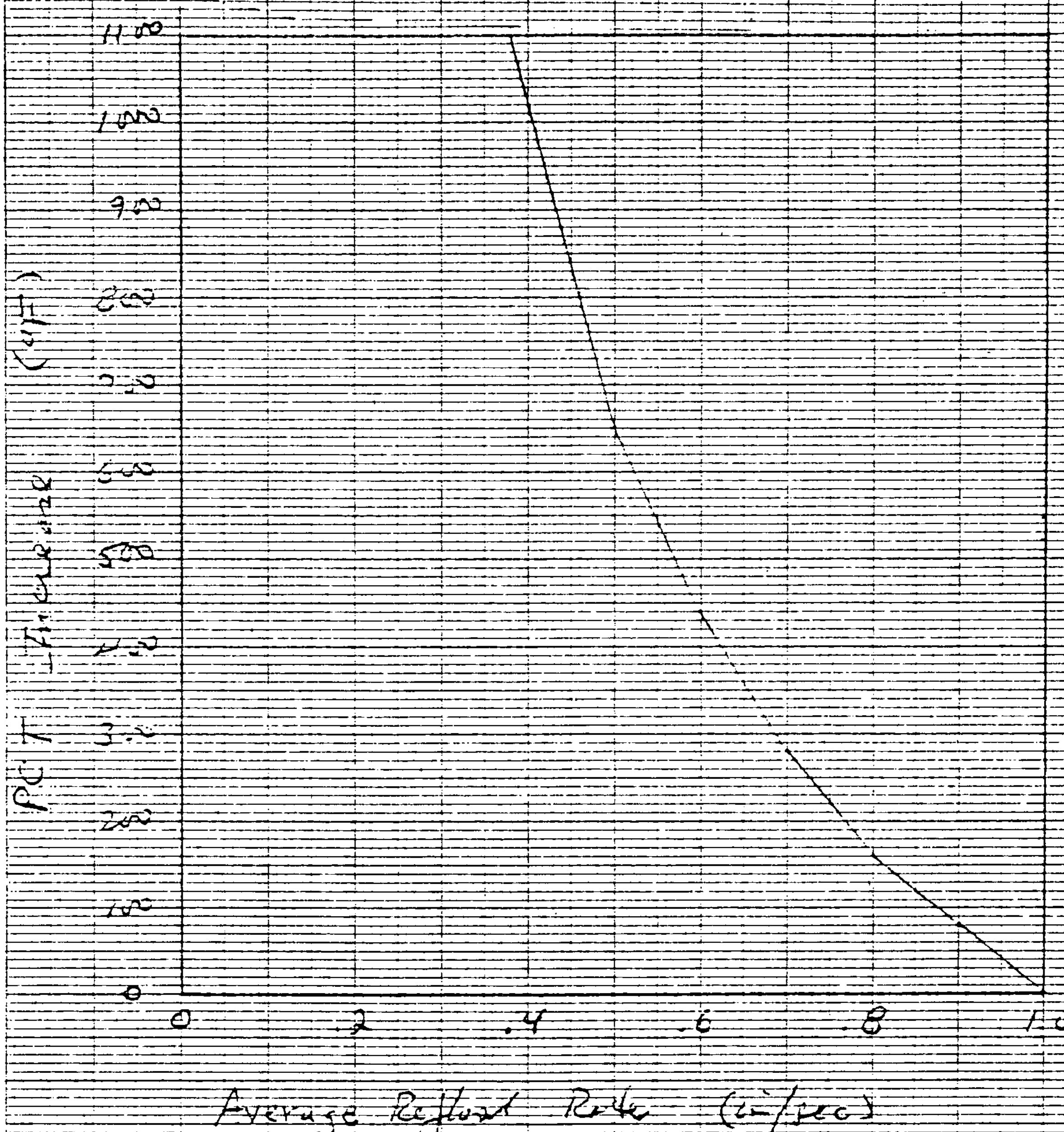


Figure 24

Calculated Penalties and Credits

Peak Clad Temperature

vs.

