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A Study of an Unintentional Shutdown (Scram) Problem in the Operation of the High Flux Isotope Reactor

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A STUDY OF AN UNINTENTIONAL SHUTDOWN (SCRAM) PROBLEM IN THE OPERATION OF THE HIGH FLUX ISOTOPE REACTOR

by

Cheng Ping Wu

A report submitted in partial fulfillment
of the requirements for the degree

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Cheng Ping Wu

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by

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Utah State University, 1970

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The High Flux Isotope Reactor (HFIR) at the Oak Ridge National Laboratory experiences occasional operating difficulties caused by temporary low voltage conditions on its main power feeder line. The low voltage causes the shutdown of one or more of the three primary coolant pumps. When two of these pumps are shut down, the capacity of the cooling system is reduced to the point where the reactor is scrammed due to a high inlet coolant temperature.

This study was made to determine the cause of the high inlet temperature when operating with one primary coolant pump and to suggest possible changes in the system or its operation to eliminate the undesired shutdowns.

Three methods are suggested for elimination or reduction

of the scrams: (1) Cross connect the suction side of the primary coolant pumps so any pump can circulate coolant through all three heat exchangers; (2) Reduce the power to flow ration of the reactor if the coolant flow drops below a set value; (3) Put two primary pumps and one secondary pump on each of the two feeder power lines to reduce the possibility of having more than one pump shut down. Combinations of these methods are also possible.

(56 pages)

CHAPTER I

INTRODUCTION

The High Flux Isotope Reactor (HFIR) is a research reactor built primarily to produce experimental quantities of the known elements, such as californium, which are heavier than plutonium. Its exceptionally high neutron flux provides a sufficient population of neutrons to produce more than trace quantities of the transuranium isotopes with exceptionally short half-lives and whose production rate by neutron capture is low.

The HFIR utilizes a flux trap fuel assembly designed specifically to produce an intense flux of thermal neutrons (5×10^{15} cm^{-2} sec^{-1}) at a thermal power level of 100 Mw. The basic system components which make up the HFIR are: the reactor core, the pressure vessel, the primary coolant system, the secondary coolant system, the primary coolant cleanup system, and the pool coolant system, as shown in Figures 1 and 2.

The reactor core, shown in Figure 3, is composed of two concentric, cylindrical fuel elements. The plates are 0.05 inches thick and 24 inches long. The inner element, which contains 171 plates, has a 5 inch inner diameter and a 10 1/2 inch outer diameter. The outer element has inside and outside diameters of 11 and 17 1/8 inch, respectively. Each fuel assembly contains approximately 9.4 kg of ^{235}U. The fuel region is centered within a cylindrical beryllium reflector approximately 1 foot thick. This, in turn, is surrounded by a water reflector of effectively infinite thickness.

The 5 inch diameter center of the fuel assembly is used for irradiation of target materials. Water in this area forms the "flux trap" by slowing the neutrons originating in the encircling uranium fuel. The trap makes available more thermal neutrons per second for bombarding targets than can be obtained with standard research reactors. This high neutron flux hastens the transmutation of materials under bombardment.

The reactor core components, which include the fuel assembly, control plates, beryllium reflector, and target array, are contained in the reactor vessel. They are supported on two concentric cylinders, or pedestals, which are bolted to a cylindrical piece known as the fuel and reflector support assembly. This, in turn, is bolted to the lower part of the reactor vessel. A pair of cylindrical shrouds extends above the top of the fuel element and enclose the upper part of the control cylinders. This provides the required coolant flow behavior and protects the control cylinders from the turbulence of the vessel inlet coolant stream.

The primary coolant system, with which this project is more concerned, is shown in Figure 4 and consists of the reactor vessel,

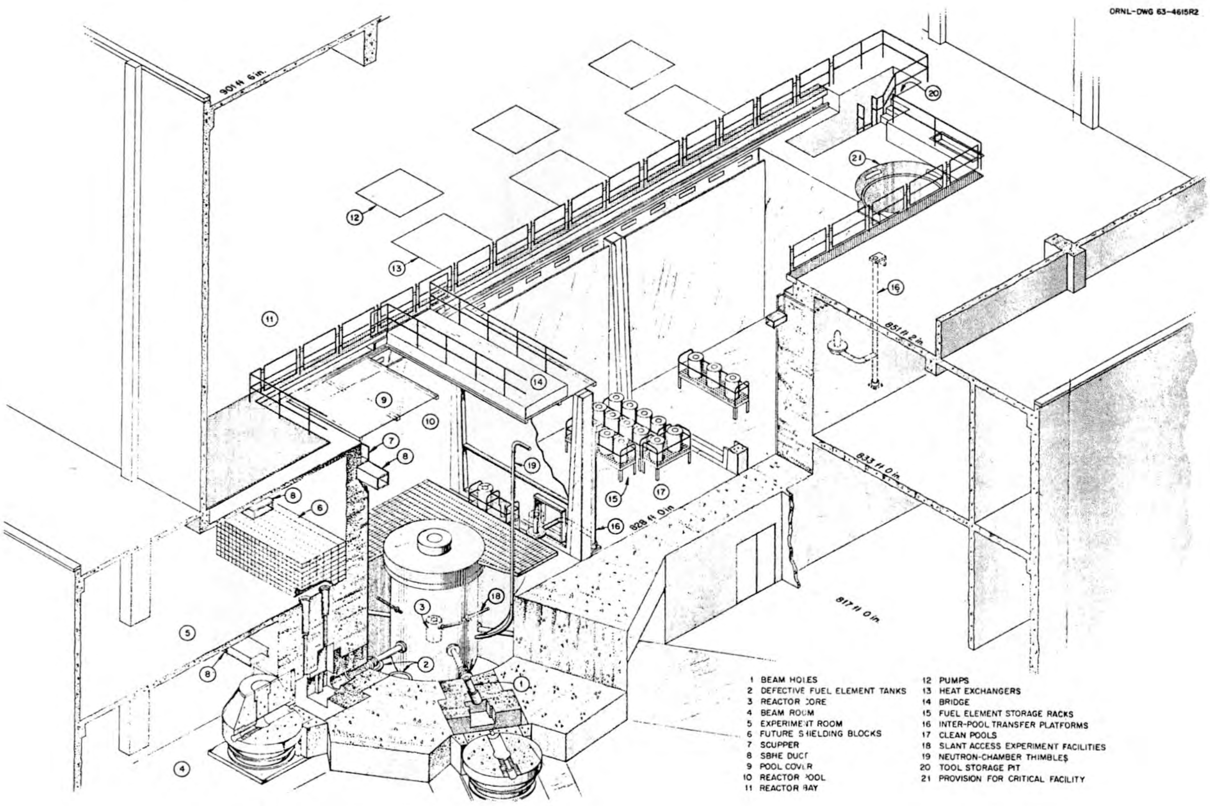

Figure 1. Isometric of pools and experimental facilities.

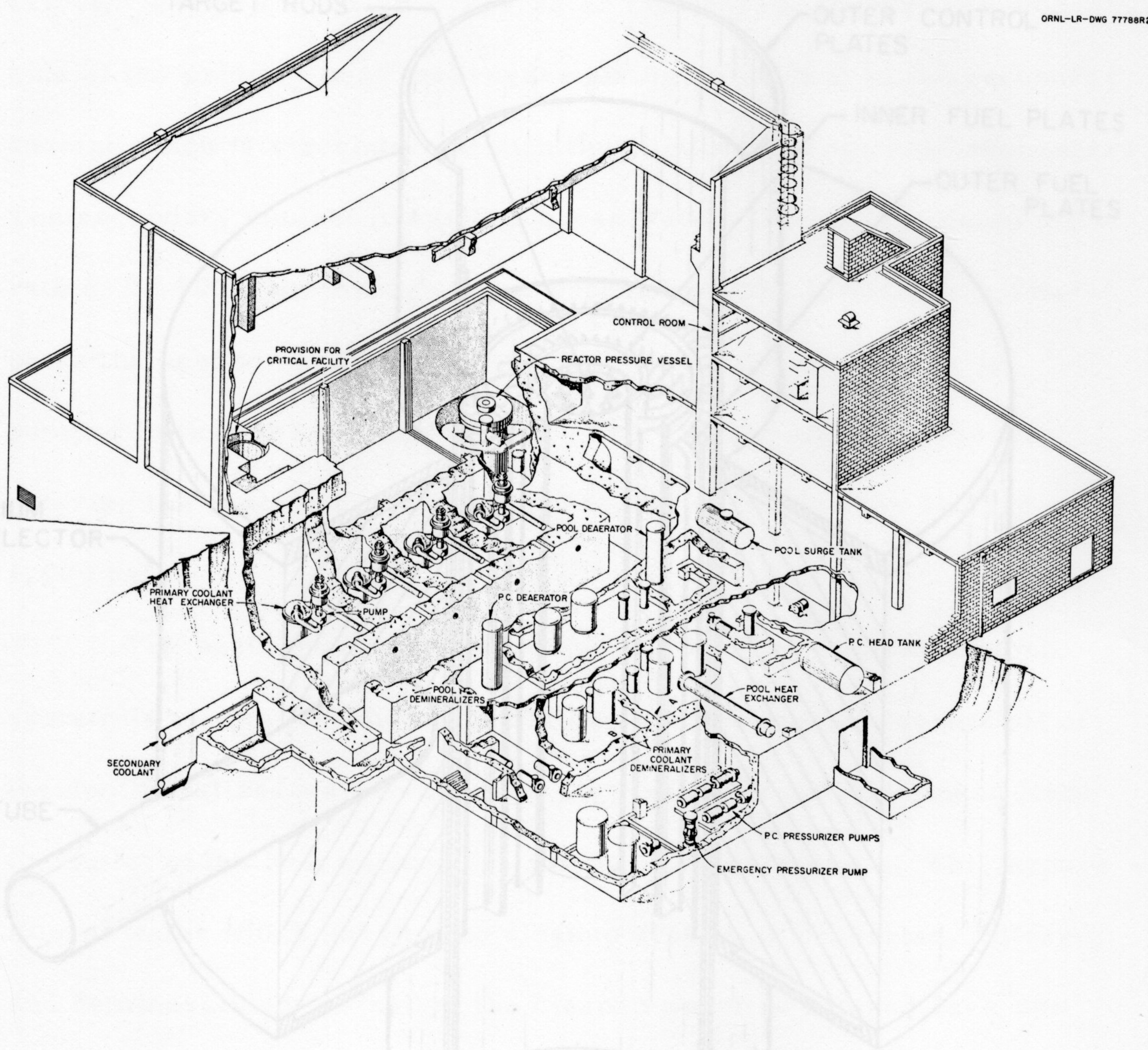

Figure 2. Isometric of water and control wing.

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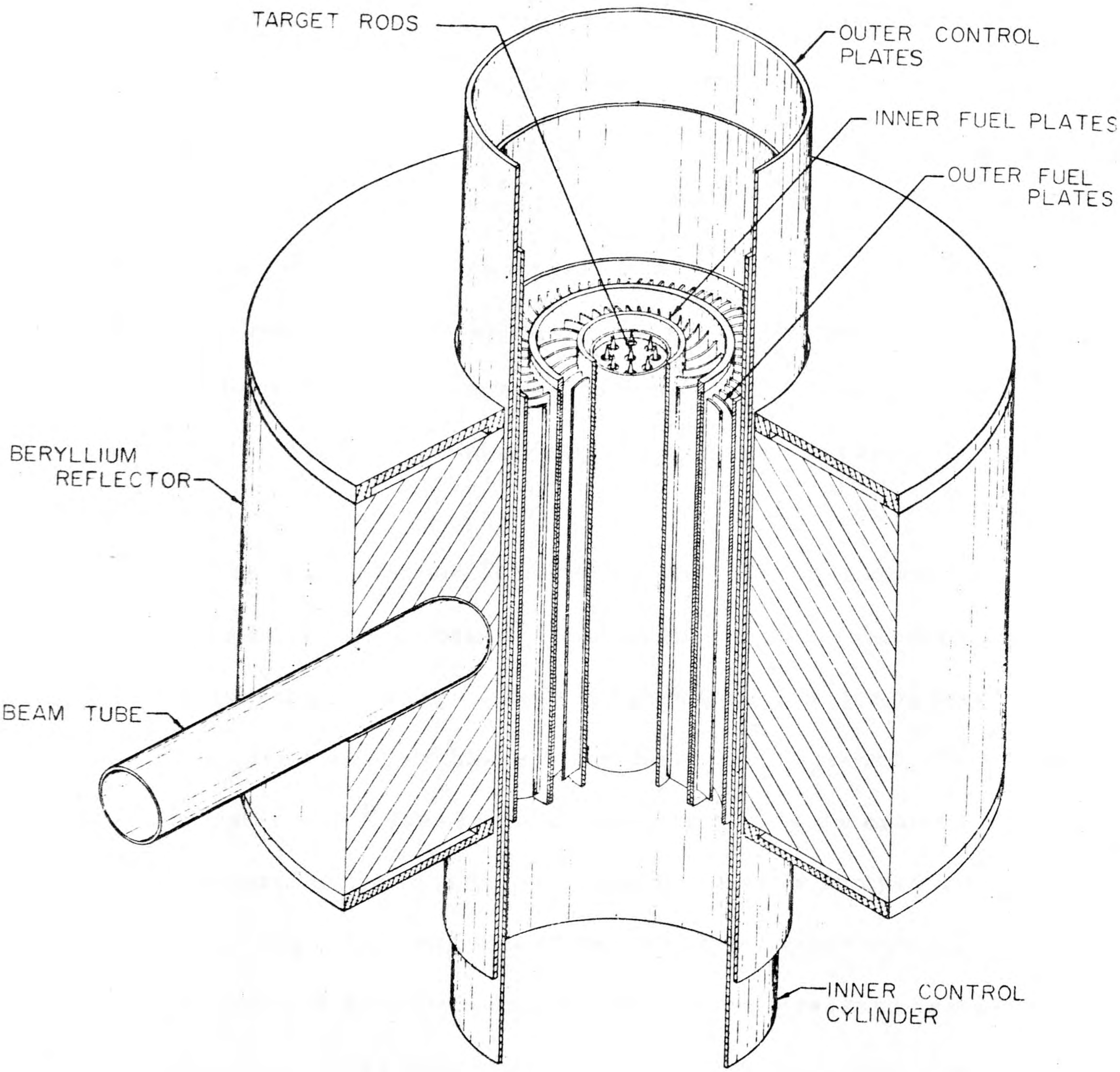


Figure 3. Illustration of core.

three heat exchangers, three primary coolant pumps, and the associated piping and control valves. The HFIR core is cooled by demineralized water circulated through the reactor vessel at the rate of 15,800 gpm at full reactor power. Design inlet temperature is 120° F, and exit temperature is 167°F. The water then passes through the tube side of the primary heat exchangers giving up its heat to the secondary coolant which is circulated through the shell side of the heat exchangers. The secondary coolant (treated process water) then is circulated at the rate of 22,500 gpm through a conventional cooling tower which dissipates the heat to the atmosphere. The cooled primary water is pumped back to the reactor vessel. The primary coolant pressure entering the reactor can be adjusted to any value between 300 and 1,000 psi.

The basic purpose of providing the primary coolant cleanup system is to remove contamination and to maintain the correct pH of the demineralized water. About 200 gpm flow of water is taken from the outlet of the heat exchangers for the cleanup system. This bypass flow of water which goes to the cleanup system is deaerated, filtered, and demineralized. Finally, the cleanup water is pumped back into the 16 inch diameter inlet line of the primary coolant system. The core is enclosed by a pressure vessel, and the pressure vessel is immersed in a water pool which provides biological shielding.

In the next section, the hot slug problem in the primary coolant

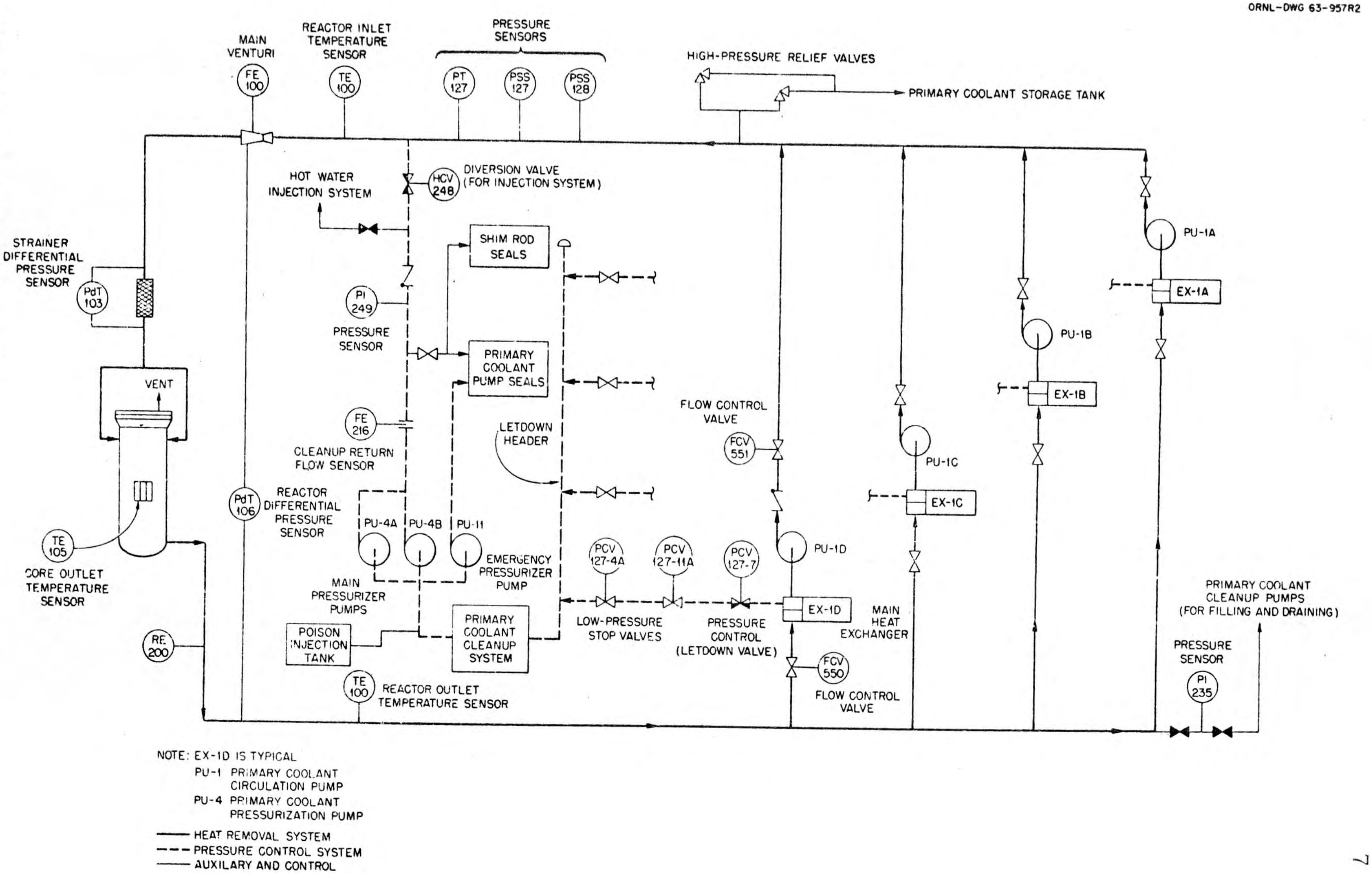


Figure 4. Primary coolant system.

system will be described, and subsequently possible suggestions will be made for improvement in the reactor coolant system and/or control system to reduce this problem.

CHAPTER II

DISCUSSION OF THE PRIMARY COOLANT SYSTEM AND THE HOT SLUG PROBLEM

The flow of demineralized primary coolant water enters the HFIR pressure vessel through two 16 inch pipes from either side, as shown in the simplified schematic flow diagram in Figure 5. The design temperature and pressure of the flow at the inlet are $120^{o}F$ and 600 psi, respectively. The water flowing at the rate of 15,800 gpm passes through the reactor core and exits through the 18 inch pipe under the core. The outlet temperature and pressure of the exit water are $167^{o}F$ and 526 psi, respectively. Thus there is a pressure drop of 74 psi across the core. The water flowing at this rate, temperature, and pressure are then distributed to three of the four identical parallel heat-exchanger circulation-pump combinations. The temperature of the water coming out of these heat exchangers is $120^{o}F$. This water is then pumped back by the centrifugal pumps into the inlet pipe (3, 1).

The primary cooling system is designed to remove essentially all the energy from the core. The vertical shaft centrifugal primary coolant pumps take their suction from individual heat exchangers. Each pump is located in the same cell as its associated heat exchanger, as shown in Figure 6. Each main pump delivers about 5,000 gpm of water against a 365 foot head of water when three pumps are operated in parallel. Each pump is driven by a 600 hp squirrel cage induction motor, which operates on 2400 volt a.c. Besides the a.c. motor driving each pump there is a d.c. "pony" motor directly coupled to the shaft. The d.c. pony motor is provided to maintain approximately 10 percent coolant flow in case of a power outage which causes the a.c. motor to stop functioning.

In all of the following discussion it is assumed that the secondary coolant system will never fail because of the diesel power which comes on in the event of failure of the main power. If in any case the secondary coolant system stops functioning completely then there is no way in which sufficient heat can be transferred from the primary to the secondary coolant system and a scram (shutdown) will occur.

There are various possibilities that are encountered due to a transient drop in the power supply voltage. The temporary time period for which voltage is reduced determines how many pumps will be shut off. If the low voltage is less than one-tenth of a second then no pump will stop and the system will remain in normal operation. If the power outage is between one-tenth of a second and one second then one of the three primary pumps will be put out of operation. Since two primary pumps are still operating at their full power,