Initial Power Level

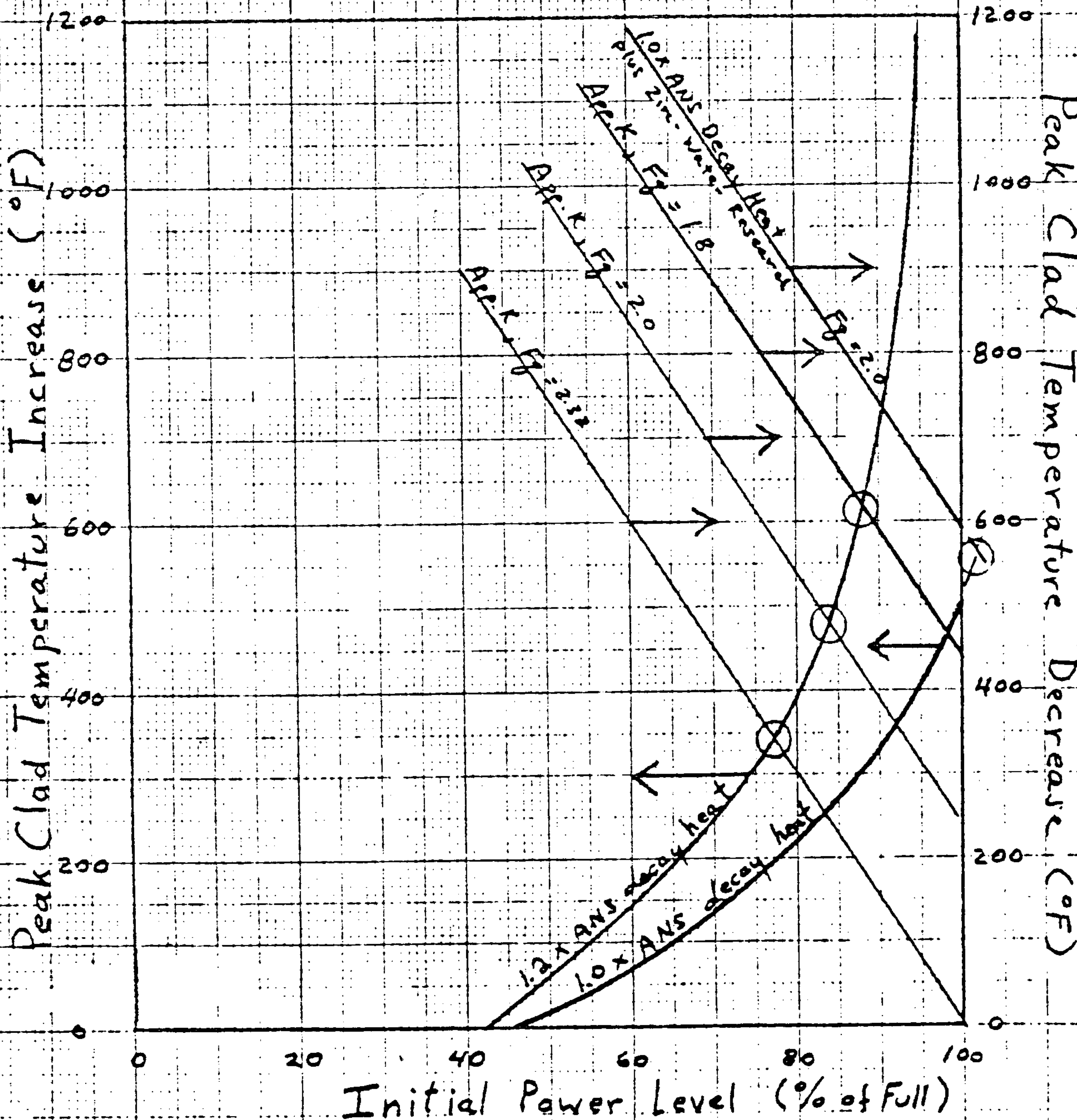


Figure 25

SAFETY EVALUATION SUPPORTING CONTINUED SAFE OPERATION OF WESTINGHOUSE DESIGNED TWO LOOP PLANTS

Introduction

In all modern nuclear plants designed by the Westinghouse Electric Corporation, the emergency core cooling (ECC) injection water is delivered to the reactor vessel by accumulators and by high and low pressure pumped injection systems. In Westinghouse designed two-loop plants, high pressure pumped injection and low pressure accumulator water, which is injected into the cold legs, reaches the lower plenum through the downcomer. The low pressure injected water, and in some plants a portion of the high pressure pumped injection water, is directly injected into the upper plenum through two injection nozzles located approximately 80° from the hot leg and penetrating the reactor vessel and the core barrel. Typical design injection rates for two-loop plants are 700 gpm/pump for high pressure injection and 2000 gpm/pump for low pressure injection.

The Westinghouse ECCS analyses (Reference 1) have assumed that the water injected by both high and low pressure pumped injection systems contributes to refill and reflood of the core after initial blowdown following a LOCA. In those analyses for two-loop plants it was assumed that the entire volume of of water injected into the upper plenum from the low pressure injection pumps passes through the core to the lower plenum without being significantly impeded and without cooling the fuel elements. These aspects of the ECCS evaluation model had been previously reviewed and approved by the staff in 1971 and remained in effect through the final model review in 1975 (Reference 2). The staff evaluation was based on the conclusion that although the phenomena modelled might not be fully representative of the actual physical situation, there was no analysis to contradict the belief that they were conservatively accounted for in the evaluation model.

Since the issuance of the staff evaluation in 1975, the staff has been continually reviewing new information on liquid-vapor interaction, counter-current flow regimes, and core thermal response to ECCS injected above the core. During this period, several small scale and separate effects model tests performed in this country and abroad have been reviewed by the staff (References 4 through 10). A large part of the review was carried out in conjuction with the evaluation of the upper head injection (UHI) emergency core cooling system design proposed by Westinghouse (Reference 11) for some new facilities. The information gained as a result of that review indicated that the interaction between the water flowing through the core and the hot fuel rods cannot be ignored and, that under certain circumstances, this interaction may impede core flooding and reduce the conservatism existing in the two-loop ECCS evaluation model analysis.