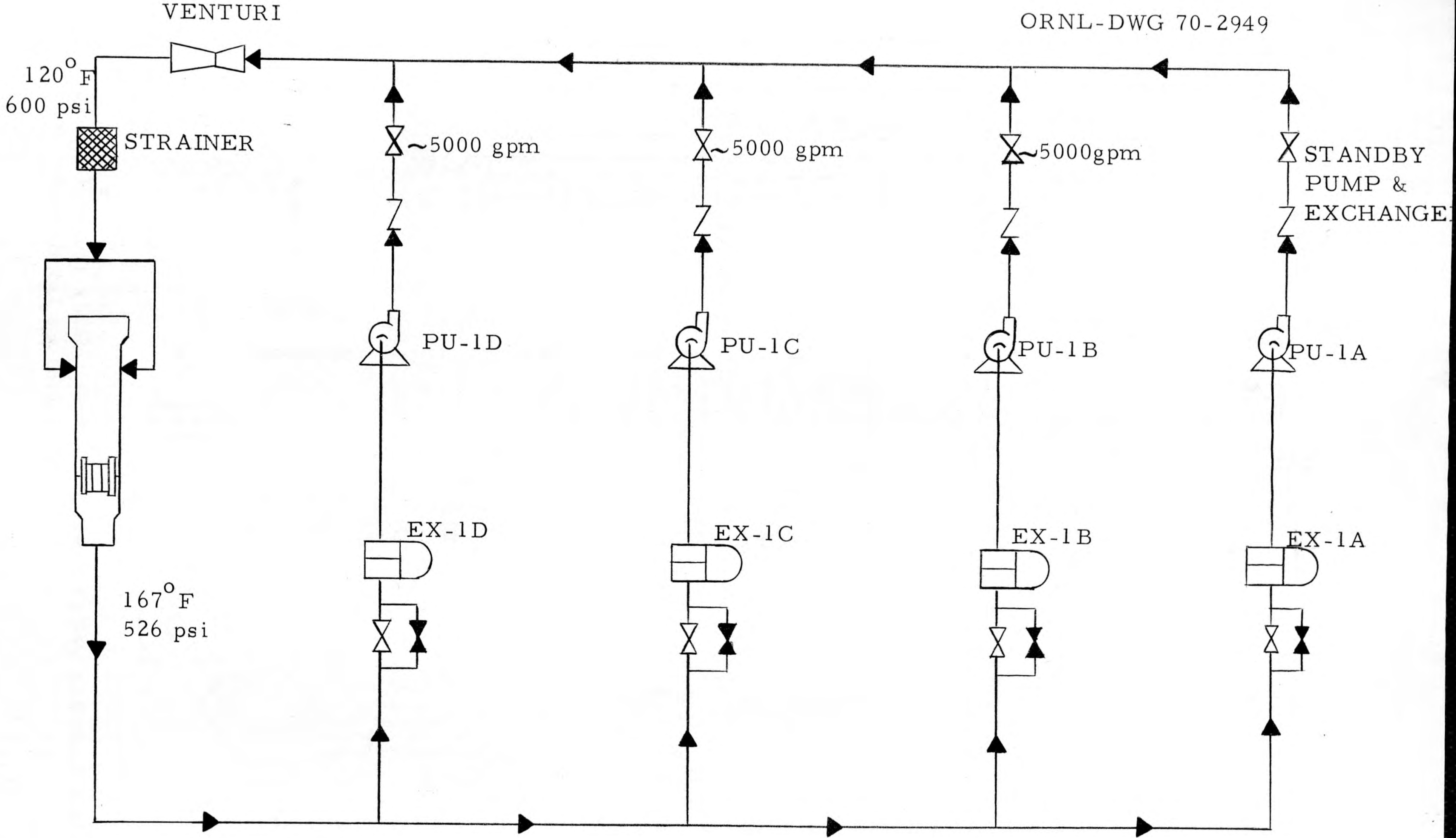


Figure 5. Simplified schematic flow diagram of primary coolant system.

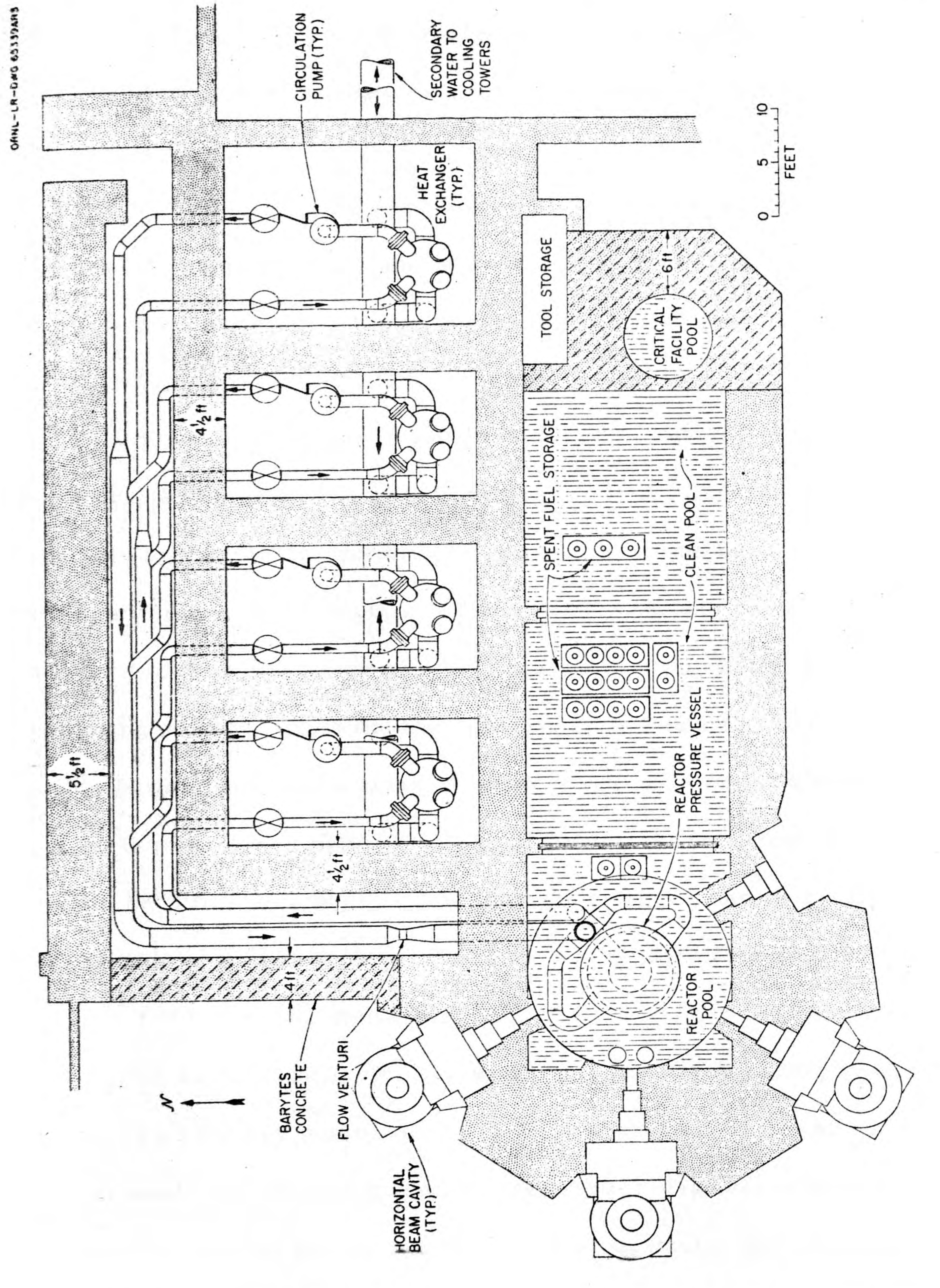


Figure 6. Reactor shield, heat exchanger cells, and pool structures--horizontal section.

there is still enough circulation of water to maintain the temperature of the coolant below the reactor inlet high-temperature trip level. Moreover, due to the automatic power regulation feature which functions according to flow rate the power of the reactor is reduced in this case. If the power outage is for more than one second, but less than a few seconds, then there is a possibility that two or three primary pumps will go off (5).

One possibility due to temporary power outage is that all three primary pumps will go off and coast down. This causes the flow rate to drop to approximately 10 percent of the normal amount in the primary coolant system. The reduction in the coolant flow reduces the thermal power proportionally by an automatic power regulating device. Thus the thermal power is reduced to 10 Mw. At this power level circulation of water in the primary coolant system is maintained by the three d. c. pony motors. Once the reactor becomes steady at 10 Mw thermal power it can be returned to operation at a higher power level by systematically restoring the power to the main a. c. motors.

The worst case is the one in which, due to temporary power outage, two primary pumps out of three are shut off while the third pump is still running at full power. Since each heat exchanger is connected to its own pump, the failure of a particular pump causes the loss of its associated heat exchanger. When two pumps are put out of operation the total area for heat exchange is reduced to one-third of the normal value, and thus water flowing out of the heat exchanger exceeds the allowable reactor inlet temperature. This hot water which causes this problem is known as a "hot slug." All of this happens in a matter of a few seconds. When the hot slug reaches the inlet temperature sensor it causes a reactor scram.

Because of the short life (23 days), high cost of the HFIR fuel element, and the high rate of growth of xenon and samarium after shutdown, it is desirable to keep the reactor at a power as high as possible, even during the abnormal condition discussed above. In the following section some system and control modifications are suggested which should accomplish this objective.

CHAPTER III

SUGGESTED MODIFICATIONS TO REDUCE THE HOT SLUG PROBLEM

The HFIR consists of various systems and processes. Solutions to the hot slug problem, which occurs solely in the primary coolant system, can be looked at from three different approaches: (1) fluid dynamics and heat transfer; (2) reactor control; and (3) electrical circuitry.

Looking from the fluid dynamics and heat transfer standpoint, it is possible that a scram can be prevented if the primary coolant flow system is modified. As it was mentioned before and shown in Figure 5, each primary coolant pump delivers about 5,000 gpm when operating in parallel at the normal operation condition of the reactor. Now if a power outage prevails for more than one second, let us suppose two out of three pumps will coast down while the third pump is still running at its full power. When the two pumps have coasted down their associated heat exchangers will not function, and so the water coming out of the core at a temperature of 167°F will not completely cool down to 120°F. If, however, the primary coolant outlets of the heat exchangers are cross connected by a pipe, as shown in Figure 7, then the surface area for heat transfer will be greatly increased since the water will circulate through three heat exchangers instead of one. This change in the configuration of the pipe connections will assure that the temperature of the water returning to the reactor inlet is less than 135^{o}F (scram value) as shown in the following calculation (6, 4).

In normal operation the total primary loop pressure loss is 365 feet head of water (based on the pump characteristic curve in Figure 8), or

$$\Delta P = \frac{14.7}{34} \times 365 = 158 \text{ psi.}$$

In case two pumps are shut down due to power outage, flow in the primary coolant system is 9, 500 gpm. At this flow the pump head is 230 feet of water, or approximately 100 psi.

Let

ΔP_P = primary pump pressure drop at 5, 270 gpm flow rate,

$\Delta P'_P$ = primary pump pressure drop at 9, 500 gpm flow rate,

ΔP_R = reactor system pressure drop at 15, 800 gpm flow rate,

$\Delta P'_R$ = reactor system pressure drop at 9, 500 gpm flow rate,

ΔP_H = heat exchanger system pressure drop at 5, 270 gpm flow rate,

$\Delta P'_H$ = heat exchanger system pressure drop at 9, 500 gpm flow rate,

C_R = reactor pressure coefficient (assumed constant),

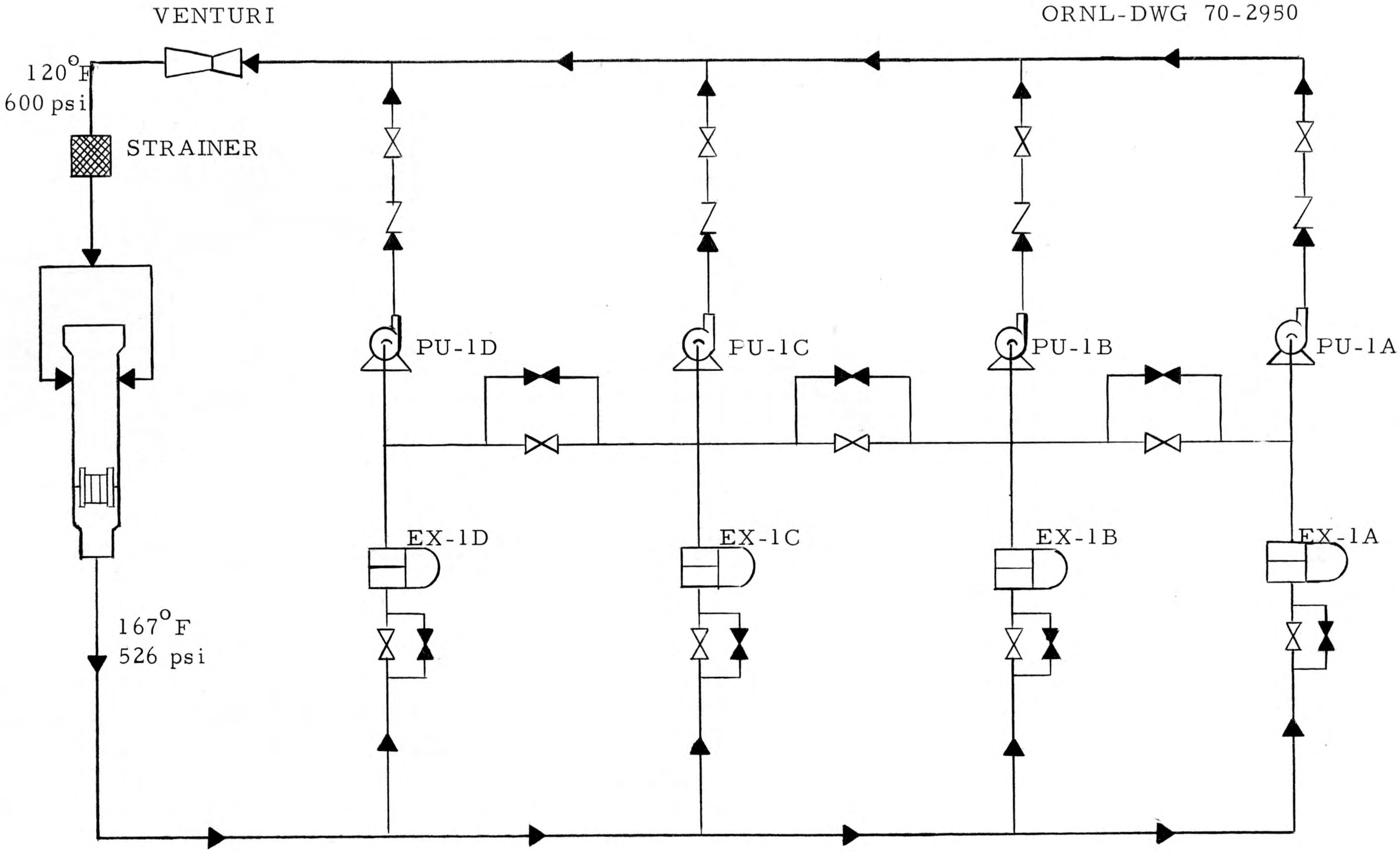


Figure 7. Modified schematic flow diagram of primary coolant system.

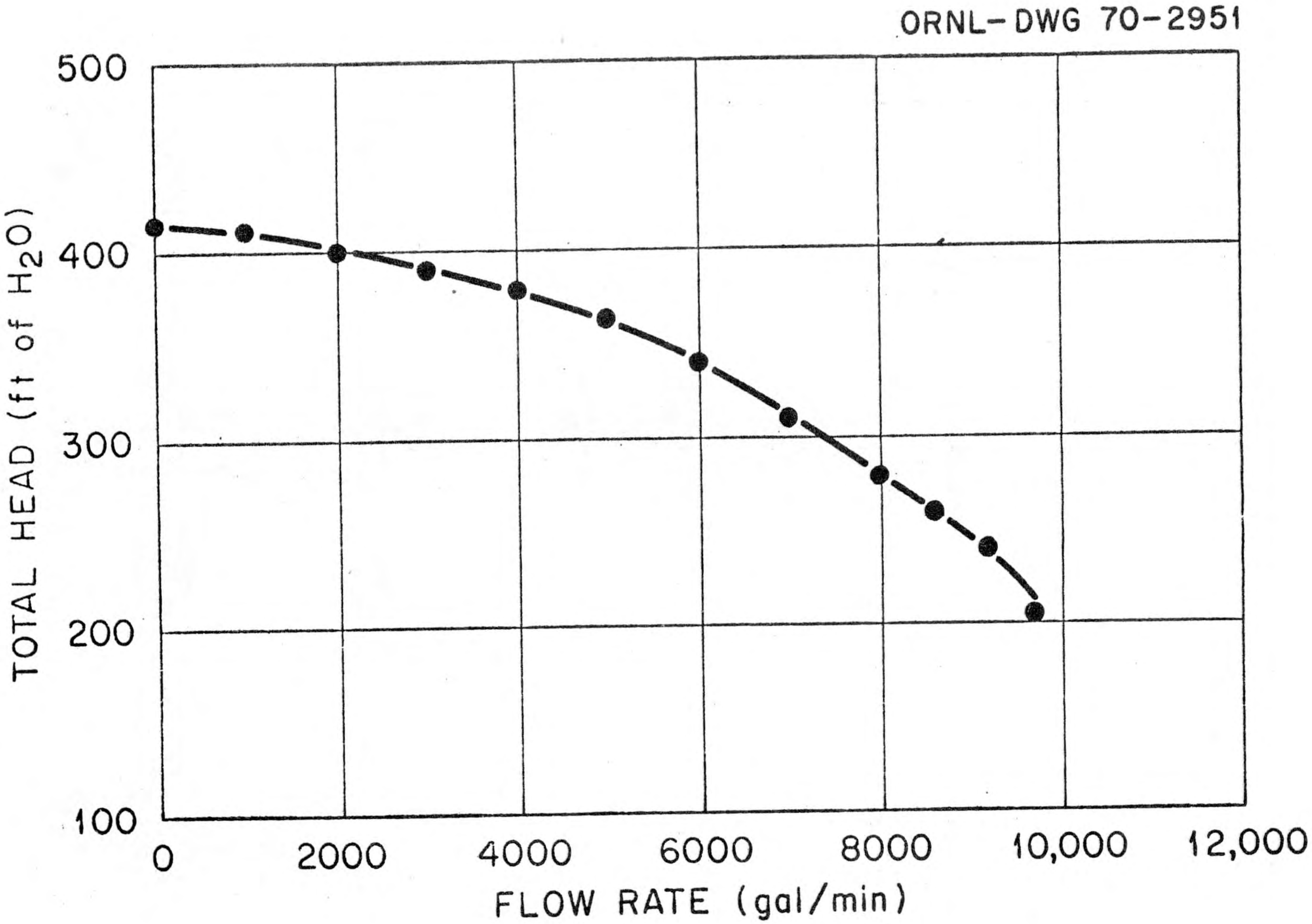


Figure 8. Flow versus head characteristic for primary pumps.

C_H = heater exchanger pressure coefficient (assumed constant),

V_R = velocity of coolant through reactor at 15,800 gpm,

V'_R = velocity of coolant through reactor at 9,500 gpm,

V_H = velocity of coolant through heat exchanger at 5,270 gpm, and

V'_H = velocity of coolant through heat exchanger at 9,500 gpm.

We have the following equations:

$$\Delta P_P = \Delta P_R + \Delta P_H , \qquad [1]$$

$$\Delta P'_P = \Delta P'_R + \Delta P'_H . \qquad [2]$$

But

$$\Delta P'_R = C_R (V'_P)^2 = \Delta P_R \left(\frac{V'_R}{V_R}\right)^2 = \Delta P_R \left(\frac{9,500}{15,800}\right)^2$$

$$= 0.36\, \Delta P_R , \qquad [3]$$

$$\Delta P'_H = C_H (V'_H)^2 = \Delta P_H \left(\frac{V'_H}{V_H}\right)^2 = \Delta P_H \left(\frac{9,500}{5,270}\right)^2$$

$$= 3.24\, \Delta P_H . \qquad [4]$$

Substituting into Eqs. [1] and [2],

$$\Delta P_R + \Delta P_H = 158 \text{ psi}, \qquad [5]$$

$$0.36\ \Delta P_R + 3.24\ P_H = 100 \text{ psi}. \qquad [6]$$

Solving Eqs. [5] and [6], we obtain,

$$\Delta P_H + 14.97 \text{ psi} \cong 15 \text{ psi},$$

and

$$\Delta P_R = 158 - 15 = 143 \text{ psi}.$$

When the cross-connection pipe is added to the heat exchanger outlets, with one pump running, the flow rate is increased slightly, i.e., to 9,750 gpm, which is the flow limit of the pump at the speed available.

Let

$\Delta P''_R$ = reactor pressure drop at 9,750 gpm flow rate,

$\Delta P''_H$ = heat exchanger pressure drop at 3,250 gpm flow rate (9,750/3 = 3,250),

$\Delta P''_H$ = primary pump pressure drop at 9,750 gpm.

By analogy with Eqs. [3] and [4] we get:

$$\Delta P''_R = \Delta P_R \left(\frac{V''_R}{V_R}\right)^2 = 143 \left(\frac{9,750}{15,800}\right)^2 = 143 \times 0.378 \cong 53 \text{ psi},$$

$$\Delta P''_H = \Delta P_H \left(\frac{V'_H}{V_H}\right)^2 = 15\left(\frac{3,250}{5,270}\right)^2 = 15 \times 0.379 \cong 6 \text{ psi},$$

$$\Delta P''_P = \Delta P''_R + \Delta P''_H = 53 + 6 = 59 \text{ psi}.$$

When the flow rate is increased to 9, 750 gpm the pressure drop of the primary pump is 205 ft head of water (see Figure 8), or

$$\frac{14.7}{34} \times 205 = 0.432 \times 205 = 87 \text{ psi}.$$

So one primary pump can deliver coolant up to 9, 750 gpm when a cross-connecting pipe has been added.

For the following calculations refer to Figure 9, 10, and 11. For a heat exchanger:

$$Q = UA\Delta T,$$

where

Q = heat transfer rate, Btu/hr,

U = heat transfer coefficient, Btu/hr ft^2 $^\circ$F,

A = heat transfer area, ft^2,

ΔT = average temperature difference between fluid streams, $^\circ$F.

It should be noted that U varies with the flow velocity. Assuming all the thermal resistance is on the tube side, which is very conservative, for the following calculations:

$$U_2 \approx U_1 \left(\frac{v_1}{v_2}\right)^{0.8} .$$

In case only one pump is working, and no cross-connection pipe is used,

$$U_2 = U_1 \left(\frac{9,500}{5,270}\right)^{0.8} = U_1\,(1.8)^{0.8} = 1.6\,U_1\,,$$

and

$$Q_1 = U_1\,A(\Delta\bar{T})_1\,,$$

$$Q_2 = U_2\,A(\Delta\bar{T})_2\,,$$

$$\frac{Q_2}{Q_1} = \left(\frac{U_2}{U_1}\right)\frac{(\Delta\bar{T})_2}{(\Delta\bar{T})_1}\,,$$

$$\frac{60}{33.33} = 1.6\,\frac{(\Delta\bar{T})_2}{(\Delta\bar{T})_1}\,,$$

$$\Delta T_2 = \frac{60}{33.33} \times \frac{45}{1.6} \cong 51^{\circ}F,$$

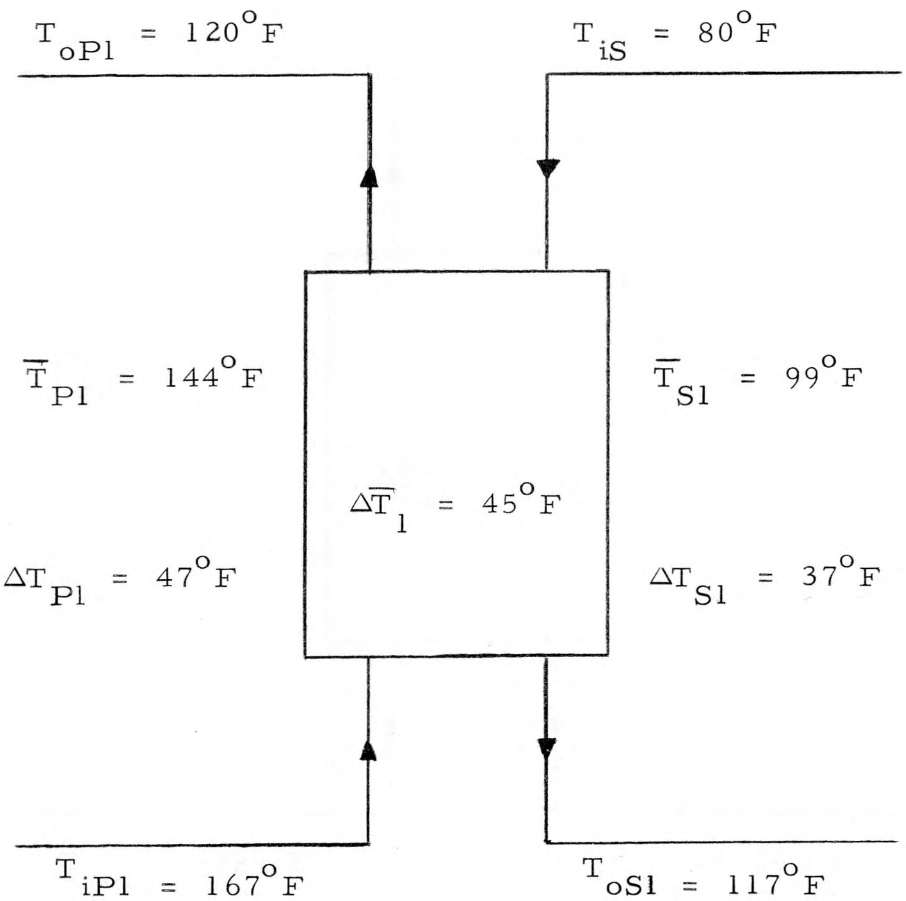
$$\Delta T_{S2} = \frac{60}{33.3} \times \Delta T_{S1} = \frac{60}{33.33} \times 37 = 67^{\circ}F,$$

$$T_{oS2} = T_{iS} + T_{S2} = 80 + 67 = 147^{\circ}F,$$

$$\bar{T}_{S2} = \frac{T_{iS} + T_{oS2}}{2} = \frac{80 + 147}{2} = 114^{\circ}F,$$