In January 1977 the NRC informed the Westinghouse Electric Corporation and licensees of the Westinghouse designed two-loop plants (Point Beach 1/2, Ginna, Prairie Island 1/2, and Kewaunee) of our concern regarding the adequacy of the existing evaluation model for two-loop plants, and has requested that appropriate actions be taken relative to these concerns (Reference 12). Two meetings have been held between the NRC staff, the licensees of affected plants and Westinghouse (References 13 and 14) during which the licensees and Westinghouse presented information aimed at justifying the adequacy of the model presently used in the ECCS analysis for these plants. In addition, the licensees submitted a report in which they documented the information presented at those meetings (Reference 15).

For some time, the NRC staff has been reviewing the available information and has concluded that the material presented does not provide sufficient justification for continued acceptance of the existing upper plenum injection (UPI) model as a conservatve representation of UPI of the emergency core cooling water. Analyses performed by the staff using the methods outlined in Reference 16 have shown that the upper plenum injection can produce, under certain circumstances, a significant increase of hot spot peak clad temperature (PCT) above the values presented by the licensees in their safety analyses. The staff concluded, therefore, that some appropriate action be taken, to quantify the impact of UPI on the ECCS performance.

Discussion

As indicated above, the Westinghouse UPI model assumes that the UPI water reaches the lower plenum without interacting with the core, consequently it does not consider either upper core quenching or steam generation effects. These simplifying assumptions produce some conservative and some non-conservative consequences. Ignoring the transfer of heat from the fuel rods to the ECC water as it moves from the upper plenum through the core is itself conservative at that point. However, by not considering the effects of heat transfer to this water, the analysis is non-conservative since the vaporization in the core and upper plenum and the resulting liquid entrainment and carryover into the steam generators, would increase steam binding effects and upper plenum pressure. This would reduce the core reflood rate. In the original submittal (1971) Westinghouse believed that the combined effect of these phenomena was insignificant and their simplified UPI model was used as a conservative representation of the physical situation.