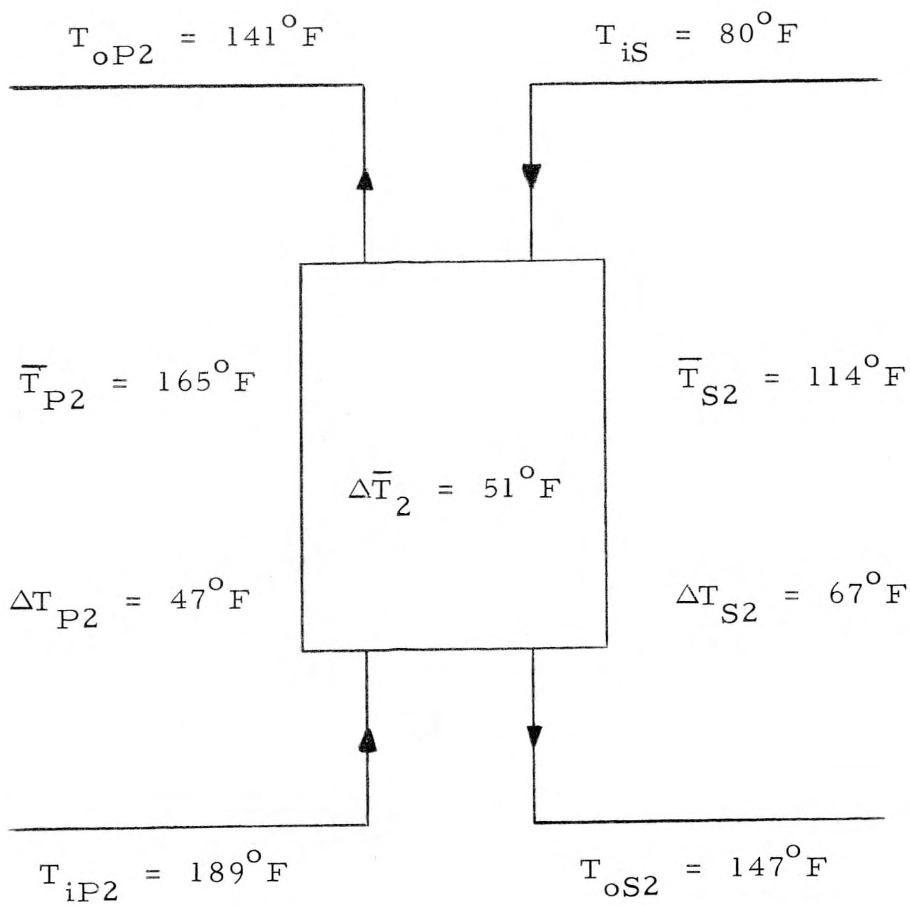
$$\bar{T}_{P2} = \bar{T}_{S2} + \Delta\bar{T}_2 = 114 + 51 = 165^{\circ}F,$$

$$\Delta T_{P2} = \frac{6,700}{9,500}\,(67) = 47^{\circ}F,$$



Primary Flow Rate = 5270 gpm   Secondary Flow Rate = 6700 gpm

Figure 9. Schematic summary of heat exchanger flow rates and temperatures in normal full-power operation.



Primary Flow Rate = 9500 gpm

Secondary Flow Rate = 6700 gpm

Figure 10. Schematic summary of heat exchanger flow rates and temperatures with one primary pump operating and no system modification.

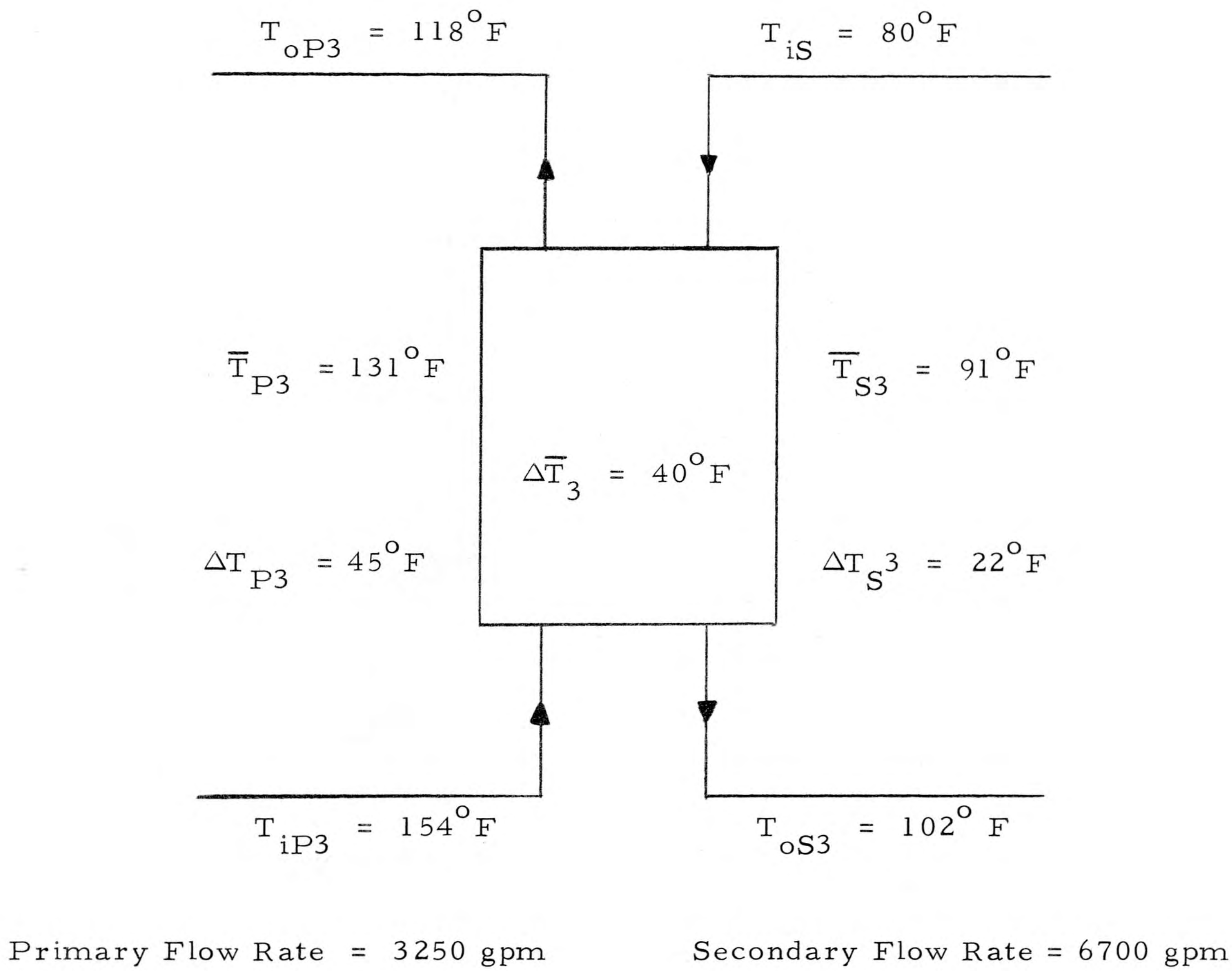


Figure 11. Schematic summary of heat exchanger flow rates and temperatures with one primary pump operating and all heat exchanger exits cross-connected.

$$T_{iP2} = \overline{T}_{P2} + \frac{\Delta T_{P2}}{2} = 165 + \frac{47}{2} = 189^{\circ}F,$$

$$T_{oP2} = \overline{T}_{P2} - \frac{\Delta T_{P2}}{2} = 165 - \frac{47}{2} = 141^{\circ}F.$$

Hence even with the very conservative assumption regarding the heat transfer coefficient, the primary coolant temperature at the exit of the heat exchanger is $141^{\circ}F$, which is greater than the scram value ($135^{\circ}F$). The same type calculations with $U_2 = U_1$ yield $T_{oP2} = 195^{\circ}F$.

Once the cross connecting pipe has been added, the coolant flow through each of the heat exchangers is approximately 3,250 gpm and

$$\frac{U_3}{U_1} = \left(\frac{3,250}{5,270}\right)^{0.8} = 0.68\ ,$$

$$\frac{Q_3}{Q_1} = \frac{60/3}{33.33} = \frac{20}{33.33} = 0.60,$$

$$\frac{Q_3}{Q_1} = \frac{U_3}{U_1}\ \frac{\Delta\overline{T}_3}{\Delta\overline{T}_1}\ ,$$

$$\Delta T_3 = \frac{0.60}{0.68} \times 45 = 40^{\circ}F,$$

$$\Delta T_{S3} = T_{oS3} - T_{iS} = \frac{Q_3}{Q_1}\ \Delta T_{S1} = 0.60 \times 37 - 22^{\circ}F,$$

$$T_{oS3} = T_{iS} + \Delta T_{S3} = 80 + 22 = 102^{\circ}F,$$

$$\overline{T}_{S3} = \frac{102 + 80}{2} = 91^{o}F,$$

$$\overline{T}_{P3} = T_{S3} + \Delta T_{3} = 91 + 40 = 131^{o}F,$$

$$\Delta T_{P3} = \frac{\text{Secondary flow rate through heat exchanger}}{\text{Primary flow rate through heat exchanger}} \times \Delta T_{S3}$$

$$= \frac{6,670}{3,250} \times 22 = 45^{o}F,$$

$$T_{iP3} = \overline{T}_{P3} + \frac{\Delta T_{P3}}{2} = 131 + \frac{45}{2} = 154^{o}F,$$

$$T_{oP3} = \overline{T}_{P3} - \frac{\Delta T_{P3}}{2} = 131 - \frac{45}{2} - 118^{o}F.$$

Hence it is seen that with the pipe cross connection the temperature of the water coming out of the heat exchangers is less than $135^{o}F$ (scram value).