The analytical model developed by the staff is described in Reference 16, In developing this model, the staff reviewed the available experimental data on the effects of top ECC injection from separate effects tests (i.e., heat transfer, quench, etc.) and integral tests (i.e., blowdown refill and reflood). In some areas, there is limited experimental data and the applicability of this data to two-loop plants is uncertain. In such cases, the staff model utilized conservative assumptions. For example, the UPI flow was assumed to be uniformally distributed above the core.

The integral tests and the staff's model (which is largely based on the separate effects tests) indicate that upper plenum injection has a significant effect on the core quenching characteristics. Upper plenum injection causes greatly increased top-to-bottom core quenching in the areas where the UPI water enters the core and decreased bottom core quenching in all regions. Upper plenum injection can therefore cause increased heat transfer and early quench in some core areas and decreased heat transfer and delayed quench in other core areas. Although not calculated, the staff's judgment is that the potentially adverse effect of upper plenum injection is inherently limited to relatively small areas of the core (less than 10 percent). Other areas of the core would receive adequate cooling water and the effect on this relatively small area would not affect gross core geometry.

The staff's model was used to evaluate the effects of UPI on PCT. It demonstrated that, with assumptions consistent with the requirements of Appendix K to 10 CFR 50, UPI effects could cause a significant increase in PCT thus causing the PCT to exceed the 2200°F limit. In addition, these staff analyses have shown the effect of UPI on PCT to be sensitive to several parameters considered by the staff. For example, a reduction in total peaking factor, F_Q, from 2.32 to 2.00 would result in a 240°F reduction in calculated PCT. Similarly, the PCT rise is sensitive to the decay heat and metal-water reaction rates assumed. For example, by assuming decay heat heat of 1.0 x ANS instead of 1.2 x ANS, the UPI effect on PCT would be reduced by several hundred degrees. The combined effects of reduced F_Q, decay heat and metal-water reaction rates may, in some cases, offset the calculated adverse effects of UPI.

We therefore conclude that in the unlikely event of a Loss-of-Coolant Accident, the core quenching characteristics would be significantly different from those calculated by the ECCS evaluation model but that these differences could not lead to the melting of a significant portion of the core.

The discussion above relating to steam binding is principally a concern in the case of a large break LOCA (>6 in.).

Observed failure statistics (Reference 18) confirm rates of 10^{-4} to 10^{-6} per reactor year in large pipes with the rates increasing as the pipe size decreases. Furthermore, only large pipe breaks in the cold legs are calculated to result in a PCT that approaches the 2200°F limit with the currently approved model. Therefore, it can be concluded that a conservative estimate of the probability of a large pipe break at a critical location in the primary coolant system is in the range of 10^{-4} to 10^{-6} per reactor year. This analysis is supported by the probability of occurrence of large break LOCA presented in WASH-1400 (Reference 3) which is 10^{-4} per reactor year.

The staff has independently determined that in the short term a sufficient level of safety exists for operating PWRs under conditions of a postulated pipe break (Reference 17). This was based on a simplified probabilistic approach that incorporates elastic fracture mechanics techniques to estimate the probability of the initiating event. Critical flaw size and subcritical flaw growth rates were determined assuming the presence of a surface flaw located in a circumferential weld of a thick walled pipe. The determination of a critical flaw size was based on an estimated fracture toughness value of K_{1C} at a minimum temperature of 200°F and a uniform tensile stress equal to the minimum material yield strength. Flaw growth rates were based on the considerations of various operating conditions producing elastically calculated stresses ranging in value from one to three times the minimum material yield strength. In using the calculated critical flaw size, the sub-critical growth rate, and an estimated probability distribution of an undetected flaw in a thick walled pipe weld, the upper bound probability of a pipe break was estimated to be at 10^{-4} per reactor year.