U_1 = transfer coefficient at 5, 270 gpm flow rate,

U_2 = transfer coefficient at 9, 500 gpm flow rate,

U_3 = transfer coefficient at 3, 250 gpm flow rate,

Q_1 = heat transfer rate at 5, 270 gpm flow rate,

Q_2 = heat transfer rate at 9, 500 gpm flow rate,

Q_3 = heat transfer rate at 3, 250 gpm flow rate,

$\Delta\overline{T}_1$ = average temperature difference between primary coolant and secondary coolant at 5, 270 gpm flow rate,

$\Delta\overline{T}_2$ = average temperature difference between primary coolant and secondary coolant at 9,500 gpm flow rate,

$\Delta\overline{T}_3$ = average temperature difference between primary coolant and secondary coolant at 3,250 gpm flow rate,

$\overline{T}_{P1}$ = average temperature of primary coolant inside the heat exchanger at 5,270 flow rate,

$\overline{T}_{P2}$ = average temperature of primary coolant inside the heat exchanger at 9,500 gpm flow rate,

$\overline{T}_{P3}$ = average temperature of primary coolant inside the heat exchanger at 3,250 gpm flow rate,

T_{iP1} = inlet temperature of primary coolant at 5,270 gpm flow rate,

T_{iP2} = inlet temperature of primary coolant at 9,500 gpm flow rate,

T_{iP3} = inlet temperature of primary coolant at 3,250 gpm flow rate,

T_{oP1} = outlet temperature of primary coolant at 5,270 gpm flow rate,

T_{oP2} = outlet temperature of primary coolant at 9,500 gpm flow rate,

T_{oP3} = outlet temperature of primary coolant at 3,250 gpm flow rate,

T_{iS} = inlet temperature of secondary coolant at 6,700 gpm flow rate,

T_{oS1} = outlet temperature of secondary coolant when primary flow rate is 5,270 gpm,

T_{oS2} = outlet temperature of secondary coolant when primary flow rate is 9,500 gpm,

T_{oS3} = outlet temperature of secondary coolant when primary flow rate is 3,250 gpm,

$\bar{T}_{S1}$ = average temperature of secondary coolant inside the heat exchanger when primary flow rate is 5,270 gpm,

$\bar{T}_{S2}$ = average temperature of secondary coolant inside the heat exchanger when primary flow rate is 9,500 gpm,

$\bar{T}_{S3}$ = average temperature of secondary coolant inside the heat exchanger when primary flow rate is 3,250 gpm,

ΔT_{P1} = primary flow temperature drop between inlet and outlet at 5,290 gpm,

ΔT_{P2} = primary flow temperature drop between inlet and outlet at 9,500 gpm,

ΔT_{P3} = primary flow temperature drop between inlet and outlet at 3,250 gpm,

$\Delta \bar{T}_{S1}$ = secondary flow temperature increase between inlet and outlet when primary flow rate is 5,290 gpm,

$\Delta\overline{T}_{S2}$ = secondary flow temperature increase between inlet and outlet when primary flow rate is 9,500 gpm,

$\Delta\overline{T}_{S3}$ = secondary flow temperature increase between inlet and outlet when primary flow rate is 3,250 gpm.

The second possible solution to the hot slug problem is from the control point of view and consists of modifying the thermal power versus coolant flow rate characteristic. The HFIR control and safety systems consist of three channels of instrumentation with concurrence of any two out of three control channels required for any corrective action. Coolant flow rates, temperature, pressure, and measured neutron flux are the parameters of the three channels. In the normal operation thermal power of the reactor is controlled by the flow rate in the closed loop primary coolant system, that is the flux-to-flow ratio (Ø/F) is held constant. The variation of the reactor thermal power with flow rate is shown in Figure 12. Thus, if for some reason the flow rate reduces to 60 percent, then thermal power also is reduced to 60 percent of the full power.

Due to temporary power outage there is the possibility of shutting down only one pump. However, this is not a real problem because there is still enough flow and heat transfer area in the primary coolant system to keep the Ø/F ratio unchanged without causing a scram. But if two pumps fail instead of one, then the flow rate is reduced to 60 percent and the heat transfer area is reduced to 33-1/3 percent. This pump failure causes the hot slug scram problem discussed above. It is possible to prevent scram due to this condition if the thermal power varies according to new characteristics as shown in Figure 11, below a certain preset value of flow rate. Since the coolant flow rate with two pumps running is approximately 85 percent of the full flow rate, this parameter can be used to cause a switch in the Ø/F ratio below, say, 75 to 80 percent flow rate. Such a flow reduction indicates the loss of more than one pump, and using the new Ø /F ratio curve would reduce the power to an acceptable value for one heat exchanger.

A third possible solution or, to be exact, partial solution to the hot slug problem involves modification of the electrical power supply setup.

The HFIR is supplied with electricity by four semi-independent systems which are illustrated in Figure 12. These four systems are identified as:

1. The normal power supply system, which handles the normal electrical power requirements. This system is supplied from TVA power from one of two independent 13.8 kv feeders.
2. The two normal emergency systems which are designed to assume certain electrical loads during an outage of the normal power system.

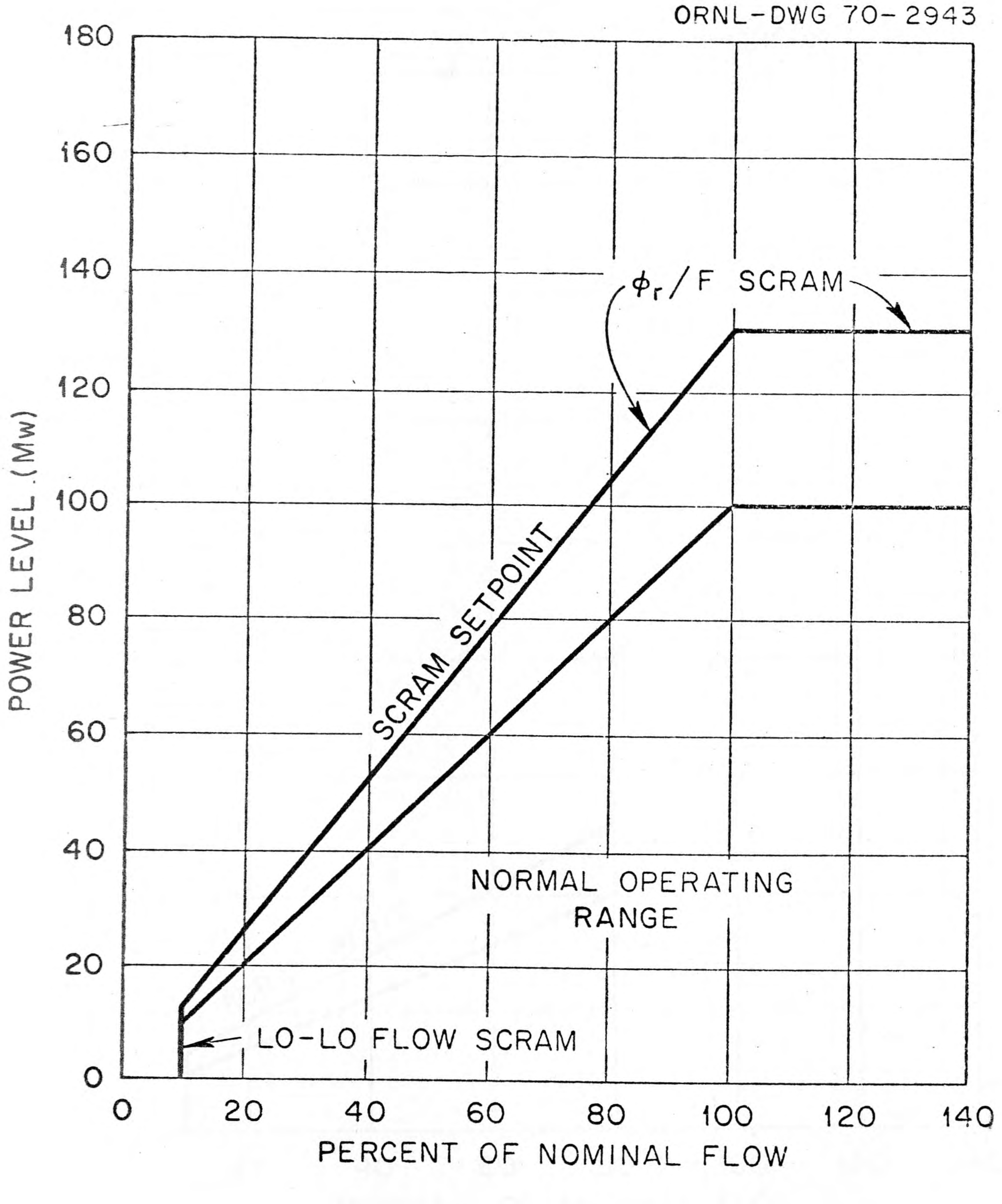


Figure 12. Characteristics of flow versus power.

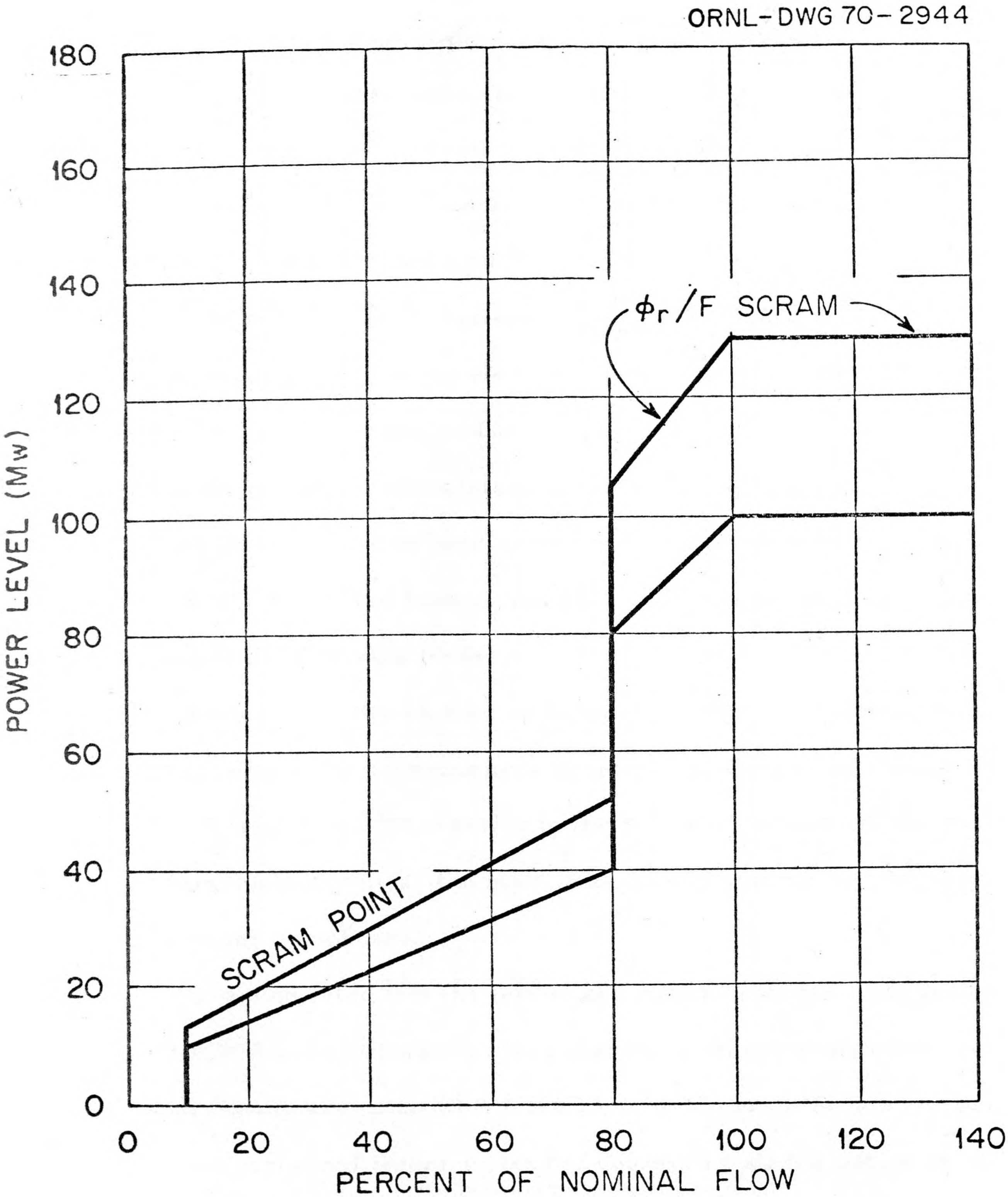


Figure 13. Modified characteristic of flow versus power.

3. The instrument power system which is supplied with current from the 13.8 kv feeder through shielded low-electrical-noise transformer or from normal emergency system Number 1.

4. The failure-free system, which is supplied with direct current from a bank of batteries.

The normal-power system consists of two 13.8 kv three-phase feeders which are run to the electrical building on separate sets of poles. The preferred feeder has a capacity of 10,000 kva. The alternate feeder is capable of maintaining the full plant load (7,000 kva) for about 2 hours and can handle one-half the load indefinitely. Upon failure of the preferred feeder, the 13.8 kv bus is automatically switched to the alternate feeder.

Power from the 13.8 kv switchgear is supplied to seven transformers. One of the transformers (Figure 13), designated substation Number 5, supplies 2400-v power to the primary coolant pumps, the secondary coolant pumps, the main pressurizer pumps, and the central chilled-water compressor.

It is suggested that the feeder and power circuitry be redesigned as shown in Figure 14 in such a way that two primary pumps and one secondary pump are connected to the 13.5 kv alternate feeder and the rest to the preferred feeder instead of connecting all the pumps to one feeder.

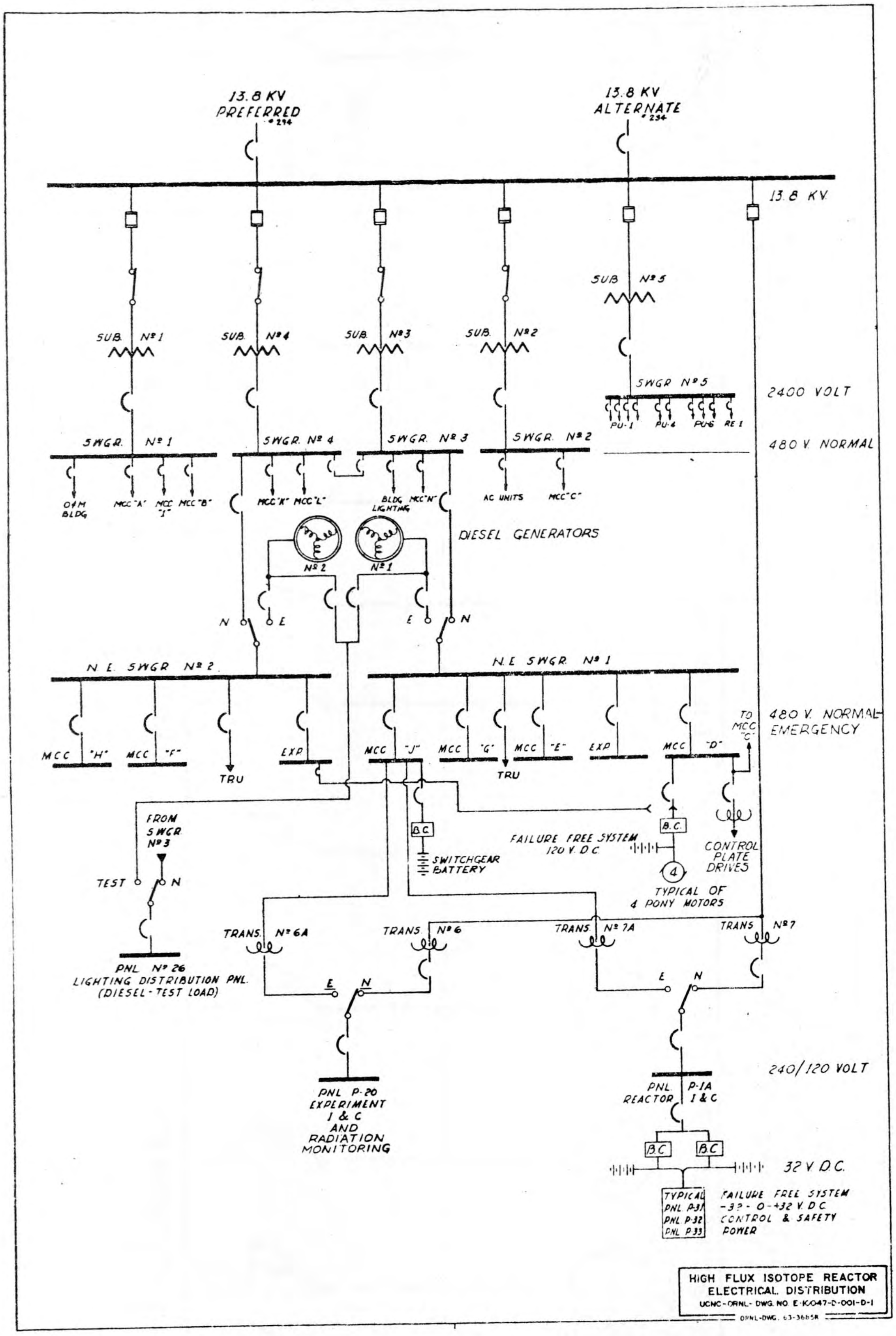


Figure 14. Electrical distribution.

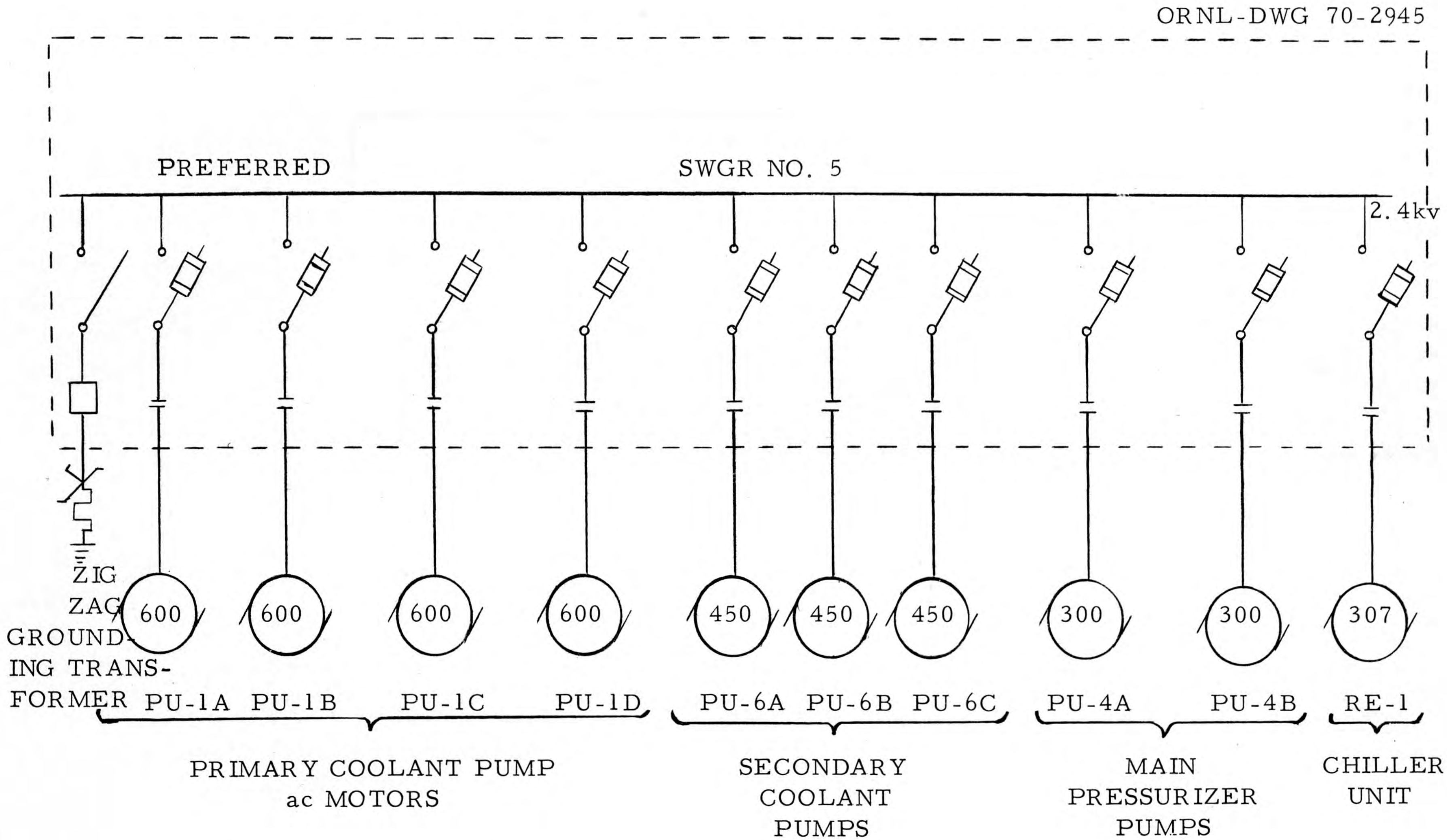


Figure 15. Normal power distribution.

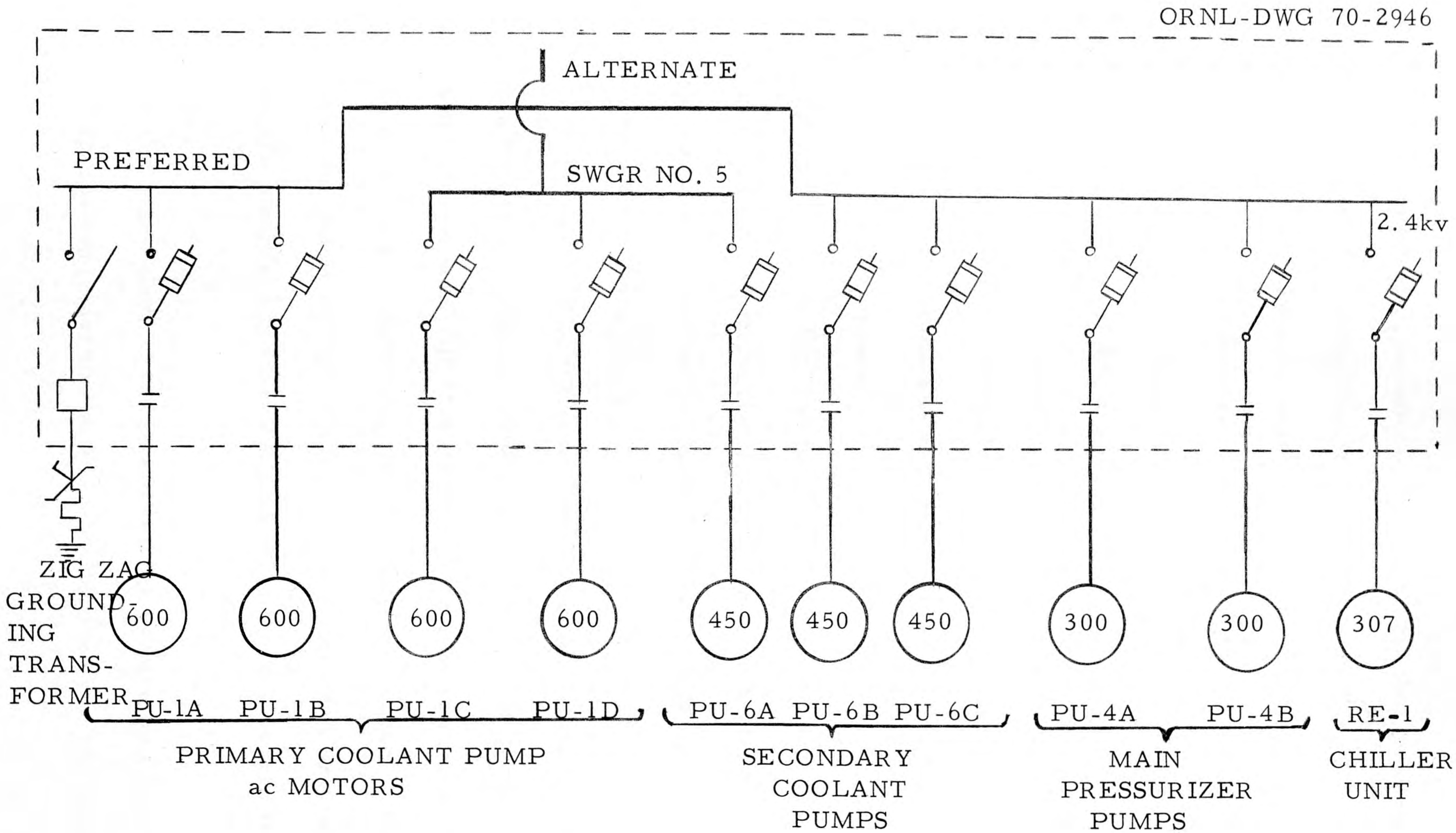


Figure 16. Modified power distribution.