The low probability of the events requiring use of UPI provides additional justification for interim acceptance of the system without having a fully qualified UPI model.

Conclusions

The staff has reviewed the information submitted by the licensees and Westinghouse (Reference 15) and has concluded that the presently existing ECCS model as applied to two-loop plants is not adequate since it does not specifically consider all the effects manifested by UPI water.

However, the staff concludes that the plants can continue to operate safely without additional restrictions for a limited time, about 30 days, period without undue risk to the health and safety of the public. This conclusion is based on (1) recent data regarding decay heat and metal-water reaction rates that show that the approved ECCS models include more conservatism than was thought to exist when 10 CFR 50.46 and Appendix K to 10 CFR 50 were promulgated by the Commission, (2) actual plant power distribution is considerably more uniform than that assumed in the current ECCS analyses of the two-loop plants, (3) the low probability of a large LOCA, and (4) even if a large LOCA were to occur, this would not lead to the melting of a significant portion of the core.

Date: December 16, 1977

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2. Letter from D. B. Vassallo (NRC) to C. Eicheldinger (Westinghouse), dated May 30, 1975.

3. "Reactor Safety Study," WASH-1400 (NUREG-75/014), October 1975.

4. Harvego, E. A., "Quick Look Report on Semiscale Mod 1 Test S-05-3 Alternate ECC Injection Test Series," Semiscale Program December 1976.

5. Peterson, A. C., "Quick Look Report on Semiscale Mod 1 Test S-05-4 Alternate ECC Injection Test Series," Semiscale Program December 1976.

6. Harvego, E. A., "Quick Look Report on Semiscale Mod 1 Tests S-05-6 and S-05-7 Alternate ECC Injection Series," Semiscale Program March 1977.

7. Crapo, H. S., Collins B. L., and Sackett, K. E., "Experimental Data Report for Semiscale Mod 1 Tests S-04-5 and S-04-6 (Base Line ECC Tests)" TREE-NUREG-1045, January 1977.

8. Feldman, E. M., Patton, M. L. and Sackett, K. E., "Experimental Data Report for Semiscale Mod 1 Test S-05-3 (Alternate ECC Injection Test)" TREE-NUREG-1052, March 1977.

9. Feldman, E. M. and Sackett, K. E., "Experimental Data Report for Semiscale Mod 1 Tests S-05-6 and S-05-7 (Alternate ECC Injection Tests)" TREE-NUREG-1055, June 1977.

10. Mayinger, F., Hein, D. and Winkler, F., "Efficiency of Combined Cold and Hot Leg Injection" Paper for ANS Annual Meeting, June 1977.

11. WCAP-8480-NP, Revision 1, "Westinghouse Emergency Core Cooling Evaluation Model Application to Plants Equipped with Upper Head Injection," August 1976.

12. Telecon M. Grotenhuis (NRC) to R. Newton (Wisconsin Electric and Power Co.), January 3, 1977.

13. "Summary of Meeting Held on January 11, 1977, Regarding the ECCS Evaluation Model for Westinghouse Two-Loop Plants," M. Grotenhuis, January 21, 1977.

14. "Summary of Meeting Held on January 26, 1977, Regarding the ECCS Evaluation Model for Westinghouse Two-Loop Plants," M. Grotenhuis, February 28, 1977.

15. "A Review of Two-Loop Upper Plenum Injection Consideration" letter on Prairie Island Docket, L. O. Mayer to V. Stello, dated April 26, 1977.

16. "Safety Evaluation Report on ECCS Evaluation Model for Westinghouse Two-Loop Plants," December 15, 1977.

17. Probabilistic Analysis Report on Reactor Vessel Supports, ACRS Subcommittee on Fluid/Hydraulic Dynamic Effects, Los Angeles, California, May 26, 1977.

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