In the following discussion it is assumed that power outage affects either the alternate feeder or the preferred feeder but not both. As mentioned before, only three of the four primary pumps are operating during normal operation. In the event power failure occurs, for instance, on the alternate feeder, and only one primary pump is receiving power from this feeder, then this primary pump and one secondary pump will shut down. On the other hand, since two primary pumps and a secondary pump are connected to the preferred feeder and these are still running at their full power, no scram action will take place.

Now let us consider a second case in which power failure still occurs on the alternate feeder, but this time instead of one primary pump there are two primary pumps running on the alternate feeder while one primary pump is running on the preferred feeder. If the duration of the power outage is short enough then there is a possibility that only one primary pump out of two running on the alternate feeder will shut down. Thus, overall, only one pump out of three will shut down which means no scram. If the duration of the power outage is long enough, then both primary pumps on the alternate feeder will shut down, and a scram will occur.

In the same way, if the power outage occurs on the preferred feeder, then the same argument and possibilities can be considered.

On the whole it can be concluded that the chance of a scram occurring is reduced by modifying the power supply circuitry as suggested above.

To avoid any possibility of a scram during a power outage as described above, we propose the fourth method as a combination of first and third methods. Once this arrangement is completed, no matter whether power failure occurs on the preferred feeder or on the alternate feeder, or whether the power outage lasts more than 2 seconds (assuming fluctuation occurs only on one of the two feeders), the coolant flowing towards the inlet temperature sensor will not reach the scram value.

CHAPTER IV

CONCLUSION AND RECOMMENDATIONS

From the above investigation we conclude that unnecessary shutdown of the reactor can be prevented if the proper modifications can be made. One of the possible solutions which is discussed requires modification of the primary coolant system. It seems that this method of eliminating scrams due to the hot slug problem might be rather expensive because of the existing design of the primary coolant system. However, there is another method which only requires modifications in the operating characteristics and which might as well serve the same purpose of unnecessary scram. The third possibility discussed involves redesigning the power circuitry. In this case a scram can still occur under a temporary power outage condition but the possibility is less compared to present condition. Finally, we conclude that a combination of two or more methods will definitely reduce the hot slug problem.

More detailed studies should be made before choosing or implementing any of the solutions suggested above. These would include an estimate of cost versus projected savings for each of the solutions.

CHAPTER V

LOCATION OF CALCULATIONS

All notes and calculations resulting from this investigation are on file at the Oak Ridge Engineering Practice School, Building 1000, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

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APPENDIXES

APPENDIX A

Appendix A

Characteristics of the high flux isotope reactor

Date of first operation 1965

Date of first operation at full power . . . . . . Sept. 9, 1966

Power . . . . 100, 000 Kw (thermal)

Core type . . . . Plates within concentric cylinders

Fuel . . . . 93% enriched uranium; Al-clad, U-Al cermet

Fuel load . . . . 9.4 kg of ^{235}U

Reactor vessel . . . . . Carbon steel, 20-ft high, 8-ft diam

Control rods . . . . . Concentric cylinders; europium oxide-aluminum, tantalum-aluminum, aluminum

Moderator . . . . Natural water

Reflector . . . . Beryllium backed by water

Coolant . . . . Water flow 16, 000 gpm through vessel

Average power density in core . . . . 2 Mw/liter

Thermal neutron flux (unperturbed) . . . .5 x 10^{15} n cm^{-2} sec^{-1}

Thermal neutron flux (with targets) . . .2-3 x 10^{15} n cm^{-2} sec^{-1}

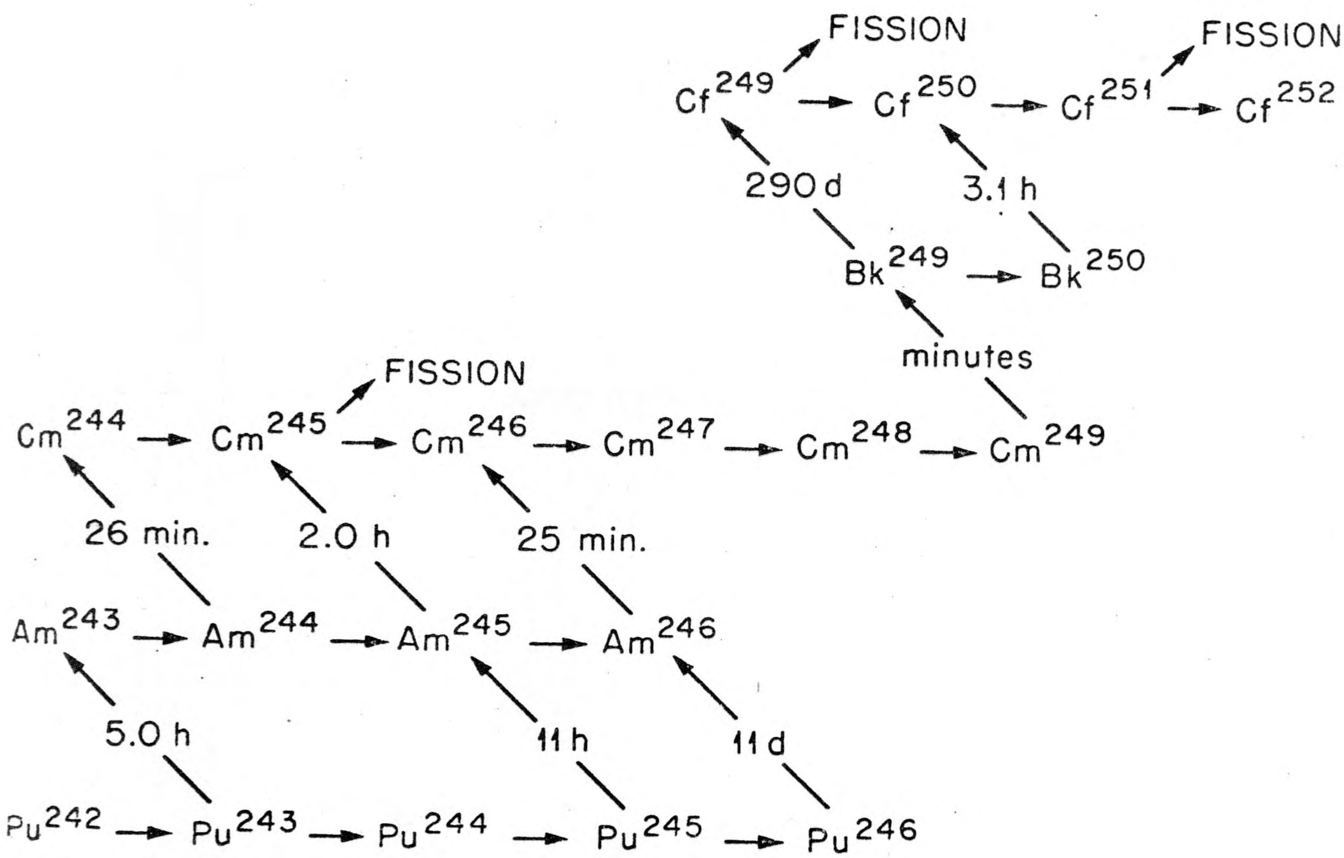
Function . . . . Transplutonium isotope production; research

APPENDIX B

Appendix B

Reaction paths in the production of Cf^{252} from Pu^{242}

ORNL-LR-DWG 50304

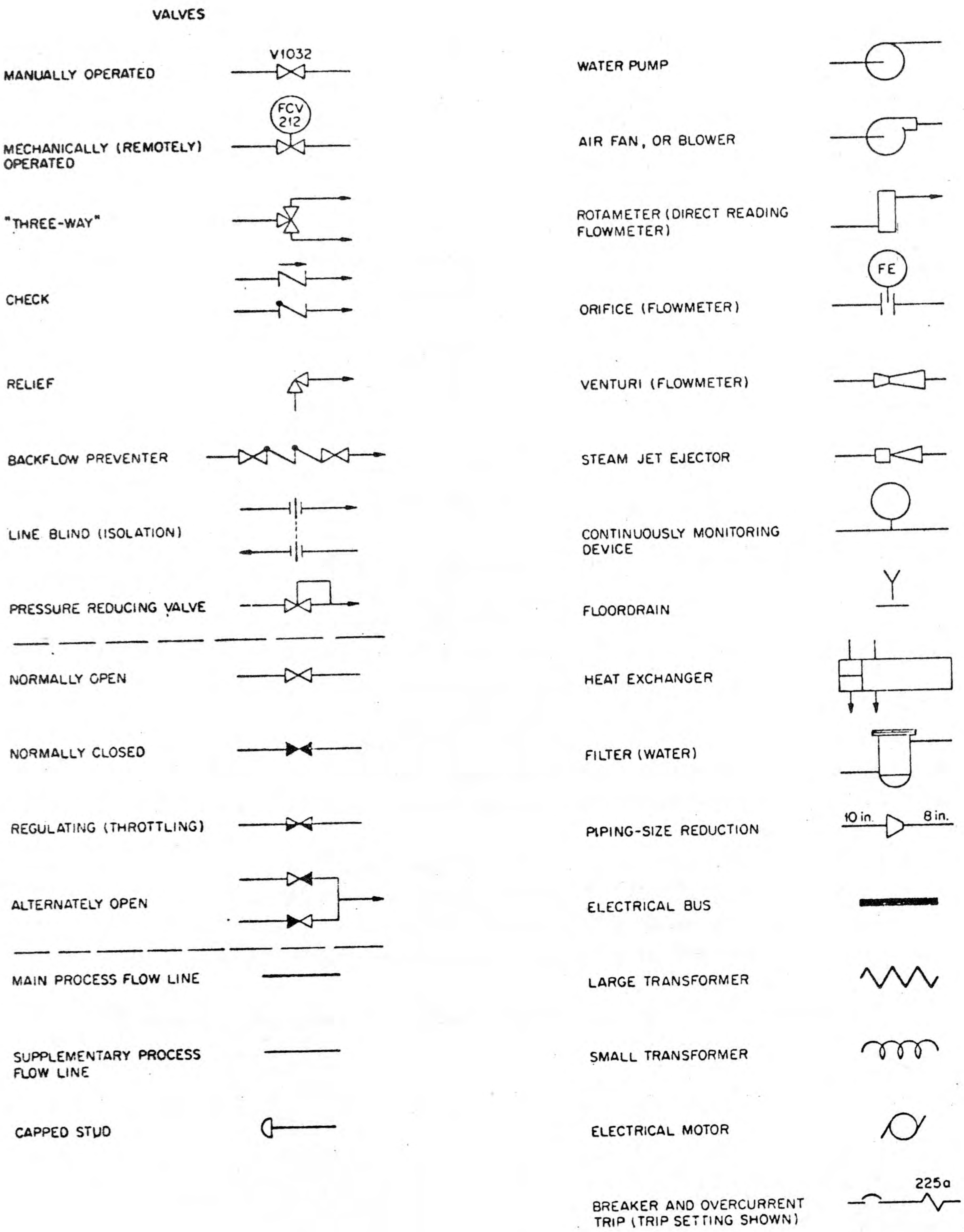


APPENDIX C

Appendix C

Drawing symbols

ORNL–DWG 63-950



VITA

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